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Beyond Iraq: The new challenges to the nuclear non proliferation regime*

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Abstract. The future of the nuclear non-proliferation regime is once again questioned. In the mid-nineties, the international community was thinking to be able to fix nuclear non-proliferation for some time. The instruments to prevent nuclear proliferation have been considerably strengthened and some have been added. An additional protocol has been adopted to give the IAEA, the means to detect undeclared activities, nuclear export controls have been reinforced, the NPT has been indefinitely extended, the CTBT has been open to signature and a mandate for a cut-off treaty adopted. But all hopes have vanished. At the dawn of the 21st century, the horizon is blurred. The pace of additional protocol is very slow. Forty-eight NPT countries have no safeguards agreements. Countries outside NPT will not join it soon. CTBTO will not enter into force in the coming years. The cut-off treaty is still in the limbo at the CD. At the same time, new worrying challenges to the IAEA safeguards and the nuclear non-proliferation regime as a whole have sprang up. North Korea has withdrawn from the NPT and is suspected to produce plutonium and enrich uranium for nuclear weapons. Iran is in violation of its undertakings and is suspected to conceal a nuclear weapon programme, Libya have unveiled and gave up nuclear ambition. An international black market of sensitive technologies from Pakistan has been uncovered. Aside, past negligence and resistance to the AP show up (South Korea, Brazil, and Egypt). Involvement of non state actors and nuclear terrorism is also a new development to deal with. How to tackle these new fearsome challenges? Some answers as PSI, UNSC resolution 1540, G8, have already been given. The answer is probably both political and technical, to allow the organisms in charge to recruit highly specialized experts and implement edge technique and to actually address the issue as the whole and to give the organism in charge the legal means, including addressing the missile proliferation, the reinforcement of the role of the Security Council, the creation of a new inspectorate. This document addresses technical and political aspects of the question.

1. Introduction

In the dawn of the 21^{st} century, for the second time in fifteen years, the future of the nuclear non-proliferation regime is once again questioned. New challenges have emerged which call for new answers. If the international community is unable to provide responses and agree on ways to strengthen it, the cornerstone which balances the thrust between the "have", the "have not" and the non parties will fall dawn. Eventually, the delicate equilibrium will collapse.

In the early nineties, the discovery of the clandestine nuclear programmes in Iraq and North Korea questioned the credibility of the non proliferation regime. The positive outcome of the crisis was the

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*the views expressed herein are those of the author and do not necessarily reflect the views of the CEA nor the French Authorities

This paper is a short and updated version of an article published in the book "Verifying Treaty Compliance – A new Scientific Discipline. Limiting the Spread of Weapons of Mass Destruction by R. Avenhaus, N. Kriakopoulos, M. Richard and G. Stein (Eds), SPRINGER adoption of a set of multilateral instruments as the Additional Protocol (AP, 1997) [1] to the safeguards agreements, supported by ad hoc monitoring and verification, aiming to strengthen the non-proliferation edifice globally, on the basis of the lessons gained and to curb nuclear proliferation, at least for some time. The AP greatly enhances the IAEA verification responsibilities, providing extended legal and technical means to detect undeclared activities. Rules of the Nuclear Supplier Group (NSG) on export of nuclear items have been reinforced to cover the transfer of dual use items (1992) and to make the Comprehensive Safeguards Agreement [2] with the IAEA a condition of nuclear trade. Negotiations for a Comprehensive Test Ban Treaty (CTBT) were completed in 1996 at the CD. In 1994, the CD adopted a mandate for the negotiation of a cut-off treaty to ban the production of fissile materials for nuclear weapons. The 1995 Review Conference extended indefinitely the NPT and adopted the principles and objectives for nuclear non proliferation and disarmament.

In ten years, the prospects have shifted from hope to pessimism. The hopes seem to have vanished. The horizon is getting blurred. The implementation of the Additional Protocol is slow. Thirty NPT countries have still not concluded a Comprehensive Safeguards Agreement (CSA). The three countries outside the NPT, Israel, India and Pakistan are not expected to join the NPT in a foreseeable future (as Non-Nuclear-Weapon State, NNWS) though India seems to get closer to the international norm as requested by the recent US India nuclear cooperation agreement. The entry into force of the CTBT is not expected in the coming years; in 1998, India and Pakistan conducted nuclear tests. The negotiation of a cut-off treaty has not yet started at the CD.

New disturbing challenges have sprang up. Clandestine nuclear programmes have been discovered in several NPT countries. After ten years of dispute with the IAEA and the international community, North Korea (DPRK) has broken the "agreed framework" and resumed its nuclear programme, broken relations with the IAEA, withdrawn from the NPT. DPRK is strongly suspected of producing plutonium and high enriched uranium for nuclear weapons while improving its ballistic missile capabilities. Iran has been caught in serious violation of its obligations and is suspected of concealing a nuclear weapon programme under the cover of a civilian nuclear one. Iran is also improving its ballistic missile capabilities. The good news is that Libya gave up forever any WMD ambition. The bad news was the unveiling of an extended foreign assistance in building Libya's uranium enrichment programme. Investigations have revealed the existence of an international clandestine procurement network based on a black market of sensitive technologies headed from Pakistan by A. Q. Khan. This discovery highlighted serious loopholes in the export control of sensitive technologies. Moreover, "negligence" in declarations of past activities and resistance to the IAEA strengthened safeguards implementation have shown up (South Korea, Brazil, and Egypt).The awareness of nuclear terrorism after the 9/11 terrorist attack and possible involvement of non-state actors which could use radioactive materials to build a radiological dispersion device (RDD) or fissile material for a crude improvised nuclear device (IND) or even steal a nuclear device, have shown up adding to the gloomy background of the non-proliferation regime's crisis. The threat of nuclear terrorism then appeared as a major one that needed to be specifically addressed through new means.

Now, the question is how these new challenges could be tackled. New bricks have been added to reinforce the edifice to combat WMD proliferation and terrorism. Answers to some aspects of the issues have already been given by UNSCR 1540, the G8 Global Partnership, the IAEA nuclear security programme, the UN convention on nuclear terrorism, the Proliferation Security Initiative (PSI) [3] the amendment of the SUA convention [4] and the Convention on Physical Protection of Nuclear Material (CPPNM) and recently, the United States Russia Global Initiative to combat nuclear terrorism (July 2006). However, an appropriate answer requires a global approach at political, diplomatic, legal and technical levels supported by an efficient monitoring and verification system. This approach should definitely draw upon the lessons and experiences gained in dealing with all the major events of last fifteen years.

2. A look backward: A half century of progress

*2.1. **Awareness of an arms control and non-proliferation policy***

Since the fifties the international community has done a great deal of work to develop protection mechanisms against and prevent the proliferation of nuclear, chemical, biological and radiological weapons (WMD) and their means of delivery through arms control and disarmament agreements. In recent years and especially in the aftermath of September 11, the international community has increased its to combat nuclear terrorism. The fear of nuclear weapons and their huge power of destruction raised the vital need to stop their dissemination and control the spread of sensitive materials and technologies without hindering the applications of nuclear energy. A network of legal instruments has been set up relying, more or less, on monitoring and verification systems. This "fearful dilemma" inspired the speech of President Eisenhower "Atom for Peace" fifty years ago (8 December 1953) [5] and gave birth to the IAEA [6] (1957). The missile Cuban crisis (1962) triggered the beginning of US –Soviet talks on nuclear arms controls and disarmament. At the start of the IAEA, only very light verification provisions have been accepted by the Member States [7]. Safeguards agreements have been slightly strengthened during the sixties. INFIRC/66/rev.2 agreement (1968) covers the front and back ends of the nuclear fuel cycle [8]. It remains the one governing the nuclear safeguards in the three States non-parties to the NPT, Israel, India and Pakistan.

*2.2.**The milestone of the NPT as the birth of verification***

The prospect of having more than 20 or 30, states possessing nuclear weapons, at the end of the 20th century and the increasing risk of nuclear war lead the United States, the Soviet Union and the United Kingdom to agree on a treaty to stop the proliferation of nuclear weapons. Only the five having tested a nuclear weapon prior to the 1st January 1967 have the legal right to have nuclear weapons. All others, by joining the NPT renounce to have nuclear weapons in return, of the access to peaceful use of nuclear energy and technologies and the undertakings from the five to pursue negotiations on nuclear disarmament [9], freezing the division in “Nuclear Weapon States” which have and “Non Nuclear Weapon States” which have not. The indefinite extension of the NPT in 1995 confirmed indefinitely this division which is still ruling all nuclear issues.

*2.3. **Iraq: the non proliferation regime challenged and strengthened***

2.3.1. 1992: A milestone for compliance monitoring and verification

In the early nineties the discovery of the clandestine undeclared nuclear weapons programme of Iraq, the opposition of North Korea to the verification of the IAEA and the decision of South Africa to reveal and dismantle its nuclear weapon arsenal triggered the strengthening of the non-proliferation regimes [10] and initiated discussion on new instruments (CTBT, Cut-Off,…) to curb horizontal and vertical proliferation of nuclear weapons. Loopholes in the instrument to detect and thwart undeclared nuclear activities and materials, to prevent illicit trafficking of sensitive technologies and stop clandestine nuclear weapon programme have been identified. Improvements to the regime for the detection of undeclared nuclear activities and undercover technology transfers have been set up with the additional protocol to IAEA safeguards agreements [1] and the establishment of the dual use items list of the Nuclear Supplier Group (NSG) [11]. At the same time, former Soviet Republics agreed to joint the NPT, the NPT has been extended indefinitely, and new instruments have been set up as the Comprehensive Test Ban Treaty (CTBT) or foreseen as the Fissile Material Cut-off Treaty.

2.3.2. Value of the Iraqi and North Korean experience.

Building on the lessons drawn from the implementation of the implementation of UNSCR 687 and following and the ongoing, monitoring and verification plan (OMV) [12] by UNMOVIC and INVO [13], advanced technologies have been identified as necessary to fill gaps in safeguards verification [14]. Iraq was a real "test-bed". From 1992 to 1998, in the framework of the extensive rights of verification provided by the OMV plan, new inspection approaches were developed and applied. Iraq was a country-wide, open field laboratory where many advanced monitoring technologies as well as new verification approaches were implemented allowing participating states and organisations to test new monitoring systems, new verification concepts and new technologies as well as to conclude new legal instruments, all of them proved valuable in dealing with verification issues in Iran and Libya. Some examples are the local and wide area environmental sampling techniques, the extensive use of overhead imagery, aerial and satellite (Figures 1 & 2), the development of open-source analysis and the use of third party information, the in-depth analysis of nuclear fuel cycle, the continuous (remote) monitoring of processes or equipments, and the development of new inspection techniques: radiation monitors, geophysical survey techniques (figures 2 & 14), interview of key personal, search for import-export information (see 4.2). As important as the collection of techniques and methods, the most important development, essential to forming a coherent picture of a clandestine complex and extended programme as a whole, was the analysis and fusion of all type of information. It was not only a technological advance; it was also a methodological one.

Iraq/Overhead imagery

In the framework of the specific context of Iraq disarmament, the U2 plane of operation "Olive Branch" flew over Iraq under UN flag for a permanent survey of Iraqi sites and search for undeclared sites and activities

Source UNSCOM

FIG. 1.

Iraq/Ground penetration radar testing

Fielded for a primary testing, a helicopter born ground-penetrating radar (antennas on each side) was deployed during UNSCOM operation "Cabbage Patch" to seek for buried SCUD missiles (1993).

Source private communication

FIG. 2.

The provisions contained in the OMV Plan goes far beyond what could be accepted by a State under a strengthened safeguards agreement (anytime, anywhere, access to any document, implement any technologies and anybody for interview. Although the Iraqi experience was unique, it is worth noting that it can be used to draw valuable lessons and translate them into manageable inspection practices, i.e. interview of key personals or visit to military sites, to allow the Agency to cope with very complex and difficult situations. In one of his reports to the Board of Governors on the implementation of NPT Safeguards Agreement in Iran, the Director General of the IAEA noted that, " *Given Iran's past concealment efforts over many years, such transparency measures should extend beyond the formal requirements of the Safeguards Agreement and Additional Protocol and include* ***access to individuals, documentation related to procurement, dual use equipment, certain military owned workshops and research and development locations***" [15] .

3. Current situation and prospect

3.1. ***Dawn of the century: Resuming challenges***

If in the middle of nineties, the international community could have been confident on its ability to curb proliferation of nuclear weapons. But, in few years the hope raised of stemming rapidly the proliferation of nuclear weapons in the aftermath of the Iraq and North Korean crisis, has fade away for some time. At the dawn of the 21st century new crisis appear that call for urgent measures to protect the peace and security. First, after the terrorist attack of September 11th a multiform terrorist threat from states of concern and sub-state groups emerged with the possible use of Weapons of Mass

Destruction. Reponses have been given by the adoption of the nuclear security programme of the IAEA and the establishment of national anti terrorism programme and cooperation between them. Then, it was the crisis induced by the violation by Iran, North Korea and Libya of their international commitments and consecutively the unveiling of a nuclear black market leaded by Pakistan. In the same time, the international community is in a conflict mood between those states which want to access freely to nuclear technologies as NPT parties and CSA states and those states which want to restrict the access to sensitive technologies. This conflict is concretised by the inability of the 2005 NPT Review Conference to agree on measures to redress the situation. In consequence of that mood, the pace of implementation of instrument like the Additional Protocol is very slow and somehow disappointing.

*3.2. **Prospects: A blurred vision of the future***

Now the view on future of international security seems blurred. The pace of additional protocol is very slow. Thirty NPT parties have not yet concluded a comprehensive safeguards agreement with the Agency. The three countries outside the NPT will not join it in a foreseeable future. The CTBTO will not enter into force in the coming years. The Cut-off treaty is still in the limbo at the CD. At the same time, new worrying challenges to the IAEA safeguards and the nuclear non-proliferation regime as a whole have sprang up, Iran Libya, North Korea once again and other minor crisis induced by resistance to the verification and past breaches. If these crises are not to be solved quickly as the Iran seems to drawl out, they may undermine significantly confidence of the States in the capacity of the international community to maintain the security by preventing the dissemination of nuclear weapons and nuclear and radioactive materials to state of concerns and sub state group.

4. Current situation: Overviews of concerns

Iran, Iraq, Libya and North Korea have all joint the NPT, the three of them at it earliest. Since the sixties, all these countries except Libya have developed or try to do it, all steps of a nuclear fuel cycle at diverse degrees: mines, uranium concentration, conversion, uranium enrichment, reactors plutonium extraction, often with foreign or international assistance. But they have also taken advantage of this assistance and the weaknesses of the then AEA Safeguards to carry out undeclared activities to end up to get nuclear weapons. Iran has been caught in significant violations of its undertakings and is suspected to conceal a nuclear weapon programme, Libya while giving up its nuclear weapon ambition has unveiled a multinational concealed procurement's network. An international black market of sensitive items organized and managed from Pakistan has then been discovered. As consequence of the additional protocol implementation, past negligence and resistance to the inspection have shown up (e.g. South Korea, Brazil, and Egypt).

*4.1. **Iraq: no longer a proliferation threat, but?***

The disarmament, the dismantlement of Iraq's clandestine nuclear programme and the monitoring of Iraq's nuclear activities under Security Council Resolution (UNSCR) 687 and following was a consequence of its discovery after the Gulf war in 1991. Before 1991 Iraq was not subject to any particular monitoring and verification under its comprehensive safeguards agreement and was able to develop unbothered its clandestine nuclear weapon programme.

Iraq
Al Atheer: weaponization site
Source IAEA

FIG. 3.

Iraq: aluminium tubes suspected to be use as part of centrifuge but proved being rocket bodies
Source IAEA

FIG. 4.

After 1998 and 7 years of extensive investigation, the IAEA/Action Team (INVO) has acquired an in-depth knowledge of all Iraq's clandestine nuclear programme nuclear activities and set up the ongoing monitoring [16], though some points remain to be clarified in particular on foreign assistance, procurement and know how in centrifuge technologies and nuclear weapons design (Figures 3 & 4). Did the A.Q. Khan's network provided Iraq with drawing as it does for Libya and possibly Iran [17]? Currently, in the aftermath of the second Gulf war, ***the threat of Iraq as a state is over. But now, the country has fallen into a bloody chaos and has become a heaven for the Islamic terrorists who may fuel the threat of WMD terrorism.***

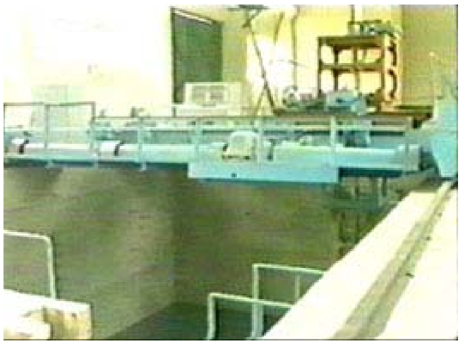
*4.2. **North Korea: The present threat?***

The first clash took place in 1992 when the IAEA came to verify the North Korea initial declaration under its safeguards agreement. Thanks to the US satellite imagery, the Agency discovered an undeclared radiochemical laboratory at the Yongbyon and inconsistencies in the declaration. The laboratories turned out being a plutonium separation facility (Figure 5). North Korea refused to grant IAEA access to its facilities. The UNSC was seized in 1994. Then North Korea withdraws for the first time from the NPT. After an agreement between the United States and the DPRK, all its suspicious nuclear activities where frozen, the IAEA being in charge of the verification of the effectiveness of the freeze. In 2002 after years of dispute between IAEA and DPRK on the verification, came a new crisis with that DPRK has declared to be capable to enrich uranium which was suspected since some time. Then, North Korea has broken the relation with the IAEA, expelled inspectors, broken the "freeze" agreement resuming operation of its 5MW_e plutonium production reactor, declared to reprocess spent fuel under control (Figure 6) and withdrawn from the NPT. The UNSC has been seized since 2003. Six parties talks has been engaged in the negotiation of a new "freeze" but they are stalled because Russia and China does not seem to be hurried to take North Korea weight off the United States shoulders. ***Now, North Korea is assumed to possess several nuclear weapons and is threatening to conduct a nuclear test.***

DPRK/Yongbyon reprocessing plant

Source ISIS

FIG. 5.



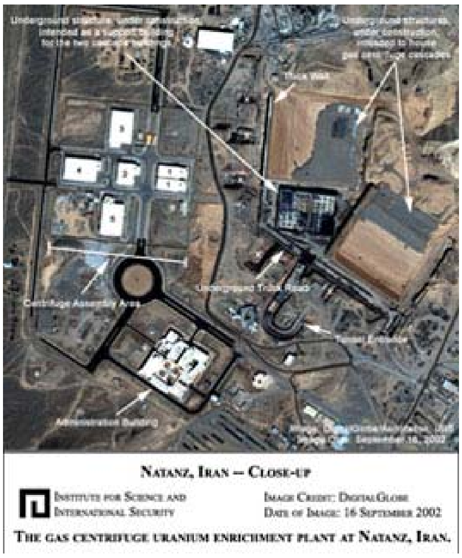
DPRK/Yongbyon, the spent fuel pond

Source IAEA

FIG. 6.

*4.3. **Iran: the threat for to morrow?***

In 2003, following information from an opposition group, IAEA inspectors discovered a large undeclared gas centrifuge uranium enrichment programme including the construction of a large plant and a pilot plant at Natanz (Figures 7 & 8) and that the centrifuge technology was provided through the network headed by Pakistani scientist A. Q. Khan as for Libya. Also they discovered that Iran was planning to construct an heavy water research reactor, well suited for plutonium production with the associate heavy water production plant at Arak, that it had imported an important stock of undeclared UF_6 a part of which has been converted to uranium metal at Esfahan (Figure 9 & 10), that it had irradiated uranium metal targets and extracted some quantities of plutonium and bismuth targets to get polonium-210 which could be use as neutron trigger in a nuclear weapon. As high enriched uranium particles on centrifuge parts has been highlighted by environmental sampling analysis, Iranian have not yet provide clear explanation arguing that the parts have been contaminated in the supplier country, Pakistan. Suspicion of undeclared activities has been raised for two sites, Parchin (Figure 11) and Lavisan-Shian (Figure 12) where the Iranians are very reluctant to grant access. Atop, IAEA's inspectors investigating on the 1987 and mid-1990's A. Q. Khan's offers discovered a document describing the procedures for the reduction of UF6 to uranium metal and the casting and machining of uranium metal into hemispheres, operation clearly related to the manufacture of nuclear weapons. Iran has yet to provide the Agency with a copy of that document and clarification.



IRAN / NATANZ
Fuel Enrichment plant (FEP) satellite view
Credit ISIS
FIG. 7.

IRAN/NATANZ
Fuel Enrichment Plant (FEP) inside view
Capacity 150 t SWU/year for 54 000 centrifuges
Source Web
FIG. 8.

IRAN/ESFAHAN
Research Centre
Credit AFP
FIG. 9.

IRAN/ESFAHAN
Uranium conversion facility (UCF).
Credit AFP
FIG. 10.

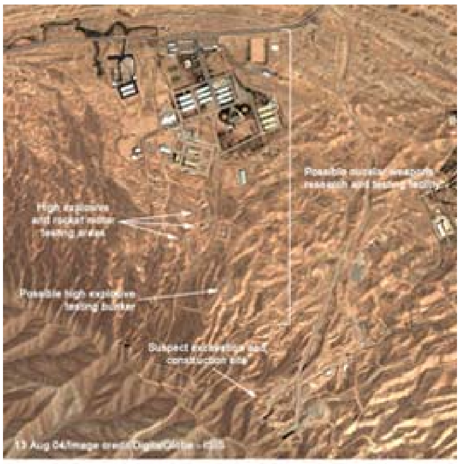
The absence of declaration of these activities to the IAEA constitutes a serious violation of the safeguards agreement. After more than three years of intensive verification the IAEA is still unable to clarified uncertainties related to the scope and nature of Iran's nuclear programme and the existing gaps in knowledge continue to be a matter of concern. Iran's cooperation and transparency has decreased and the additional protocol measures are no longer provisionally implemented. Iran continues to procrastinate and responds reluctantly to the IAEA requests. Discussions between the six (the P5 + Germany) and the United Kingdom (EU3) and Iran have revolved around the suspension of the uranium enrichment activities by Iran in exchange for assurances of supplies and other commercial benefits. Iran has rejected the proposal and resumed its enrichment activities. The construction of the heavy water reactor is still continuing. In spite of the international community efforts, the prospect of a successful resolution of the crisis seems more and more unlikely. Then, following a resolution of the Board of Governors, the DG of IAEA has transmitted the file to the United Nations Security Council;

Lavizan-Shian Site, Iran -- May 10, 2004

Lavizan-Shian Site, Iran -- August 11, 2003

IRAN/LAVISAN :A site of concern: What kind of activities has been carried out before remediation?
Credit ISIS

FIG. 11.



Overview of select facilities within Parchin, possibly involved in nuclear weapons research, testing; and possibly production, and an area of suspicious excavation and construction whose purpose is a subject of debate.

IRAN/PARCHIN: Missile and explosive test range, another site of concern.
Credit ISIS

FIG. 12.

The Iranian case raises many questions about the capacity of the IAEA and the States intelligence services to detect concealed enrichment activities, about the control of transfer of sensitive nuclear technologies, about the right for States to develop nuclear fuel cycle technologies under article IV of the NPT and about the role and responsibilities of non-NPT States such as Pakistan in the fight against proliferation. ***As it is clear that Iran's nuclear programme aims to get a nuclear weapon, then is Iran trying to gain time before withdrawing from the NPT as soon as it possesses it?***

4.4. ***Libya: A success story?***

In October 2003, the seizure of maritime transport of centrifuge components occurred and prompted the decision of Kaddafi to give up its WMD capabilities and to allow experts from United States and United Kingdom to visit its weapon facilities. On December 2003, Libya, after nine months of secrete talks with American and British officials, agree to destroy all its WMD capabilities, Nuclear, Chemical and Biological, to abide by the NPT, to allow immediate inspection and monitoring and to conclude an additional protocol. It appeared that Libya has develop mobile facilities for uranium enrichment and has acquired centrifuge components and know how from the nuclear black market network of the Pakistani A. Q. Khan, But it appear also that Libya has acquired drawing of a Pakistani nuclear weapons from the A.Q. Khan network which raise question about other countries which could have benefited of the same information (Iran, North Korea, Syria…?).

4.5. ***Pakistan and the proliferation "bazar"***

While investigating the nuclear programme of Libya, the IAEA discovered that the international black market of sensitive items, centrifuge technology and know-how, centrifuge components, Pakistani, drawings of nuclear weapon design organized by the A. Q. Khan network extended over many continents, Africa, South Asia and Middle East and countries among which there were several European ones and involved individuals and companies. Moreover, there is well founded suspicion that not only Libya and Iran but also North Korea and possibly other countries have benefited from the network. The network headed by A. Q. Khan is probably the most important proliferation issue of the beginning of the 21st century. High priority should be given to resolving the problem because it is closely linked to the other problems. The overall extend of the ramifications of the existence of the network and the nature of the trafficked items is still unknown. A. Q. Khan confessed that centrifuge technology was transferred to North Korea, but in exchange of what? Did A. Q. Khan pass nuclear weapon files to countries other than Libya? And what about missile technology undercover transfers?

4.6. *Other concerns: Korea, Brazil, Egypt*

As consequence of the Additional Protocol and strengthened safeguards implementation, past undeclared activities, some very questionable, have come to light with the extended declaration; at the same time, increased resistance to more intrusive inspections developed. In 2004, Brazil refused to allow IAEA inspectors to access the cascade hall of its uranium enrichment facility at Resende to protect proprietary information. As Brazil wants to develop nuclear energy and become a recognized producer of LEU, arrangement has been found between Brazil, IAEA and ABACC but ambiguities still remain. In 2004 again, The South Korean government has admitted to the International Atomic Energy Agency that a group of scientists secretly produced a small amount of high enriched uranium near weapons grade. It also admitted to have separated a small quantity of plutonium. All these activities were not declared to the IAEA. In December 2004, Egypt acknowledged that, between 1990 and 2003, it had conducted experiments involving the irradiation of small amounts of natural uranium in its reactors to test the production of fission product isotopes for medical purposes, and that it had not reported these experiments to the Agency.

4.7. *India*

In July 2005, a joint statement between United States President G.W. Bush and India Prime Minister Manmohan Singh followed in September by a joint statement between the French Republic President and India Prime Minister opened the door for civil nuclear cooperation between India and advanced nuclear countries as United States and France to fulfil huge India energy needs. But India is one non NPT countries, it possesses a nuclear weapon arsenal and has conducted nuclear test (1974, 1998) and their no hope it will joint the NPT in a foreseeable future. But, on the other hand, India could be granted of a "rather" good non proliferation record (unlike Pakistan). The US-India joint statement call for a clear cut between nuclear facilities dedicated to military purposes and the civilian nuclear fuel facilities, the conclusion of a new safeguards agreements and additional protocol with the IAEA and support to Cut-off treaty at the CD. Nuclear trade with India requires changing the rule of the NSG and US national legislation. In one hand, India will get closer to international nuclear norm and such a important country cannot be keep apart. On the other hand, India's proposal of separation is disappointing because some important facilities remain dedicated to the production of nuclear weapons material and transferring up to date sensitive nuclear technology could in the future turn out contributing to the nuclear arsenal modernisation.

4.8. *The threat of nuclear terrorism*

In 2001, the 9/11 terrorist attacks and the awareness of the emergence of a new threat accelerated the work on the prevention of terrorist acts and the mitigation of their consequences. The 9/11 attack and the development of sub state group activities in a context of regional crises (Afghanistan, Iraq, Iran, Chechnya, former Soviet Union Republics) have increased the importance of measures to prevent terrorist actions. To this end, the international community has developed new instruments and strengthened existing ones as the amendment to the Convention on the Physical Protection of Nuclear Material (CPPNM, 2003), the adoption of the Security Council Resolutions 1373 (2001) and 1540 (2004), the United Nation International Convention for the Suppression of Acts of Nuclear Terrorism (2005) and the nuclear security programme set up by the IAEA (2002) to help member States improve the security of their nuclear and radioactive materials and of the installations which contain them.

5. How to answer the new challenges?

To strengthen the nuclear non-proliferation regime to face the new challenges, the approach should be both a political one with the definition and effective implementation of new tools and a technological one to provide these tools with the best technologies in order to enable them to respond efficiently to these challenges. These developments should draw upon the experiences of the past (Iraq, North Korea) and the present (Iran, Libya).

*5.1. **New tools? Why and for what?***

5.1.1. Addressing the proliferation of the nuclear weapons and the nuclear disarmament

To address effectively the proliferation of nuclear weapon and answer the new challenges, the international community needs to complete the legal framework it had started to set up ten years ago as stated in NPT 1995 Conference decision 2: the Additional Protocol and the Comprehensive Safeguards Agreements as universal standards for non-nuclear-weapon States, entry into force of the Comprehensive Test Ban Treaty, negotiation of a Fissile Material Cut-off Treaty, strengthening of the nuclear export rules and to supplement it by initiative and tools which prevent the dissemination of sensitive nuclear technologies as enrichment and reprocessing.

5.1.2. Addressing the traffic of sensitive technologies

The unveiling of the extension and range of activities of the A.Q. Khan multinational nuclear technology transfer network, prompted discussions on how to improve the export control rules as they apply in particular to dual-use items and guidelines in the Nuclear Suppliers Group (NSG) and on how the IAEA and the safeguards system could better control the transfer of sensitive item by means of the Additional Protocol. International legislation has already been adopted to outlaw proliferation of WMD by implementing United Nations Security Council Resolution 1540 and to block the illicit trade with the Proliferation Security Initiative (PSI) by a group of concerned States and to amend the Suppression of Unlawful Acts at Sea Convention (SUA, see introduction) at United Nations International Maritime Organisation (IMO).

Drawing a lesson from the discovery of Libyan and Iranian undeclared enrichment programme and the role of the A.Q. Khan black market network, discussions has been conducted under the auspices of the IAEA DG on a multilateralisation of the nuclear fuel cycle and assurances of supply to prevent the dissemination of sensitive nuclear technologies and keep them under the control of the international community [18].

5.1.3. Addressing the Security of nuclear and radioactive material

In the aftermath of the 9/11 terrorist attack the international community became aware of the risk posed by the potential use of radionuclide materials as radioactive sources or spent fuel not under adequate protection, for a radiological attack, in particular materials stored in the former Soviet Union. Actions have been taken in the framework of the G8 within the Global partnership, Global Threat Reduction Initiative on radioactive sources and amendments to strengthen the Convention on the Physical Protection of Nuclear Material (CPPMN), the IAEA nuclear security programme and the code of conduct on the safety and security of radioactive sources.

5.1.4. Addressing the proliferation of means of delivery

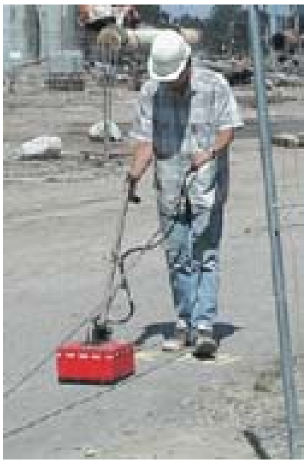
The question of means of delivery, ballistic missiles, cruise missiles and other means such as unmanned aerial vehicle (UAV) needs to be dealt together with the question of nuclear weapon proliferation or the threat of terrorism. Iran, Iraq, North Korea and Libya, all have acquired or developed various means of delivery adapted to their security environment. Delivery capabilities should be an important element in the evaluation of State proliferation potential "as a whole": 1) The possession by a proliferating state or a terrorist group of means of delivery able to carry WMD reinforces considerably the credibility level of the threatening capability; 2) Ballistic missiles have become an exchange currency in the proliferation "bazaar". It is now assumed that North Korea (and possibly other countries) has exchanged missile technology for centrifuge technology with the A.Q. Khan network; 3) Instruments to prevent the illicit export of missile technology are only on a voluntary basis. The Missile Technology Control Regime (MTCR) should be strengthened to include new means of delivery and the adherence to the Hague Code of Conduct (HCoC) promoted.

5.2. *How advanced technologies can contribute to tackle the threats?*

To be efficient and credible a non-proliferation or disarmament international instrument such as the NPT/IAEA Safeguards, CTBT, CWC, etc, should rely on efficient and credible verification tools that are capable of producing quick, un-ambiguous and reliable assessment of difficult situations (as it has arisen with the implementation of safeguards in Iran). Most of the time, investigation of suspicious activities has to be made in a hot political context and assessment of events has to be delivered within a short time period. The assessments have to be un-ambiguous, reliable, and trustworthy and, in cases of non-compliance, the evidence should be convincing; of course, efforts should be made to minimize the cost of the investigations. An organisation in charge monitoring for and verifying compliance with the non-proliferation commitments needs to a "tool box" of the most sensitive and reliable technologies. It also needs the assistance of the member States and their strong support in research and development. The importance of technologies for IAEA safeguards implementation, on-site inspections in a future CTBT and other treaties discussed extensively by a number of contributors to this volume; some examples of applications of these technologies are shown in Figures 11 & 12. satellite imaging for assessment of Iran sites Natanz, Parchin, Lavisan, Esfahan, environmental sampling and ultra-traces analysis for enriched uranium contamination of centrifuges parts in Libya, Iran and Pakistan (Figure 13), information collection and analysis for assessment undeclared past activities in Korea and Egypt, remote monitoring and advanced inspection equipment such as ground penetration radar to carry out design information verification (Figure 14). ***However, without a strong political commitment, the implementation of the best technologies is useless***.

Environmental monitoring
Ultra-traces analysis of environmental samples at the DASE clean laboratory
Source CEA/DAM/DASE

FIG. 13.

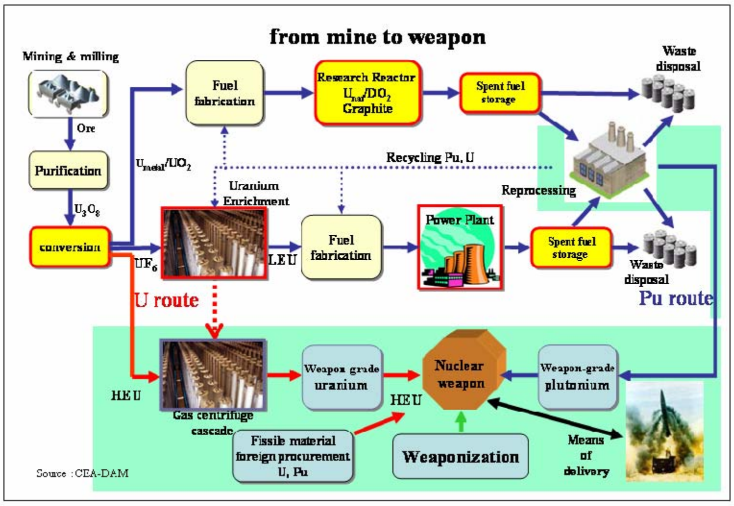


Geophysical survey
Ground Penetration Radar for DIV
Source CEA/DAM/DASE

FIG. 14.

5.3. *A focus on nuclear fuel cycle and materials* *[19]*

The discovery in the past few years that several non-nuclear-weapon States were able to conceal for decades research and development activities on sensitive technologies, procurement or manufacture of related equipment (e.g., centrifuge components, laser enrichment parts), often through a clandestine procurement network of technology and know-how, and fissile material production activities, (as UF_6, U metal, UHE, plutonium, Polonium) mainly related to uranium conversion and enrichment and reprocessing, have raised new concerns and call for new types of responses.



Diversion of nuclear fuel cycle sensitive technologies

Source CEA/DAM

FIG. 15.

The findings of the IAEA have highlighted the possibility of a scenario where one non-nuclear-weapon State masters centrifuge enrichment technology or laser enrichment technology and constructs and operates an enrichment facility for peaceful purpose under the umbrella of IAEA, possibly with foreign open assistance and parallel, the State constructs an undeclared facilities at a concealed site, using the acquired know-how and operates this facility to manufacture nuclear weapons components (Figure 13). At a later time, the State may decide that the time is ripe to dismiss its commitments and to withdraw from the NPT with a three-month notice under the provisions of NPT article X. The concern created by this possible scenario has generated several proposals to address the problem; among these are the proposal of President Bush to not allow access to sensitive technologies to State which do not already have a complete nuclear fuel, the idea put forward by the Director General of IAEA that sensitive facilities should operate under international control or the French proposal based on adherence to the additional protocol and several other criteria. ***The capacity of the IAEA to detect concealed sites hosting undeclared uranium enrichment/reprocessing activities and to develop adequate technological means for doing so should be addressed and improved*** in the framework of the IAEA Safeguards Research and Development Programme, Member States Support Programmes (MSSP) and recommendations of the SAGSI and the Committee 25.

5.4. An EU security prospective

The European Defence & Security Policy has taken into account the threats of proliferation of weapon of mass destruction and hyper terrorism. Following the Thessaloniki summit joint declaration, the European strategy has been spell out in the document "A secure Europe in a better world". Actions on technologies research and development have been launch as the "Preparatory Action in the field of Security Research and development (PARS) to prepare the future and support international organisation [20].

6. Verifying compliance is the key but how to enforce it?

6.1. Compliance

Whatever the potential efficiency of the new instruments and of the strengthening of the ones in forces, the international community will decide and the power of the range of technologies developed to support them, all these efforts could be meaningless if there is no way to redress non compliance once it has been establish. In that perspective several issues are developed hereafter.

*6.2. **Verifying compliance/The needed authority and resources***

Compliance can only be verified effectively to the degree that States honour their political commitments. The organisation which is charged with the responsibility to verify compliance should be given the necessary authority from the international community to do so. For example, the IAEA in implementing the Comprehensive Safeguards Agreements and Additional Protocols should have the appropriate authority to be able to carry out verification activities at suspicious sites or facilities while respecting the sovereignty of the State. The major challenge of organisations like the IAEA is confronted is to be able to monitor commitments compliance as a whole. That means it should be able to detect undeclared material and activities and illegal transfers (for example the development of a clandestine uranium enrichment capacity with transfers of know-how and technologies through an undercover network and sensitive equipment fabrication delocalisation in a third country. To answer this challenge, IAEA should rely on advanced information processing and detection technologies as remote monitoring and sensing (satellite imagery), environmental sampling for forensic investigation, modern inspection equipment. Agency should also be able to collect, process, fusion and analyse information from diverse sources: verification activities, states declaration, open sources and "third party" information (as intelligence). As the Agency is not a research and development institution and has no mean to collect intelligence, the support of member state to provide adequate know-how, equipment and training is a requirement to allow the organisation like IAEA to discharge their responsibilities. In return, to take advantage of these improved capacities efficiently and without the risk of sensitive information dissemination, international bodies have to improve their management, operation mode and procedures.

*6.3. **Enforcing compliance / a renewed role of the UN Security Council***

All the efforts of the international community to patch loopholes and strengthen tools to stop proliferation and prevent nuclear terrorism would be domed to fail if states do not give themselves appropriate collective means. Atop these means is the power and involvement of the United Nation Security Council (UNSCR) to resolve crises. The role, composition and mandate of the Security Council should be revisited: extended and reinforced and backed by most leading States among them the P5 / G8 + India and some others. The UNSCR should also possibly rely on "body" of experts and inspectors to be able to intervene on all WMD issues. However, the way the Security Council is currently answer to Iran' breaches of undertakings does not seem to go in that direction.

7. An attempt to look forward

For the time being, considering the current situation, the prospects of stemming proliferation are not encouraging. Adherence to international instrument to strengthen nuclear non proliferation regime and stop development of nuclear weapons are on non proliferation are proceeding at slow pace. Given the urgent challenge the non proliferation regime has to face, the outcome from the 2005 NPT Review Conference are particularly disheartening. Thought the challenges are now clearly identified, States parties to the NPT were unable to agree on how to strengthen the implementation of the treaty. Proliferation issues are not on the way to be settle except for Libya and Iraq, most of the files do not get to close soon. North Korea has withdrawn from the NPT and is still a threat. By now nobody can have a clear view on DPRK capacity: how large is their weapon-grade plutonium stockpile? How many nuclear weapons they do have now? Are they able to produce high enriched uranium? How much are they able to produce? Meanwhile, they continue to develop ballistic missiles able to carry WMD payload possibly nuclear at long distance and have still the capacity to trade missile technology against nuclear technology. Iran is challenging the international communities and the IAEA. It has resume its conversion and enrichment activities and is rejecting all technological and commercial offers any other countries would have accepted, giving the feeling that it is obviously trying to save time to be able to advance as far as possible its nuclear weapon programme.

8. Conclusion

In the beginning of the 21st Century, the international community has to tackle new threats which challenge the nuclear regime and call for appropriate responses. These threats are various and linked together: Threats of clandestine nuclear weapon programme development under the umbrella of the NPT membership as for Iraq, Iran, Libya and North Korea; threats of the dissemination of sensitive technologies and nuclear material from States non NPT parties as for Pakistan and the A.Q.KHAN nuclear black market; threats of hyper terrorism from sub state groups acquiring radioactive or fissile material, threats of the development and acquisition by the same entities of WMD means of delivery. Loopholes in the non proliferation regime and limits of instruments have been identified. Thinking drawing on from past crisis experience have been carried out by responsible States and organisations in charge. New proposals and multilateral initiatives and agreements have been table to strengthen the different aspects of the regime with a focus: on export controls and the prevention of the diversion of sensitive technologies of the nuclear fuel cycle, and on the security of fissile and radioactive materials. National legal disposition should outlaw proliferation related activities and trade as call for by UNSCR 1540. Implementation of these initiatives and operation of these new tools should rely on the further development and use of advanced technologies at national or international level (IAEA, CTBTO,…). But all the efforts of the international community to strengthen tools to stop proliferation and prevent nuclear terrorism are bound to fail if states does not give itself appropriate means by ***promoting universal adherence*** to a strengthened non proliferation regime, ***enforcing compliance to undertakings*** through ***appropriate verification systems based on the best technological means*** and successful inspections under a strong mandate and ***bringing a strong political and technical support:*** to the organisations in charge and ***effectively implementing the international law through the United Nation Security Council (UNSCR)*** the role of which should be revisited: extended and reinforced, backed by a possible "body" of expertise and inspectorate corps.

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[3] The Proliferation Security Initiative (PSI) is a global effort that aims to stop shipments of weapons of mass destruction (WMD), their delivery systems, and related materials worldwide.

[4] The purpose of the SUA Convention is to ensure that appropriate action is taken against persons committing unlawful acts against ships. The 2005 Conference considered the adoption of two Protocols incorporating substantial amendments aimed at strengthening the SUA treaties in order to provide an appropriate response to the increasing risks posed to maritime navigation by international terrorism.

[5] United States of America President Eisenhower (1953): "*Atom for Peace*" speech to the United Nations General Assembly on the 8th December 1953.

[6] The International Atomic Energy Agency (AIEA) has been created in 1957.

[7] The AIEA The Agency Safeguards System first type of agreement covered only reactors rated less than 100 Mw_{th} then extended later to reactors of any sizes (IAEA/INFCIRC/26, 1961).

[8] The Agency's safeguards system, as approved by the Board of Governors in 1965, and extended in 1966 and 1968, covers reactors of all size, reprocessing plants and fuel fabrication plants. (IAEA/INFIRC/66/rev.2, 1968).

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[10] Jill Cooley (1998): "The Programme to strengthen the Effectiveness and Improve the Efficiency of Safeguards", International Seminar on the 1998 preparatory Committee for the 2000 NPT Review Conference, ANNECY, France.

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[18] Multilateral Approaches to the Nuclear Fuel Cycle Expert Group Report IAEA/INFCIRC/640 (Date: 22 February 2005).

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Towards wider adherence to the strengthened safeguards system: Additional protocols and small quantities protocols

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Abstract. During the past few years, the IAEA Secretariat has been implementing a Plan of Action to encourage and facilitate the conclusion of safeguards agreements and additional protocols. Such outreach activities by the IAEA and its Member States have contributed to the IAEA's effectiveness and efficiency objectives through a rapidly increasing number of States with additional protocols. Since 2001, the number of States with additional protocols in force has more than tripled from 24 to 78, while 110 States have signed additional protocols. Of particular importance for the strengthening of safeguards, more than 80% of all States with safeguards significant nuclear activities have concluded additional protocols.

Following the IAEA Board of Governors' decisions in 2005 regarding small quantities protocols (SQPs), the Secretariat's outreach activities have also promoted the acceptance of the new standardized SQP text, which allows for initial reports, early information about new facilities and IAEA inspections, while making SQPs unavailable for States with planned or existing nuclear facilities. Since the Board took its decisions, nine States have accepted the modified SQP text and the IAEA has written to all relevant States proposing to modify the existing SQPs.

For States that have not yet decided to conclude an additional protocol, the Secretariat has found that this is generally because of policy reasons or administrative hurdles or because of legal or technical obstacles. Specifically, a number of States need to put in place an effective national system for accounting and control of nuclear material and activities. Many States have taken action to strengthen such controls in order to be able to report under their safeguards agreements and additional protocols. This is a positive development from the point of view of strengthened safeguards, and also strengthens the authorities' hand in preventing and detecting attempts at illicit trafficking.

Accordingly, the IAEA has refocused its outreach activities to ensure that the State representatives responsible for reporting understand the reporting obligations under their safeguards agreements and additional protocols. A crucial challenge is to give an accurate account to State representatives of the level of efforts involved in implementing the strengthened safeguards system. On the one hand, it is critical not to overload State representatives with information, thus giving the impression that the implementation of strengthened safeguards system brings undue efforts to States with limited nuclear programmes. On the other hand, it is equally important to provide enough information so that States can provide accurate declarations on time, once the policy decision has been taken to amend an SQP and/or conclude an additional protocol.

1. Instruments of the strengthened safeguards system

1.1. Additional protocols

It is generally agreed that an effective treaty verification system should serve as a confidence-building measure and encourage full implementation. According to the 16 principles for treaty verification proposed by the UN Disarmament Commission and adopted by the United Nations General Assembly, verification should "promote the implementation of arms limitation and disarmament measures, build confidence among States and ensure that agreements are being observed by all parties”[1]. The traditional IAEA system for verification under the Treaty on the Non-Proliferation of Nuclear Weapons (NPT), with its roots in the early 1970s, was mainly focused on verifying that nuclear material declared by States was not being diverted. However, it became evident in the aftermath of the 1991 Gulf War that the IAEA safeguards system needed to be better equipped to ensure that States with comprehensive safeguards agreements (CSAs) honour their undertakings to accept safeguards on all nuclear material in all peaceful nuclear activities[2]. That is, the IAEA should be able to draw soundly-based safeguards conclusions regarding the non-diversion of declared nuclear material and, if an additional protocol is in force or otherwise applied, regarding the absence of undeclared nuclear material and activities for the State as a whole.

The refocusing of the safeguards system to the State level and the strengthening measures introduced in the early 1990s were important steps to responding to these requirements. However, it soon became clear that for enhanced effectiveness and efficiency the IAEA required expanded information on States’ nuclear fuel cycle-related activities, better access to nuclear sites and other related locations, as well as the optimized use of technology (in particular environmental sampling) and less administrative constraints. All of those aspects were addressed in the Model Protocol Additional to the Agreement(s) between State(s) and the International Atomic Energy Agency for the Application of Safeguards[3], which was approved by the Board of Governors in 1997 with a view to strengthening IAEA safeguards.

The conclusion of additional protocols to safeguards agreements (based on the Model Additional Protocol) with all States, is central to the IAEA’s ability to meet the expectations for the safeguards system and to transform the system into a robust and cost-effective treaty verification system. Additional protocols are, in the words of the IAEA Director General, sine qua non for effective verification[4].

Conclusion 1. The entry into force of additional protocols with all States with safeguards agreements is key to an effective and efficient IAEA safeguards system, and in particular for ensuring that safeguards are applied on all nuclear material in all peaceful nuclear activities in States with comprehensive safeguards agreements.

1.2. Small Quantities Protocols (SQPs)

Since 1972, the IAEA has been offering a State that has no nuclear material or other items subject to regular inspections under IAEA safeguards agreements a protocol to its CSA which, as long as the State fulfils certain criteria in terms of nuclear material inventories[5], holds in abeyance the implementation of most safeguards measures. Beginning with the 2003 Safeguards Implementation Report (SIR), the IAEA Secretariat drew Member States’ attention to the fact that SQPs limited the Secretariat’s authority to apply key measures for drawing soundly-based safeguards conclusions for the States concerned[6]. To address this problem, the Secretariat proposed to the Board to discontinue

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[1] A/51/182 G (1996).

[2] See paragraph 2 of INFCIRC/153 (Corrected), The Structure and Content of Agreements between the Agency and States Required in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons, IAEA, Vienna (1972).

[3] The text of the Model Additional Protocol is reproduced in INFCIRC/540 (Corrected), IAEA, Vienna (1997).

[4] Introductory Statement to the Board of Governors by the IAEA Director General, 8 March 2004; the text is available online at: http://www.iaea.org/NewsCenter/Statements/2004/ebsp2004n002.html.

[5] The quantity of nuclear material subject to safeguards should be below the threshold established in paragraph 37 of INFCIRC/153 (Corrected) and there should be no nuclear material in a facility.

[6] “For a State in which an SQP is implemented but which does not have an additional protocol in force, the IAEA has only very limited means to evaluate any potential nuclear activities which might need to be declared, or to confirm that the State meets or continues to meet the conditions required for having an operative SQP”, Safeguards Implementation Report for 2003, IAEA, Vienna (2004).

the use of SQPs or at least modify the standardized text of the SQPs to allow for inspections, some reporting on nuclear material inventories and notification to the IAEA of decisions to construct a nuclear facility.

In September 2005, the Board decided that:

SQPs should remain part of the IAEA safeguards system, subject to modifications in the standardized text[7] and change in the criteria[8] for an SQP;
henceforth, it would approve only the standardized text for SQPs based on the revised standardized text and subject to the modified criteria; and
the Director General would be authorized to conclude with States with existing SQPs exchanges of letters giving effect to the revised standardized text and the modified criteria.

At the same time, the Board called upon States concerned to conclude such exchanges of letters as soon as possible. It also requested the Secretariat to assist States with SQPs, including non-member States, through available resources, in the establishment and maintenance of their State systems of accounting for and control of nuclear material (SSACs)[9].

To implement the Board's decisions, the IAEA has written to 87 States with operational SQPs, proposing to amend these to reflect the revised standardized text. After a little more than a year, seven States have amended their SQP and two States have either signed or submitted for Board approval an SQP based on the revised standardized text (see Annex 2).

Conclusion 2. The Board of Governors decision in 2005 to modify the standardized text of an SQP constitutes an important safeguards strengthening measure that will improve the technical basis for drawing soundly-based safeguards conclusions for States with SQPs.

2. Intensified Secretariat efforts

Upon approval of the Model Additional Protocol in 1997, the Board tasked the Director General with negotiating additional protocols with all States that have safeguards agreements[10]. In 2000, the IAEA General Conference defined the following elements of a plan of action to promote the entry into force of safeguards agreements and additional protocols[11]:

Intensified efforts by the Director General to conclude safeguards agreements and additional protocols, especially with those States having substantial nuclear activities under their jurisdiction;
Increased bilateral and regional consultations among Member States at both technical and political levels, with a view to promoting the domestic process to conclude safeguards agreements and additional protocols;
Assistance by the IAEA Secretariat and Member States to other States by providing the knowledge and technical expertise necessary to conclude and implement safeguards agreements and additional protocols;

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[7] The modified standardized text of an SQP endorsed by the Board of Governors has the effect of requiring States to provide initial reports on nuclear material, to report to the IAEA any decision to construct or authorize construction of a nuclear facility and to allow for IAEA inspections (GOV/INF/276/Rev.1 and Corr.1, IAEA, Vienna (2005)).

[8] While the 'old' SQP was available to a State with a nuclear facility as long as it did not contain nuclear material, the Board of Governors decided that a State with a planned or existing facility should no longer be eligible for an SQP.

[9] Pursuant to paragraph 8 of INFCIRC/153 (Corrected) a State with a comprehensive safeguards agreement pursuant to the NPT (including a State with an SQP) is required to establish and maintain an SSAC.

[10] This includes States with an INFCIRC/153-type safeguards agreement, a voluntary offer agreement or with an INFCIRC/66-type agreement.

[11] IAEA General Conference Resolution, as documented in GC(44)/RES/19, IAEA, Vienna. (2000).

Reinforced coordination between Member States and the IAEA Secretariat in their efforts to promote the conclusion of safeguards agreements and additional protocols; and
Consideration by Member States, subject to progress made under the above, of further steps to promote safeguards agreements and additional protocols, including an appropriate international meeting.

To focus the IAEA's efforts for implementing this mandate, the Secretariat developed the "Plan of Action to Promote the Conclusion of Safeguards Agreements and Additional Protocols[12], which was endorsed by the General Conference[13]. It serves to ensure a coordinated effort by the Secretariat and interested States in support of a wider adherence to the strengthened safeguards system.

Beginning with a seminar in Lima in 2001, the Secretariat has launched an intensified effort to assist States in better understanding the policy, legal and technical aspects of the IAEA's strengthened safeguards system. Further efforts have comprised, inter alia, regional, interregional and subregional seminars for Government officials, the publication of topical booklets and contributions to enable developing country representatives to travel to receive advice and training. These activities have been funded chiefly through extrabudgetary contributions.

To realize the maximum contribution to the effectiveness and efficiency of the safeguards system, the IAEA has given priority in its outreach activities to the conclusion of additional protocols with all States with significant nuclear activities[14]. The Plan of Action makes the distinction between States with significant nuclear activities and without such nuclear activities, devising different outreach approaches for these categories of States. A further distinction as far as outreach activities are concerned is made between Member States and non-Member States of the IAEA, the latter category of which lack regular interaction with the IAEA and lag behind with regard to the conclusion of both safeguards agreements pursuant to the NPT and additional protocols.[15]

Figure 1 shows a slowing trend in the rate of conclusion of additional protocols around 2001, with only 60 States having submitted additional protocols for Board approval. The reversal of that trend coincides with the launch of the Plan of Action that year. Since then, there has been a steady growth in the number of States concluding additional protocols. As of 10 October 2006, 115 States had submitted additional protocols for Board approval; there were 110 additional protocol signatories and 78 States with additional protocols in force, including 75 States with CSAs[16].

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[12] The text of the Plan of Action is available on line at:
http://www.iaea.org/OurWork/SV/Safeguards/sg_actionplan2006.pdf.

[13] IAEA General Conference Resolution, as documented in GC(50)/RES/14, operative paragraph 22, IAEA, Vienna (2006).

[14] In this paper, a State with significant nuclear activities is defined as a State in which the IAEA performs routine inspections.

[15] Whether a State is a Member State of the IAEA or not has no bearing on its NPT obligation to conclude a safeguards agreement with the IAEA or on the IAEA's obligation, in accordance with such agreements, to verify that the State is complying with its commitments.

[16] In GC/(50)/RES/14, operative paragraph 15, the General Conference "noted that in the case of a State with a CSA supplemented by an AP in force, these measures represent that enhanced verification standard for that State." , IAEA, Vienna (2006).

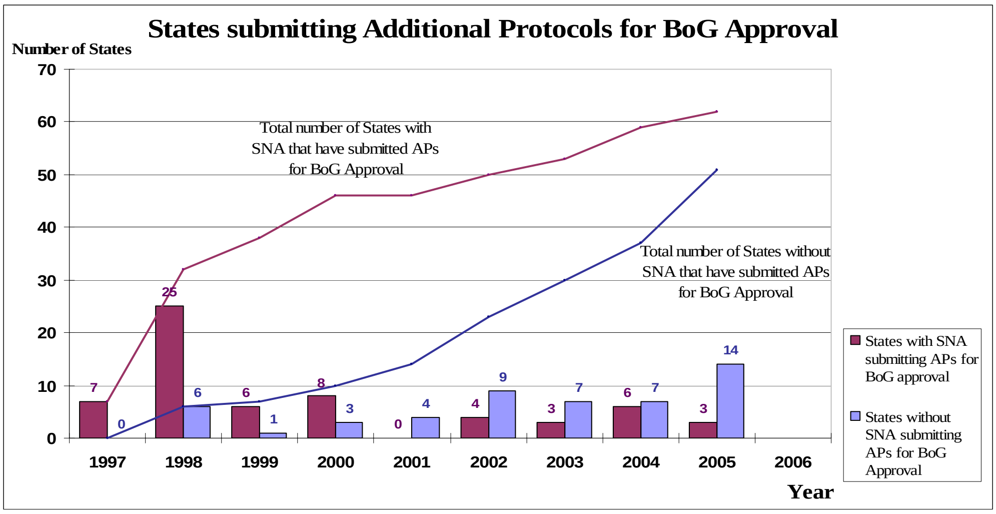

FIG. 1. States submitting additional protocols for BoG approval.

Conclusion 3. The IAEA's effort to facilitate wider adherence to the strengthened safeguards system through the implementation of the Plan of Action has had a significant impact on the pace of conclusion of additional protocols.

Figure 1 also compares the rate of conclusion of additional protocols in States with and without significant nuclear activities. It shows that, by the year 2000, most States with such nuclear activities had already submitted additional protocols for Board approval. However, for each year since 2001, the majority of States submitting additional protocols for Board approval has been in the category of States without significant nuclear activities.

Not all States with significant nuclear activities have been in a position to endorse the Model Additional Protocol at an early stage. Several States with significant nuclear activities have sought and received IAEA assistance and advice in their national decision processes. In a number of instances, the IAEA has hosted national negotiation teams and provided speakers at national inter-agency seminars involving relevant Ministries and Government authorities. Since 2001, such national additional protocol events have been organized by Algeria, Colombia, Haiti, Malaysia, Mexico, Switzerland, Thailand and Vietnam, while Belarus, Morocco, Saudi Arabia, Tunisia and Ukraine have dispatched negotiation teams to IAEA headquarters. Eleven States with significant nuclear activities — including three States with INFCIRC/66-type safeguards agreements — have so far not submitted additional protocols for Board approval.

Conclusion 4. The IAEA's outreach to Governments has facilitated the conclusion of additional protocols by many States without significant nuclear activities. With regard to States with significant nuclear activities, the IAEA has often assisted the relevant decision-making processes through negotiations, technical training and contributions to national seminars.

3. Lessons learned

3.1. Outreach to States with SQPs

With regard to its outreach activities, the Secretariat has found it particular challenging to give an accurate account of the efforts involved in implementing strengthened safeguards. While this might seem to be rather straightforward, in reality it is a difficult task. Government officials hesitate to recommend to their authorities to make commitments that are perceived as potentially burdensome. Moreover, providing too much detailed information on reporting requirements risks losing the message that additional protocol implementation is straightforward for States without significant nuclear activities.

This situation is especially true for States with very small public administrations, such as least developed countries (LDCs) and small island developing countries (SIDS). Such States constitute more than half of the 87 countries with operative SQPs and almost all of the 30 States party to the NPT that have not yet concluded safeguards agreements pursuant to the NPT (see Annex 1).

Another lesson learned from these outreach activities is the need to keep reporting requirements simple. In recognition of this, the IAEA has introduced a new, simplified form for the Initial Report on Nuclear Material which is required under the modified standardized text of the model SQP[17]. Work is underway to develop a simplified format for the required additional protocol declarations for States with SQPs. Moreover, in 2006 the Secretariat published a user-friendly booklet providing an overview of safeguards reporting requirements for States with limited nuclear material and activities[18].

Conclusion 5. For wider adherence to the strengthened safeguards system among States with SQPs, it is important to simplify safeguards reporting requirements and to provide an accurate impression of the real efforts involved.

3.2. Government outreach and SSAC training for Non-Member States

Technical training of State officials to develop an effective SSAC is closely related to the aim of concluding safeguards agreements and additional protocols. Without well-functioning SSACs, States are unable to fulfil their safeguards obligations, and without understanding the technical aspects of the safeguards system many States hesitate to conclude the related instruments. However, when reaching out to Government officials, it is important to arrive at the right balance between policy content and detailed technical information.

In 2005, the Board requested that the Secretariat assist States with SQPs, including non-Member States of the IAEA, to establish and maintain SSACs. Since non-Member States are normally not involved in IAEA technical cooperation and other training, provided that funding is available for a safeguards event for non-Member States, the Secretariat has so far been combining outreach efforts with detailed legislative and technical training. However, the Secretariat has found that the most effective outreach is achieved when focus is initially placed on policy and SSAC training is provided further downstream, once a State has made the policy decision to conclude an additional protocol.

Conclusion 6. With regard to non-Member States of the IAEA, it is important not to mix messages, but to focus outreach assistance initially at the policy level and subsequently on more technical issues.

3.3. Factors influencing the conclusion of additional protocols

In the course of the implementation of the Plan of Action, the Secretariat has been consulting with more than 150 States. This experience indicates that there are various factors that may prevent a State from concluding an additional protocol, and it suggests how the IAEA might be able to assist the State in this regard.

While it is not for the Secretariat to second-guess States' rationale for taking sovereign decisions, the Secretariat has found that these impediments tend to fall into one or several of four categories: technical factors, legislative factors, policy-related factors and administrative factors. Table 1 gives an overview of these factors influencing the conclusion of additional protocols. The IAEA has been able to help to address some of these factors, through its outreach activities, training and legislative assistance, as summarized below.

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[17] Paragraph 62 of INFCIRC/153 (Corrected).

[18] Non-Proliferation of Nuclear Weapons and Nuclear Security: Overview of Safeguards Requirements for States with Limited Nuclear Material and Activities, Jan Lodding and Bernardo Ribeiro, IAEA, Vienna (2006).

Table 1. Factors influencing the conclusion of additional protocols.

Type of factors	Example	IAEA assistance
1.Technical factors	Lack of a designated or effective SSAC.	SSAC training
2.Legislative factors	Lack of appropriate legislation and/or regulations required for the implementation of additional protocols.	IAEA legislative assistance
3. Policy-related factors	Low priority attached to nuclear disarmament and non-proliferation in relation to other national concerns or expectation of security or economic benefits in return.	Government outreach, technical and legislative assistance
4.Administrative factors	Limited administrative links of the authorities dealing on a day-to-day basis with IAEA affairs to those dealing with the conclusion of international instruments, such as Ministries of Foreign Affairs/Legal Affairs..	Outreach seminars

3.3.1. Outreach activities

The IAEA outreach to Governments aims to provide information on the policy, legal and technical aspects of the strengthened safeguards system. The Secretariat encourages States nominating participants for outreach seminars to pair up officers dealing with non-proliferation issues or the conclusion of international legal instruments, with experts on radiation protection and regulatory control, in order to facilitate interaction between those officials that would typically be involved in concluding and implementing safeguards agreements and additional protocols.

Generally at such seminars, the IAEA speakers describe the policy role of IAEA verification, the background of the system, the provisions of INFCIRC/153 (Corrected) and INFCIRC/540 (Corrected) and the verification activities carried out in the field and at IAEA headquarters. Invited speakers talk about nuclear disarmament and non-proliferation policies and about their national experience in concluding and implementing CSAs and additional protocols. Among the most useful elements of such outreach seminars have been bilateral consultations between IAEA teams (policy, legal and technical experts) and representatives of the States, at which each State’s particular situation is addressed. A number of States have provided extrabudgetary funding for the Secretariat’s outreach activities, including Australia, France, Japan, Sweden and the United States of America.

3.3.2. IAEA training for SSACs

The SSAC training is an off-shoot of the IAEA’s internal inspector training programme. Over the years, in response to rising demand for training among Member States, the Safeguards Training Section in the Division of Technical Support has expanded the services offered to train representatives of SSACs. For a number of years, the Russian Federation and the USA have taken turns hosting a yearly interregional training course on nuclear material accountancy and control. In recent years, additional regional training on national implementation of the strengthened safeguards system have been hosted by, among others, Australia, Brazil and Japan, and national training courses have been arranged in response to requests from States in Central and Eastern Europe and elsewhere. In 2005, the IAEA introduced its International SSAC Advisory Service (ISSAS), whereby the IAEA, at the request of a State, dispatches a team to review the SSAC and provide recommendations on how it might be improved. ISSASs have so far been provided to Indonesia and the Republic of Korea and are scheduled, in the coming months, to Serbia and Singapore.

3.3.3. IAEA legislative assistance

For many years, the IAEA has offered assistance to States seeking guidance on issues of nuclear legislation. This type of legislative assistance related to nuclear and radiation safety, security and safeguards has been provided in connection with the IAEA's specialized services, such as the Operational Safety Assessment Services (OSART), the Physical Protection Assessment Services (IPPAS) and the Safety and Security of Radioactive Sources Infrastructure Appraisal Services (RASSIA). The IAEA has increasingly emphasized the importance of a comprehensive approach to nuclear legislation, taking into consideration the legal framework incorporating safety, security and safeguards. Regional Technical Cooperation projects on legislative assistance for the safe and peaceful use of nuclear energy have been carried out in most regions, and continue to be implemented in Africa and East Asia[19]. In terms of safeguards legislation, a European Union funded project was introduced in 2004 to help States introduce legislation for the implementation of additional protocols. The IAEA has prepared modules covering aspects of nuclear legislation and published a two-volume Handbook of Nuclear Law to help States in this regard[20].

Conclusion 7. With regard to those States that have not yet decided to conclude an additional protocol to the safeguards agreement, the Secretariat has found that this is generally due to technical, legal, policy or administrative factors.

4. Integrated safeguards

The 2001-2005 IAEA Medium Term Strategy set the objective of introducing integrated safeguards through the conclusion of additional protocols with most States and with almost all States with significant nuclear activities. This objective has been partially met as; by the end of 2005, 58% of all States (and 82% of States with significant nuclear activities) had signed additional protocols. However, only 47 (65%) of the 73 States with significant nuclear activities had actually brought their additional protocol into force, and State-level integrated safeguards approaches had only been approved for 11 of those States (including Canada and Japan that have large, complex nuclear fuel cycles). As shown in Figure 2, with regard to States with significant nuclear activities the gap between those with additional protocols signed and in force began to narrow only over the last few years. As recently as the end of 2003, only 23 States with significant nuclear activities had additional protocols in force, a number which had doubled by the end of 2005.

The link between the slow ratification procedures in States with significant nuclear activities and the putting in place of integrated safeguards is the fact that as a precondition for implementation the IAEA must be able to draw the broader safeguards conclusion that all nuclear material in a State remained in peaceful activities. This process may take from two and six years for a State with significant nuclear activities. Also, for the IAEA to be able to draw this broader conclusion for a State, an additional protocol must be in force or otherwise applied.

[19] RAF/0/015 and RAS/9/023, respectively.

[20] Handbook of Nuclear Law, IAEA, Vienna (2003); available in Arabic, English, French, Russian and Spanish. The English version is available on line at http://www-pub.iaea.org/MTCD/publications/PDF/Pub1160_web.pdf.

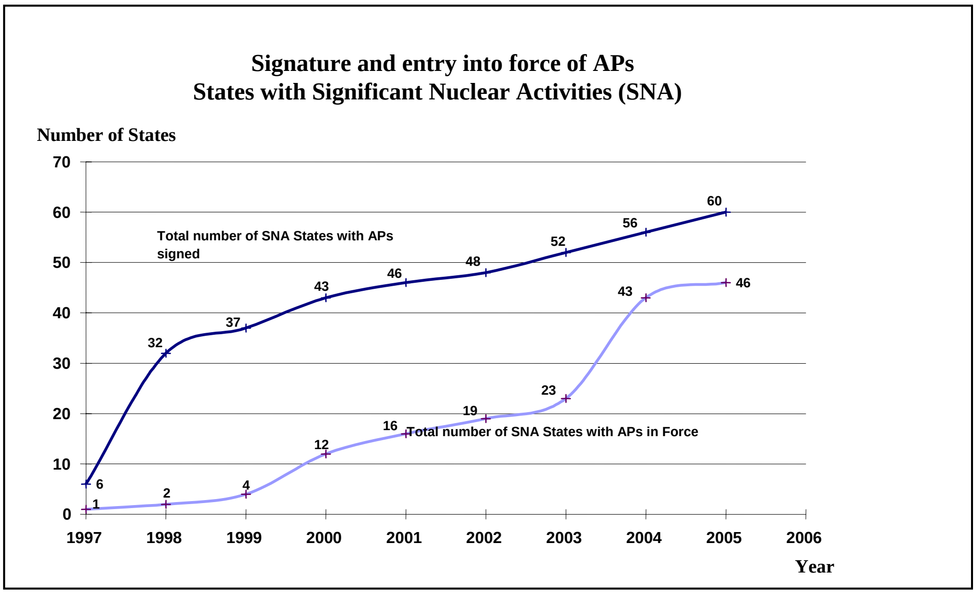

FIG. 2. Signature and entry into force of additional protocols – SNA.

Conclusion 8. The slow rate of completion of additional protocol ratification procedures in several States with significant nuclear activities has caused delays in the putting in place of integrated safeguards.

5. Combating covert supply networks

The recent discovery of clandestine nuclear supply networks related to the sensitive aspects of the nuclear fuel cycle, involving entities in more than 30 countries, is one of the most worrisome developments in the field of non-proliferation of nuclear weapons. It has led to added focus by the international community on the need to improve national controls of weapons of mass destruction (WMD) material and technology and to prevent the proliferation of such items to or through non-State actors.

Various initiatives have been taken to try to address the problem, including United Nations Security Council resolutions 1540 (2004) and 1673 (2006) and the Proliferation Security Initiative (PSI). However, notwithstanding seemingly good intentions, such initiatives – even those developed within the framework of the UN Security Council – have been perceived by some as being imposed by a group of like-minded States rather than being truly multilateral instruments.

There has been a growing realization of the importance of the strengthened safeguards system in this regard. CSAs require the establishment of SSACs to monitor nuclear material flows. By concluding additional protocols, States also willingly commit to the monitoring of exports of specified equipment and non-nuclear material and to reporting to the IAEA their exports and (on request) imports. Thus, the strengthened safeguards system is the only multilaterally negotiated international instrument that provides for the strengthening of national controls of transfers of nuclear material and related technology. Such controls are central to the ability effectively to interdict illicit trafficking in such material and technology.

This link is recognized in the IAEA’s Nuclear Security Plan of Activities, which includes the training of SSACs among its activities. It is also reflected by the fact that several States, in their reports to the 1540 Committee (established pursuant to the UN Security Council resolution 1540), have identified the conclusion of CSAs and additional protocols and the introduction of related legislation as elements of their national implementation of that resolution.

J. Lodding

Conclusion 9. Safeguards agreements and additional protocols help to strengthen nuclear security both nationally and globally, by requiring States to establish SSACs and to monitor exports and imports of nuclear material and related technology.

ANNEX 1

Overview of the Status of Conclusion of Additional Protocols and Outstanding NPT Safeguards Agreements

Current situation as of 10 October 2006

Of the **115** States that have not yet brought into force additional protocols with the IAEA:

- There are **31** States with which additional protocols have been signed but are not yet in force.

Albania	Andorra	**Belarus**	Benin	Cameroon	Cape Verde	**Colombia**
Comoros	Costa Rica	Gabon	Guatemala	Honduras	**Islam. Rep. Iran**	**Kazakhstan**
Kiribati	**Malaysia**	Mauritania	Mauritius	**Mexico**	**Morocco**	Namibia
Niger	**Nigeria**	**Philippines**	**Russia**	Singapore	**Thailand**	The FYROM
Togo	Tunisia	**USA**				

- There are 6 States for which additional protocols have been approved by the Board of Governors but have not yet been signed.

Algeria	Central African Rep.	Liechtenstein	Rep. Moldova	Senegal	**Serbia**

- There are **78** States that have not yet submitted additional protocols to the Board of Governors for its consideration.

Angola	Antigua & Barbuda	**Argentina**	Bahamas	Bahrain	Barbados	Belize
Bhutan	Bolivia	Bosnia & Herzegovina	**Brazil**	Brunei Darussalam	Burundi	Cambodia
Chad	Congo Rep.	Côte d'Ivoire	**DPRK**	Djibouti	Dominica	Dominican Republic
Egypt	Equatorial Guinea	Eritrea	Ethiopia	Gambia	Grenada	Guinea
Guinea-Bissau	Guyana	**India**	**Iraq**	**Israel**	Kenya	Kyrgyzstan
Lao PDR	Lebanon	Lesotho	Liberia	Malawi	Maldives	Micronesia, Fed. States
Montenegro	Mozambique	Myanmar	Nauru	Nepal	Oman	**Pakistan**
Papua New Guinea	Qatar	Rwanda	St. Kitts and Nevis	St. Lucia	St. Vincent Grenadines	Samoa
San Marino	Sao Tome & Principe	Saudi Arabia	Sierra Leone	Solomon Islands	Somalia	Sri Lanka
Sudan	Suriname	Swaziland	**Syrian Arab Rep.**	Timor-Leste	Tonga	Trinidad & Tobago
Tuvalu	Un. Arab Emirates	Vanuatu	**Venezuela**	**Vietnam**	Yemen Rep.	Zambia
Zimbabwe						

Bold indicates States with significant nuclear activities; Underline indicates States with outstanding NPT obligations to conclude comprehensive safeguards agreements.

ANNEX 2

SQP: exchange of letters - current situation as of 10 October 2006			
	State	**Letter sent**	**Amended**
1	**Afghanistan (LDC)**	12-Dec-05	
2	*Andorra*	*14-Dec-05*	
3	*Antigua and Barbuda (SIDS)*	24-Aug-06	
4	**Azerbaijan**	12-Dec-05	
5	*Bahamas (SIDS)*	24-Aug-06	
6	*Barbados SIDS)*	24-Aug-05	
7	*Belize*	12-Dec-05	
8	Benin (LDC)	*12-Dec-05*	
9	*Bhutan (LDC)*	14-Dec-05	
10	Bolivia	1-Sep-06	
11	*Brunei Darussalam*	15-Dec-05	
12	**Burkina Faso (LDC)**	12-Dec-05	
13	*Cambodia (LDC)*	12-Dec-05	
14	Cameroon	12-Dec-05	
15	*Cape Verde (LDC, SIDS)*	*12-Dec-05*	*27-Mar-06*
16	Costa Rica	1-Sep-06	
17	**Croatia**	15-Dec-05	
18	**Cyprus**	15-Dec-05	
19	*Dominica (SIDS)*	15-Dec-05	
20	Dominican Republic (SIDS)	1-Sep-06	10-Oct-06
21	**Ecuador**	13-Dec-06	7-Apr-06
22	**El Salvador**	1-Sep-06	
23	Ethiopia (LDC)	12-Dec-05	
24	***Fiji***	10-Dec-05	
25	*Gabon*	*12-Dec-05*	
26	*Gambia (LDC)*	15-Dec-05	
27	*Grenada (SIDS)*	24-Aug-06	
28	Guatemala	1-Sep-06	
29	*Guyana*	24-Aug-06	
30	**Haiti (LDC, SIDS)**	1-Sep-06	
31	**Holy See**	5-Sep-06	11-Sep-06
32	Honduras	11-Sep-06	
33	**Iceland**	12-Dec-05	
34	**Jordan**	14-Dec-05	
35	*Kiribati (LDC, SIDS)*	14-Dec-05	
36	**Kuwait**	15-Dec-05	
37	Kyrgyzstan	15-Dec-05	
38	*Lao People's Democratic Rep. (LDC)*	12-Dec-05	
39	Lebanon	14-Dec-05	
40	*Lesotho (LDC)*	15-Dec-05	
41	**Madagascar (LDC)**	12-Dec-05	
42	*Malawi (LDC)*	12-Dec-05	
43	*Maldives (LDC, SIDS)*	12-Dec-05	
44	**Mali (LDC)**	12-Dec-05	18-Apr-06
45	**Malta**	15-Dec-05	
46	Mauritania (LDC)	*14-Dec-05*	
47	Mauritius	8-Dec-05	
48	**Monaco**	12-Dec-05	
49	**Mongolia**	14-Dec-05	
50	Myanmar (LDC)	15-Dec-05	

	SQP: exchange of letters - current situation as of 10 October 2006		
	State	**Letter sent**	**Amended**
51	Namibia	12-Dec-05	
52	*Nauru*	15-Dec-05	
53	*Nepal (LDC)*	15-Dec-05	
54	**New Zealand**	14-Dec-05	
55	**Nicaragua**	1-Sep-06	
56	*Oman*	14-Dec-05	
57	**Palau (SIDS)**	9-Dec-05	15-Mar-06
58	**Panama**	1-Sep-06	
59	*Papua New Guinea*	9-Dec-05	
60	**Paraguay**	1-Sep-06	
61	Republic of Moldova	14-Dec-05	
62	*Saint Kitts and Nevis (SIDS)*	24-Aug-06	
63	*Saint Lucia (SIDS)*	14-Dec-05	
64	*Saint Vincent and the Grenadines (SIDS)*	15-Dec-05	
65	*Samoa (LDC, SIDS)*	9-Dec-05	
66	*San Marino*	14-Dec-05	
67	<u>Saudi Arabia</u>	*12-Dec-05*	
68	Senegal (LDC)	12-Dec-05	
69	**Seychelles (SIDS)**	8-Dec-05	
70	*Sierra Leone (LDC)*	*12-Dec-05*	
71	Singapore	12-Dec-05	
72	*Solomon Islands (LDC, SIDS)*	9-Dec-05	
73	Sudan (LDC)	5-Dec-05	
74	*Suriname (SIDS)*	24-Aug-06	
75	Swaziland	14-Dec-05	
76	**Tajikistan**	15-Dec-05	6-Mar-06
77	The Former Yugoslav Rep. of Macedonia	12-Apr-06	
78	*<u>Togo (LDC)</u>*	*12-Dec-05*	
79	*Tonga (SIDS)*	12-Dec-05	
80	*Trinidad and Tobago (SIDS)*	24-Aug-06	
81	*Tuvalu (LDC, SIDS)*	8-Dec-05	
82	**Uganda**	8-Dec-05	
83	United Arab Emirates	15-Dec-05	
84	**United Republic of Tanzania (LDC)**	5-Dec-05	
85	Yemen, Republic of (LDC)	12-Dec-05	
86	Zambia (LDC)	2-Dec-05	
87	Zimbabwe	5-Dec-05	
	Total	**87**	**7**

Key
<u>States</u>: *States that have signed NPT safeguards agreements with SQPs but not brought these into force (8 States)*
States: States that have Additional Protocols in force (27 States)
States: Non-Members of the IAEA (34 States)

SQP: conclusion new SQP - current situation as of 13 September 2006				
	State	**Approved**	**Signed**	**In force**
1	Central African Republic (LDC)	*07-Mar-06*		
2	*Comoros (LDC, SIDS)*	*16-Jun-05 (old text)*	*2005-12-13 (new text)*	
	Total	**2**	**1**	**0**

A formal approach for verifying treaty compliance

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Abstract. International treaties and conventions enunciate desirable broad objectives to limit or eliminate weapons of mass destruction. Similar objectives are set forth in treaties dealing with the environment, safety of transports and waste disposals, etc. Treaties requiring verification of compliance often generate controversies centered about the effectiveness of the verification regime in detecting non-compliance and of sanctions as a mechanism to enforce compliance. The debates in the public forum do not provide enough factual information to allow for a dispassionate analysis of the issues; the cause of the treaties would be advanced by conducting the debates on an objective basis. This paper presents a formal approach for addressing the verification question as a technical issue using objective criteria. The approach is based on the observation that treaties can be modeled as processes with feedback. The use of feedback control concepts makes it possible to analyze treaty goals, verification objectives and monitoring regimes and evaluate their contribution to detecting non-compliance. Examples from the NPT, CFE, BWC, CWC, CTBT, the UNSCR 687 implementation (Iraq) and the Kyoto Protocol are used in order to illustrate the issues that need to be addressed. Treaty goals are classified into quantifiable and non-quantifiable categories. The verification objectives articulated in each treaty are examined for their relevance to the stated treaty goals and the notion of quantitative and qualitative measures of discrepancy are introduced. Similarly, the monitoring regimes for each treaty are examined with respect to verification objectives and treaty goals. The paper demonstrates that uncertainties in the specification of treaty goals, limitations in the verification regimes and deficiencies in the monitoring systems decrease the probability of detecting non-compliance. It also includes suggestions on how to increase the probability of detection.

1. Introduction

International treaties and conventions requiring binding commitments on the part of the member states and establishing appropriate compliance verification regimes constitute a primary assurance against such risks. For instance, the Treaty on the Non-proliferation of Nuclear Weapons (NPT) with its safeguards system represents the cornerstone in the nuclear field, aims to give assurance about the peaceful nature of the nuclear activities and to prevent proliferation of nuclear weapons. The NPT, with its long history and positive experience since 1968, has become a model, and its elements have influenced other fields where proliferation concerns exist, such as the biological and chemical industries. In the environmental field, the Kyoto Protocol represents the major international effort to reduce the emissions of greenhouse gases.

In spite of the considerable achievements of the NPT and other arms control treaties, one can raise some serious questions about their effectiveness both in terms of achieving their stated goals and verifying compliance. The Kyoto Protocol became controversial even since its inception. With respect to the NPT, the number of States having nuclear weapons programs, proven and suspected, or possessing nuclear weapons has increased. Also, the original safeguards system has been demonstrated to be inadequate to detect clandestine nuclear weapons activities. Consequently, improvements have been sought through the Additional Protocol. For other treaties such as the Biological Weapons Convention (BWC) the feasibility of an effective verification regime has been questioned. The text of a Fissile Material Cut-off Treaty (FMCT) tabled recently by the US in the Committee on Disarmament does not even include provisions to provide for a verification mechanism on the grounds that the treaty is not effectively verifiable. It rests on the parties to verify compliance. This view, however, is not shared by other States.

Contentious as the subject of verification may be, its significance is probably secondary to the broader question of the effectiveness of a given treaty. Has the NPT been successful in limiting the proliferation of nuclear weapons? Or has the Chemical Weapons Convention (CWC) contributed to the elimination of chemical weapons? How good have the various arms control treaties been in enforcing compliance? For these and other arms control and disarmament treaties the answers to such questions are frequently based on a mixture of facts, alleged facts, broad assertions, preconceived notions and ulterior motives. In this paper we argue that it is possible to develop such a mechanism, although it should be realized that, treaties, although formally are legal instruments, in effect, their operation is to a large extent influenced by political considerations. An equally important consideration is the cost of implementing verification regimes or enforcing compliance. Determining trade-offs between the effectiveness of a treaty and the associated costs may not be amenable to strictly formal analysis. Even with these caveats in mind, an objective mechanism would allow for a more dispassionate analysis of the operation of existing treaties and provide guidelines for negotiating more effective future treaties. The question then becomes whether objective measures and criteria could be devised for addressing issues such as compliance and effectiveness. It is of timely relevance for the enhanced NPT safeguards under the additional Protocol.

We present a formal approach for addressing issues of verification and compliance. It is based on the observation that the operation of a treaty is a *process* that can be modeled, analyzed and evaluated using feedback systems concepts. The approach relies on the integration of methodologies and technologies for introducing some degree of objectivity to these issues. To analyze and evaluate performance of treaties, existing or under negotiation, some of the questions to be answered are:

- What is a given treaty expected to achieve?
- What is the monitoring mechanism?
- How is non-compliance detected?
- How is compliance enforced?
- How well does the verification system work?
- How effectively does the treaty operate or will operate?
- How is performance measured?
- What changes in the verification regime could improve the performance of a treaty?
- How is the feasibility of a given performance objective evaluated?

2. Treaty models

Treaties that list obligations of the States Parties with respect to specified activities, require monitoring of the activities and verification of compliance with the obligations, and provide for sanctions in case of non-compliance can be modeled as feedback control systems [1]. The specified activities define the treaty *process* with the obligations being the treaty goals or *reference inputs* in control systems terminology; these are typically listed in the preamble of a treaty in the form of motivating factors and desirable objectives, and in the articles describing the obligations of the States. Discrepancies, *error*, between the obligations and the on-going activities constitute non-compliance. The role of the verification regime is to detect discrepancies between the specified activities and the treaty goals. Some treaties require explicitly that compliance be enforced; in others the requirement is implicit. The enforcement mechanism, in effect the *controller*, is a collection of tools including consultations, clarifications complementary access, special or on-site inspection and, ultimately, sanctions if evidence of non-compliance has been found. This top level model is shown in Figure 1.

Although the reality of the operations of any treaty is a rather complicated affair, the decomposition of the treaty into five major components connected in a feedback arrangement allows for a systematic analysis of the issues associated with each of these components and the interrelationships among them. Starting with the central question of non-compliance, one can immediately conclude that the answer is not a simple yes or no. Discrepancies between treaty obligations and the information produced by the monitoring system may indicate either non-compliance with the obligations, or imperfect operation of the verification regime. The latter can have different causes. Regardless how well-designed the monitoring system is, if some treaty goals and obligations are ambiguous or not measurable, there is no way one could determine whether these discrepancies should be attributed to non-compliance or system errors. Even in cases where the goals are measurable and the monitoring system is sufficiently well-designed to generate the necessary measurements, discrepancies are bound to arise due to the measurement uncertainties. The safeguards system for declared nuclear facilities is a good example of the latter.

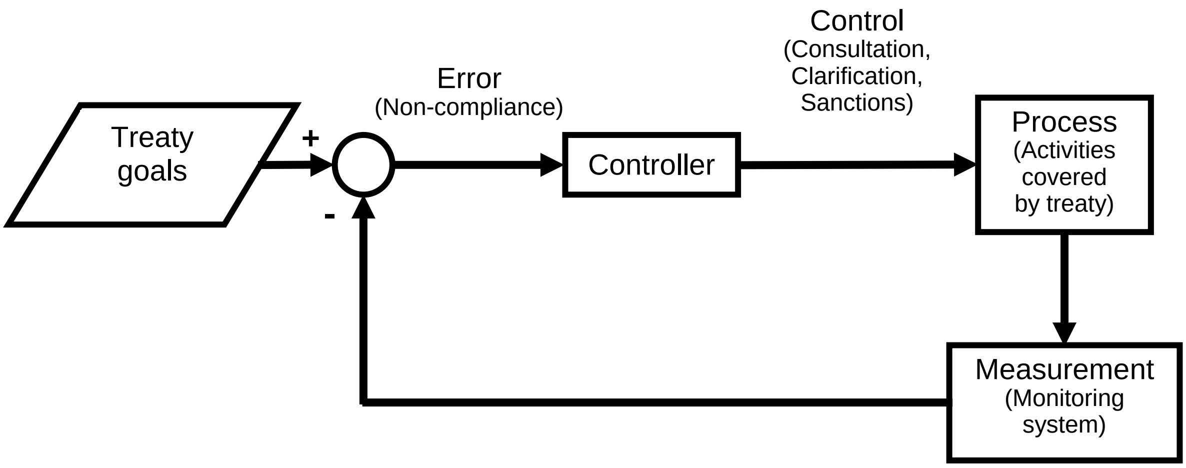

Figure 1. A generalized model of multilateral treaties as a feedback control system.

Through similar reasoning one can easily conclude that a verification regime can generate discrepancies that may not necessarily be attributable to non-compliance. Such discrepancies may be labeled benign and caused by insufficient specification of the processes covered by the treaty, incomplete design of the monitoring system and improper analysis and evaluation of the discrepancies between goals and measurements, in addition to the possible ambiguities in the specification of the treaty goals.

Viewing treaties as feedback control systems allows also for a more systematic analysis of measures adapted to redress the situation and ultimately of sanctions as the corrective mechanism. Assuming that discrepancies detected by the verification regime could likely be attributed to non-compliance, the question becomes what would be the appropriate corrective mechanism for enforcing compliance. These mechanisms, or control functions, need to be designed as to minimize the discrepancies to some acceptable level within specified periods of time. On the basis of the experiences with existing treaties, it is not clear whether such mechanisms can be devised.

In the following sections we will address specific issues associated with each of the components of the system shown in Figure 1.

2.1. Obligations of State Parties

One would expect that one of the least problematic components of a treaty regime would be the goals of the treaty. Yet, they give rise to a number of problems both in terms of their feasibility and relevance. To better analyze their impact on the operation of a treaty it is necessary to group them into distinct categories on the basis of some distinctive properties. Two such categories are *qualitative* and *quantifiable.* The latter have attributes that can be measured, analyzed and evaluated. Qualitative attributes cannot be measured and, consequently, are interpreted subjectively. Table 1 gives some examples of quantifiable and qualitative attributes from some selected treaties [2].

It is not difficult to see that qualitative goals cannot be evaluated using objective criteria. How does one measure the pursuit of “negotiations in good faith on effective measures relating to cessation of the nuclear arms race at an early date”, or not to “assist, encourage or induce in any way, anyone to engage in any (prohibited) activity…"? Even quantifiable goals with measurable attributes present difficulties. Although it is relatively easy to assign measures for instantaneous events such as nuclear test explosions, it becomes more difficult for goals that relate to processes such as development. For example, to measure the obligation to “never…develop…chemical weapons” requires definition of the development process and its associated attributes. Another difficulty with quantifiable goals is associated with their feasibility. The idealistic goal of not carrying out “any nuclear weapon test explosion” is, in practice, modified by the limitations imposed by the detection technologies at some acceptable cost, in this case the number and locations of the stations of the International Monitoring

	Treaty Goals - Obligations of States Parties			
Attributes	NPT	CFE	CWC	CTBT
Quantifiable	• not manufacture • not transfer • not receive nuclear weapons	• maintain a secure, stable and balanced overall level of conventional armed forces in Europe • eliminate capability for launching surprise attack	• never develop, produce, otherwise acquire, stockpile, retain, transfer • never use • destroy facilities and weapons	• not carry out any nuclear weapon test explosion • prohibit and prevent any nuclear explosion
Qualitative	• not in any way assist, encourage or induce other States to acquire nuclear weapons • not seek or receive any assistance from other States to acquire nuclear weapons • pursue negotiations on cessation of nuclear arms race and on complete nuclear disarmament	• not threat or use force against territorial integrity, or political independence of any State • prevent any military conflict in Europe • achieve greater stability and security in Europe	• not engage in any military preparation for chemical warfare • not assist, encourage or induce in any way anyone to engage in any activity prohibited by the manufacture of chemical weapons	• refrain from causing, encouraging, or in any way participating in the carrying out of any nuclear weapon test explosion or any other nuclear explosion

Table 1. Quantifiable and qualitiative attributes of selected treaties.

System [3]. Increasing the number of monitoring stations around the globe would lower the detection threshold at an increased cost. The issue of cost, including intrusiveness and potential loss of trade secrets, has also become a stumbling block on the verification issue for the BWC [4]. Cost also plays a role in modifying treaty goals. In the CWC a toxic chemical used as chemical weapon in World War I was not placed in Schedule 1 along with the other toxic chemicals used as chemical weapons but in Schedule 3, with much less stringent verification procedures, because it is produced for non-prohibited purposes in very large quantities. Applying the verification provisions for Schedule 1 would entail unacceptable costs in terms of resources and intrusiveness. The question then becomes, how does one determine what is an acceptable level of cost? The quantifiable goals can be with acceptable cost can be defined as feasible, while those entailing unacceptable costs become non-feasible.

Another perspective for viewing the treaty goals as reference inputs is their internal consistency and overall relevance. The treaty on Conventional Forces in Europe (CFE) is a good example. It became irrelevant by the time it entered into force, because the threat of surprise attack had already disappeared [5]. How significant is the counting of armaments toward achieving the objective of preserving peace and stability in Europe? Not all treaty goals are of equal value. Thus, in order to be able to perform an objective evaluation of the performance of a treaty goals need to be classified in order of significance relative to the objectives of the treaty. The idea of ranking is incorporated into the assignment of toxic chemicals into Schedules on the basis of risk to the CWC with Schedule 1 chemicals posing the highest risk and Other Chemical Production facilities the lowest [6]. The need for internal consistency is evident by the issues arising when a treaty is not universal. The broad objective of the NPT is “prevention of wider dissemination of nuclear weapons”. It can be argued that the safeguards system has been reasonably successful in achieving that goal among the member States; although the abandonment by South Africa of its nuclear weapons program was due to internal political considerations [7]. Yet, nuclear weapons have proliferated among the non-NPT States. If the prevention of proliferation is to be the paramount objective, obligations of the States Parties with respect to States outside the treaty domain need also be specified. Ambiguity in obligations is also present in the Kyoto Protocol in which the question of the binding nature of compliance has not yet been resolved [8].

2.2 Process models

One of the biggest sources of uncertainty in the verification of compliance is the lack of complete and precise definitions of the set of activities that come under the purview of a treaty. For some treaties such as the CFE these are reasonably well defined. The CFE has compiled lists of treaty-limited equipment (TLE), sets quantitative ceilings for each category and specifies geographical boundaries where the treaty applies. In most other treaties there are uncertainties about the scope of activities covered by the treaty. The cause of some of these uncertainties is the genuine inability to develop precise definitions. For example, the expression "any nuclear weapon test explosion" in the CTBT leaves open the question of what precisely constitutes a test explosion; in other words, how far can the weapon development process can be advanced without crossing the "test" threshold? A much more difficult issue is that of dual use items or activities. Where and how does one draw the line between allowed and prohibited? The BWC offers the best example of the "dual use dilemma"; biochemical agents and techniques for administering them developed for the benefit of mankind can also be used for destructive purposes [9]. Similar, but not as crucial dilemmas arise from technological developments; most instruments that have peaceful applications can also be used in the development of prohibited weapons.

The difficulties in making precise definitions of the activities covered by a treaty could make problematic the implementation of the enhanced safeguards under the Additional Protocol in a politically neutral manner. For declared nuclear activities the process is the well defined nuclear fuel cycle with a beginning, an end and precisely defined intermediate steps for each type of facility. On the other hand, under the Additional Protocol ambiguities may arise on a number of points as for example, in drawing the line between "theoretical or basic scientific research" and "nuclear fuel cycle-related research and development activities" [10]. A much bigger challenge is the attempt to view the

State "as a whole" and apply State-level safeguards in order to obtain a "comprehensive ‘picture’ of a State’s nuclear programme and nuclear ambitions and identify any potential indications of diversion of nuclear material or of undeclared nuclear material or activities" [11]. There is a fundamental difference between declared and undeclared facilities and activities. In contrast to the well defined activities associated with the nuclear fuel cycle of declared facilities, the activities within a State are highly complex; in addition, there are few models of clandestine activities as the operation of the Khan network has shown [12]. Trying to separate which activities may have some relation to a clandestine nuclear program using technical in contrast to political criteria may not be an easy task particularly when the objective is to identify nuclear "ambitions". There is no known model of using measurements to predict intentions of States with some reasonable degree of confidence. In order for the enhanced safeguards to provide reasonable assurance that undeclared nuclear activities could be detected by creating a comprehensive picture of a State there needs to be a fairly well defined set of activities that would be indicative of clandestine diversion or nascent intention to develop nuclear weapons. To do so, two major obstacles need to be overcome. The first is that intentions change depending on external conditions. The second is the link between industrial development and the ability of a State to produce nuclear weapons. It may be easier to identify suspect activities in a developing State with basic industrial infrastructure than in a developed State with extensive declared nuclear activities. The challenge for the IAEA is to come up with evaluation criteria that are State-neutral. As a first step, it is necessary to identify physical processes within a State that could be associated with some phase of a clandestine nuclear development program.

Another factor affecting the completeness of the activities falling under the purview of a treaty is the influence of various affected parties on the negotiations. The best example of the impact of such influence is the verification regime of the CWC. One of the goals of the CWC verification system is the detection of diversion from permitted uses of Schedule 2 chemicals that pose a high risk to the objectives of the treaty to prohibited ones. Yet, detection of diversion is impossible [13], because no materials balance approach has been incorporated into the treaty; during the negotiations the chemical industry had a strong influence and was able to limit the extend of the verification activities on the grounds that it would be intrusive, costly and a threat to confidential business information.

Ideally the treaty processes on which monitoring and verification is applied should be closely linked to the feasible treaty goals. The looser the linkage, regardless of the reasons, the lower will the probability of detecting non-compliance be. Thus, the existence of qualitative and/or infeasible goals in combination with incomplete specifications of the processes covered by a treaty increases the uncertainty of detecting non-compliance and decreases the confidence level in the outputs of the verification regime.

2.3. Treaty monitoring as measurement systems

The information generated by treaty monitoring systems may be grouped into three major measurement categories according to their sources. The first comprises reports, submissions, declarations, notifications or other similar words that denote measurements generated by each State and sent, at specified instances, to the other States, an international authority or both. The second category consists of measurements generated by inspectors either through human sensing, instrument readings, or extraction of data from files. The third covers all instrument readings regardless how the measurements are done, in situ or remotely, without the intervention of inspectors. As with any measurement system, treaty monitoring systems entail uncertainties, which affect the accuracy of detecting non-compliance.

The performance of detectors is optimized on the basis of the statistical characteristics of the signals to be detected and of the uncertainties (noise) associated with the measurements. Thus, detection of non-compliance requires knowledge of the characteristics of both. Of the three categories of data sources, the one about which we have the most information and know how to deal with is that of the instruments. Although some instruments are specifically designed for treaty-monitoring purposes [14], the majority is used in a wide range of applications of which, treaty monitoring is a very small part [15]. As a matter of fact, instruments designed specifically for treaty verification purposes create a unique problem. Because of the limited demand, their production is not cost-effective and their supply cannot be assured [16].

Concerted efforts are being made to find ways to use commercially available instruments in treaty monitoring systems or at least to design instruments that could be used in as many as possible treaty monitoring systems. One category of monitoring technologies comprises instruments carried by inspectors to specified locations. The broadest category by far is that of remote monitoring which is further subdivided into two major categories. One consists of instruments in situ transmitting locally collected information to a remote location. The second, properly referred to as remote sensing, comprises instruments collecting the information remotely and processing the data either at the point of collection or at some other remote location. Remote sensing covers technologies detecting signals across a wide spectral band that includes optical, infrared, multi-spectral, radar and hyper-spectral images. While remote monitoring by in situ instrumentation is applicable in cases where the measurements points are known and are not expected to change over time, such as the safeguarded nuclear facilities under the NPT and the monitoring stations of the International Monitoring System of the CTBT, remote sensing is applicable to cases where the target is a broad geographical area, or a specific location where access is problematic for whatever reason.

To utilize the potential of remote sensing technologies in treaty monitoring, a major obstacle needs to be overcome, namely, the identification of target or targets to be monitored. Satellites can generate very detailed images on the ground and can be used in many treaty monitoring applications [17], but, no mechanism has been found to use them in detecting, *e.g.*, undeclared facilities [18]. In monitoring for non-compliance remote sensing can be a useful tool when used in combination with other indicators in order to identify locations where suspect activities might take place. A promising area where some work is being done is the combination of satellite imagery with seismic data from the International Monitoring System of the CTBT to improve the location detection capability of the verification system [19].

For all the state of the art capabilities of instruments and the powerful computational tools, very little is known on how to combine them into integrated monitoring systems optimized for detecting non-compliance. While communications allow the transport of large quantities of data at high speeds over great distances combining, filtering the data to obtain information that is better than the some of the parts generated by the individual sensors remains a challenge. The problem is immediate and acute as it arises in the processing of information from diverse sources under the integrated safeguards system. Most of the discussion centers on the data processing technology, while little is known about the algorithms to be used in processing the data in order to extract information that would aid the detection of non-compliance [20][21][22]. Such algorithms cannot be developed without first identifying the activities which would be in non-compliance and describing them using quantitative models. Modeling is perhaps the most important element in the operation of a treaty. Its applicability ranges from describing specific physical phenomena such as the transport of pollutants or radioactive particles in the atmosphere to understanding the decision processes that lead to a decision on whether a State should or should not sign a treaty [23][24][25].

2.4 Sources of error in detecting non-compliance

As it has been mentioned earlier, non-compliance is the discrepancy (error) between the obligations assumed by the States with respect to the required characteristics of the processes specified in the treaty and the information about these characteristics generated by the monitoring system. In an ideal system discrepancies would only be caused by non-compliance. In reality, they are caused by uncertainties present in the various components of the system which are unrelated to non-compliance. To begin with, for qualitative treaty obligations there is no measurement mechanism to collect data for use in detecting non-compliance. Even for measurable treaty obligations, if the cost of the monitoring system which would be necessary to detect non-compliance is above a defined threshold is unacceptable or if current technologies to implement it are not available to allow detection below that threshold, in effect some non-compliance cannot be detected. As a result, non-compliance is subject to objective analysis only for those treaty goals which are *feasible*.

A more significant source of error is the incomplete specification of the processes covered by the treaty. As we have seen, for most treaties, the measurements on the processes that are subject to monitoring may not generate the proper type, quantity or quality of data which are necessary in order to create a picture of the process which would allow detection of non-compliance. The limitations imposed by the CWC on the monitoring of industrial activities are a good example of intentionally incomplete specifications. Oversimplification of atmospheric transport processes results in errors in the detection of non-compliance with the Kyoto Protocol. By contrast, the challenge facing the IAEA in developing an effective integrated safeguards system is to develop a workable definition of the activities which could be monitored under the Additional Protocol.

Another source of errors in detecting non-compliance is the monitoring system. Measurements are subject to random errors which introduce uncertainties in the detection of non-compliance. The impact of those uncertainties on the detection probability can, in most cases, be calculated, because the statistical characteristics of the instrument noise and of inventory sampling errors are either known or can be measured. More significant are the errors introduced by deficiencies in the design of the monitoring system itself. Even if the process to be monitored is completely specified, if the number and frequency of measurement is not sufficient to create a complete picture of the process, detection of non-compliance will suffer. For example, the reporting requirements for Schedule 3 chemicals in the CWC are limited to reporting production figures per year in ranges making detection of diversion impossible. Similarly, the frequency and scope of inspections at chemical facilities have not been determined by the verification requirements of each facility but have been restricted on the basis of extraneous considerations. There is a fundamental flow in the habit of specifying the architecture and operation of the monitoring system in the treaty text. The design and operation of the monitoring system for any process is a purely technical subject to technological and economic constraints. Making the specification of the monitoring system an element of the treaty negotiations introduces political considerations that inevitably limit the ability of the system to collect sufficient information in order to maximize the probability of detection of non-compliance. In addition, technology constantly evolves; changes in manufacturing processes, new measurement techniques, improved accuracy, faster communications and improved computational techniques make obsolete the monitoring system specified in the treaty text. History shows that changes in the text are next to impossible to make, in effect, perpetuating the sub-optimal operation of verification systems. Unless the design of monitoring systems is removed from the negotiating process and done professionally, the probability of detecting non-compliance cannot be maximized.

2.5 Minimization of discrepancies as a control function

The preceding analysis has shown that discrepancies detected by a monitoring system may not be attributed only to non-compliance; it may be caused by ambiguities in the definition of the goals, incomplete specification of the treaty processes subject to verification, deficiencies in the design of the monitoring system or any combination of these. To optimize the operation of the treaty in order to maximize the detection of non-compliance one needs to minimize the impact of the other sources of detected discrepancies. Figure 2 shows the actions that need to be taken for each of the different error sources.

Of the four categories of discrepancies the one least amenable to technical analysis is non-compliance, because no independent authority exists to enforce it. The remaining three, the problem of insufficient measurements can be solved technically, although, in the final analysis, the decision to accept the technical solution is political. The necessary modifications to the goals and activities can also be determined through rigorous analysis and implemented during the treaty review conferences, again, a political action.

Even if enforcing compliance is a difficult problem there is a need to develop some objective criteria for doing so. As a starting point it needs to be recognized that non-compliance is not a scalar function; it has many components; for example, not declaring an enrichment experiment in a laboratory has a different weight from assembling a nuclear device. In fact, one of the most common forms of non-compliance is absence of declarations or errors in them. A necessary condition for developing an enforcement mechanism is the ranking of the feasible treaty obligations on the basis of their significance. One then can seek enforcement mechanisms that are appropriate for each category. Devising such mechanism is a subject beyond the scope of this paper.

On the basis of the preceding discussion, a more detailed model of the treaty process than that given in Figure 1 is shown in Figure 3.

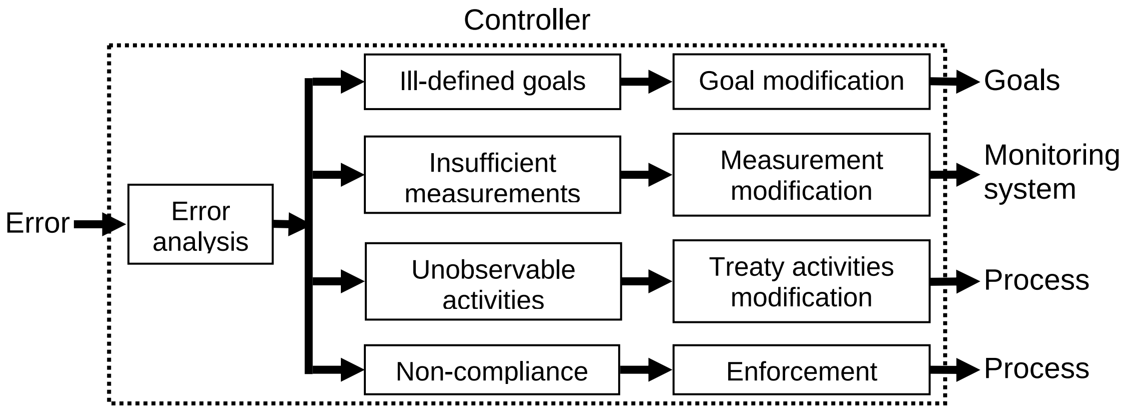

Figure 2. Causes of discrepancies in the detection of non-compliance and correction mechanisms

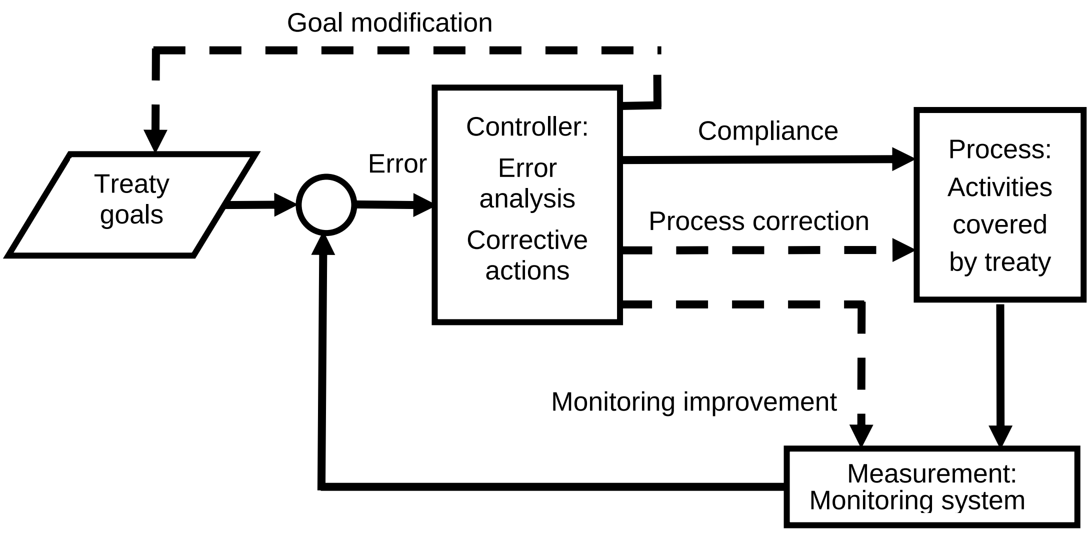

Figure 3. A treaty model taking into account the major categories of detector discrepancies.

3. Conclusions

Multilateral treaties requiring monitoring, verification and enforcement of compliance rely on measurements and evaluation of data for the detection of non-compliance by State with the obligations incorporated in the treaties. By using a feedback control system model we have shown that discrepancies detected by the verification system may not necessarily indicate non-compliance. Instead, they could be caused by systemic defects of the treaties. We have shown that defects could be found in the obligations undertaken by the States. More serious defects are built into the verification systems mandated by each treaty particularly in the specification of what, how and when information is to be collected and used in the effort to detect non-compliance. A major defect of such treaties is the notion that the purpose of the monitoring mechanism is verification of compliance. That is an impossible task. At best, the monitoring system can be designed to provide information maximizing the probability of detecting non-compliance. The formal approach presented in this paper describes an objective mechanism for doing so.

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Identification of high risk intermediaries in global networks transferring sensitive technology and information*

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Abstract. The management challenge facing modern non-proliferation regimes is one of long-term, multi-step technology diffusion. Because terrorist groups and other non-state sanctioned actors have demonstrated that sensitive technology can be acquired from both members and non-members of the export control regimes and then retransferred through intermediate transactions, technology diffusion in the global economy suggests that indirect and intangible technology transfer mechanisms are broadly influential in building capacity for weapons programs. This paper defines a methodology for investigating these transfer mechanisms and identifying patterns of sensitive technology trade within the context of global economic networks. Applying a network paradigm to sensitive technology transactions, statistical models of trade can infer underlying network structures that facilitate the diffusion of technology and information. This approach shows that relatively compact sets of intermediate countries can be positioned, in the sense of "economic distance", between suppliers and recipients to efficiently transfer sensitive technologies. By identifying the sensitive information network pathways and intermediaries of concern, policy makers can better evaluate enhancements to multilateral supply regimes and promote an overarching standard of behavior beyond existing export controls that facilitates legitimate, legal trade networks while reducing the risk of sensitive technology transfer.

1. Introduction

Traditional export control regimes proceed from the premise that selected items of "sensitive technology" can effectively be denied to potential proliferants for whom intentions or dual-use rationales are suspect. The global effectiveness of strictly bilateral policy in this mold was quickly recognized as limited, and the technologically well-endowed countries of the world have allied in a series of multilateral regimes (typified by the Nuclear Suppliers Group [NSG]) built on the same basic principle. Recent world events prove that even these multilateral regimes seem unable to prevent determined proliferants from acquiring weapons of mass destruction (WMD) technology. The economic theory on prohibition contends that as long as demand exists, even in markets where a particular good or service is controlled, supply has incentives to meet demand via the price mechanism. These conditions then facilitate the creation of a "black market" whereby items in demand can be transferred through uncontrolled channels. One such mechanism is transfer via intermediate actors.

The traditional view of a "black market" is a set of clandestine transactions in which both the sellers' and buyers' identities are obscured. Another possibility is a sequence of open transactions over which the coupling between sellers and ultimate buyers is obscured. This model is particularly compelling for dual-use technology transmission in voluntary multilateral regimes, and was described in concrete terms involving enrichment equipment and Iraq [1]. Recent revelations on clandestine trade networks offer another example that suggests networks may function as "gray markets", within which the force of export controls and other regulations is significantly diluted. Although export

* Based on exploratory investigation performed for the U.S. Department of Energy Office of Export Control Policy at Pacific Northwest National Laboratory by T.W. Wood, B.A. Reichmuth, M.R. Weimar, and R. O'Brien.

controls may attempt to monitor and control WMD proliferation from a material perspective, it is technology and information diffusion in the global economy which also suggests that indirect and intangible transfer mechanisms are broadly influential in building capability for weapons programs.

Understanding the long-run risk of technology transfer in the global economy requires viewing it as a network in which technology transfer motivated largely by economic processes is sometimes subverted to weapons uses. This paper describes a methodology for which the structure of this network can be illuminated using economic analysis tools adapted to and focused on the sectors that supply sensitive technology and the legitimate dual-use sectors they supply. A key point is that economic considerations largely determine the structure of the network, which determines “paths of least resistance” for the diffusion of sensitive technology to potential proliferants. This approach shows that a relatively compact set of intermediate countries is positioned, in the sense of “economic distance” to both the supplier groups and the sensitive countries, to transfer sensitive technology efficiently. The difficulty of preventing sensitive information transfer can not be overstated. For example, the only international frameworks for controlling information flows can be demonstrated by intellectual property (IP) laws which have been proven ineffective, in particular by their deliberate focus on protecting profits and not necessarily on preventing the spread of information.

Although the policy implications of this work have not been fully analyzed, a theme that emerges strongly is one of explicit recognition and special treatment for countries that are well positioned to act as intermediaries in technology transfer.

2. Context for the Problem

The time scale of the non-proliferation problem and the export control measures that contribute to its management is a long-term one, measured at least in decades and perhaps longer. This time scale is set by deeply rooted technical and socio-political factors. The first of these is the probable timetable for abandoning nuclear weapons capability by the current nuclear weapons states under the Non-Proliferation Treaty (NPT). However unlikely and difficult, until this is done there will be incentives (perceived or real) for countries to seek nuclear or other WMD capabilities. Second, sustainable development of the third world which is necessary to place an economic foundation beneath global security is a gradual process. Even without the constraints posed by limiting fossil fuel usage due to global warming concerns, this will take many decades at current rates of convergence.

In addition, the prospect for increased utilization of nuclear energy in developing regions (particularly Asia) will force expanded security related responsibilities for the private sector and the non-proliferation regimes [2]. Reactor capacity uprates, life extensions, aging facility replacement, and population growth coupled with international liberalization of energy markets are expected to drive the demand for nuclear energy within the next several years. Within the next 20 years, the World Nuclear Association reports the world’s nuclear powered generating capacity is expected to grow between 71 and 175 GWe with a 13,500 - 33,000 MT increase in reactor uranium requirements [3]. Potentially helping to spur a nuclear expansion, internationally-focused nuclear programs such as the U.S. Global Nuclear Energy Partnership (GNEP) will ultimately increase the legitimate transfer of nuclear technology. Therefore it may be more applicable that new multilateral trade controls not only focus on technology transfer based on illicit trade, which is ultimately held more accountable by short-run export controls, but for sanctioned trade which will certainly increase, thus increasing proliferation risk through a wide spread nuclear renaissance.

In the short-run, specific (i.e., list-based) measures must target the direct transfer of key items that currently constrain known proliferants. Success in the short-run may better enable us to deal with the long-run problem. In the long-run, economic forces are powerfully aligned to cause both the advance of technology and its proliferation throughout the world. This suggests that the objectives of non-proliferation policy generally, including export controls, must be both short- and long-run.

Generally speaking, the advance of technology is the single most dominant feature of the modern global economy. The incentives for technological progress come from its productivity-enhancing (and thus profit-generating) effects. The incentives for diffusion of technology among the world’s countries are also powerful. The coupling of increasingly skilled labor with even outmoded (by the standards of the G-8) technology could bring enormous productivity and welfare gains to the developing world.

The fundamental and very powerful nature of these incentives insures that *technology* proliferation is inevitable. We must look beyond traditional export controls to manage its unintended consequences.

3. The Economic Model

3.1. An Economic Perspective on Technology Transfer and Sensitive Technology

From a long-run perspective, technology is transferred by a myriad of processes that include, but go beyond, the purchase of equipment and materials in international trade. Aside from equipment, devices, and other means of production "technology" is essentially information and is intangibly transferred by its utilization in legitimate economic applications, by exposure to its use in scientific settings, and by communication of the underlying science in both R&D and formal educational settings. If one considers the full spectrum of these information transfer mechanisms, a process of technology diffusion among countries can be inferred.

Indeed, a diffusion model has long been the standard paradigm for analysis of long-run technology transfers, and this model of technology transfer exists in many contexts [4]. A diffusion-modeling context allows some direct inferences about the rate of transfer, which is a key element in assessing the probable time involved in attaining several thresholds of technological competence. Such a context also strongly motivates interest in the inter-country "paths" along which diffusion occurs and directly identifies a set of intermediaries likely to play important roles in transfer between any two countries or sets of countries. Within the export control community, the possibility of intermediaries in the transfers of technology has been recognized for some time as a short-term issue in the form of transshipment or dual-use fronts. It is becoming recognized as a longer-term issue as well, in cases where close trading partners of proliferant countries are increasingly technologically well endowed [1].

Generally, the logic of technology transfer by diffusion is one of proximity (or distance) relations among actors, which serves to mediate transactions of various types. A simple example applied to international trade is the gravity model, which predicts with good accuracy the magnitude of trade flows between countries based on their economic characteristics and some set of constraining or impeding factors including physical distance, language, and other cultural or political factors. Boisso et al. provide a good general review of international trade gravity model specifications [5]. Gravity models have been broadly employed as explanatory or predictive of trade using observed measures of activity and distance. Their success is widely acknowledged, and several economists have advanced theoretical considerations that would motivate their use [5][6][7]. Our methodology adopts a gravity framework for modeling sensitive technology trade, but reverses the computational flow of the model to focus on estimating the "effective economic distance" or "net impedance" for sensitive technology trade between countries or sets of countries. These distance relationships can then be used to define a network structure in which the "shortest" or "minimum impedance" paths can be identified. Since the network is defined using data on selected trade in sensitive technology items, these are the paths deemed preferentially likely as indirect routes for diffusion *of this technology* via trade.

3.2. Gravity Model Formulation

In a gravity model, trade between two countries *i* and *j* is taken to vary as some inverse power of distance. The term gravity model is from the analogy to Newton's formula for gravitational force. A simplified formulation is:

$$T_{ij} = [(A_i * A_j)^{\alpha}] / d_{ij}^{\beta} \qquad (1)$$

Where A is some measure of activity ("economic mass"), d is distance, and parameters α and β are correction factors minimizing some error criterion on T. From equation (1) the resulting measure for distance is [7]:

$$d_{ij} = [(A_i * A_j)^{\alpha}/T_{ij}]^{1/\beta} \qquad (2)$$

The activity variable A is typically a common measure such as gross domestic product (GDP). More sophisticated models could be framed around sector-specific formulations, where A would be the output of specific sectors (and T would be trade related to these[1]). T is typically *total* trade – that is, bilateral imports plus exports for *either* country *i* or *j*. Many models in the literature include numerous variables that condition trade in addition to the basic activity and distance terms.

Our application takes a different approach. We use a gravity model concept, and in particular, its associated measure of distance, to illuminate the *implicit structure* of trade networks. In this context, we note that "economic distance" is in fact a function of the multiple facets of physical distance, density of transport infrastructure, and cultural factors such as common language, politics, etc. Further, the combined effect of these variables *is not directly observable but can be inferred from observed trade and activity data*. This approach uses a "latent" or implicit concept for economic distance. The literature on social network analysis has recently developed a rich statistical framework that includes a concept of latent (unobservable) social space in which actors and their relations exist [8][9]. While we do not use the statistical estimation framework advanced in these papers, the concept of a latent space is the same.

4. Approach

Our preliminary work within this network analysis context was exploratory calculations to determine if the identity of strong ties in the trade network was robust over typical values of the gravity model parameters. This was the case, with the network structure largely insensitive to variations in the parameters of the underlying gravity model, within the range typical in the economics literature. We then used more specialized trade flows in lieu of total trade. This narrowing of the focus of the model has large effects on the resulting network structure and the resulting low-impedance paths for technology transfer. Rauch argues that trade in differentiated products with high search costs should be viewed as organized within networks [10].

It is important to remember that the economic distance measures calculated in this way are significantly different from geographic distances. While geographic distances are one factor that influences economic distance, many other factors including language [11][12], transport costs, culture [13] and institutional trade barriers also influence economic distance. By comparing economic distance measures against the corresponding physical distances, a weak but present correlation indicates that physical distance indeed explains only a small fraction of the total impedance to sensitive technology trade.

Generally, measures of distance are typically constrained to follow some axioms that ensure they are "metrics" and which follow from basic concepts of Euclidean spatial organization. Among these is the property that the distance between any two points must be less than the summed distances to a third point. This property (known as the "triangle inequality") is sometimes used as a constraint when formulating statistical estimates of economic distance. Adopting such a constraint when exploring the trade routes problem throws potentially useful information away[2], since one seeks an indicator of which indirect or circuitous routes sought present lower effective barriers to trade. As it turns out, using the typical gravity model formulation for economic distance often employed in trade theory, but treating distance as a quantity that cannot be directly observed, leads to measures of inter-country distances that are frequently not metric. We exploit this non-metric property to define "shortest" or "minimum impedance" paths between every pair of countries in the model. Despite the

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[1] For example one might not assume that trade in wheat or oil would be particularly diagnostic to the propensity to trade in computers.

[2] There are other reasons not to insist on metricality in a distance measure for trade networks. The reality of proliferant behavior in technology markets is that "direct" acquisition is often carefully avoided. Even for trade networks in general, a concept of distances must accommodate the fact the "direct" distance from some *i* to some *j* simply does not exist.

large number of possible paths given a multiple member network, this can be done relatively quickly using a simple recursive minimization algorithm.

The results from simple network models of this type can give a reasonable first-order understanding of the relative degree of risk among countries for indirect technology transfer for a broad class of transfer mechanisms *involving trade*. Within this general conclusion there are several sets of important qualifications that should govern the refinement and use of such models and results.

First, there is the general observation that “traffic” in sensitive technology is not strictly or sometimes even essentially *economic*. As shown by recent events, there is often a strongly personal or ideological component to the social networks involved. A model such as ours that uses only trade data fails to represent these factors. The broader social network literature offers many possible fruitful directions for generalization in this domain. We remain convinced that the economic component is dominant and thus offers the best starting point for building such network models.

Second, the methodology used here is highly abstract in the sense that it represents an unspecified class of transfer mechanisms. While all are presumed to involve trade, there is no specification of the underlying process. Thus, very short-term, intentionally deceptive mechanisms (transshipment, re-export) are implicitly included, as are longer-term, less intentional mechanisms such as diffusion of intangible technology involving labor force mobility and similar factors. The explicit consideration of these mechanisms will tend to both sharpen and complicate the network modeling by accounting for such variables as the stock of technology in a country, the ability to absorb technology as manifested in scientific infrastructure, patents, etc. The use of a metric that explicitly accounts for the technological character of an economy would be attractive. Much of the literature on technology diffusion in general suggests useful approaches.

Third, the model parameters used in this analysis are such that the distance relations are invariant to the scale of trade in the following sense. If, for some country, both the total trade with a partner and the partner’s GDP were to increase by the same multiple, the economic distance to that partner would be unchanged. An implication of this is that the distances to any partners for which total trade and that partner’s GDP is in a fixed ratio will be equal. Thus, a very small trading partner could be deemed to have the same economic proximity as a very large one, while having an absolute trade volume that was much less. This scale invariance property of the distance relation is appropriate when considering technology transfers that are small relative to total trade. If the transfers of interest are a substantial fraction of total trade, the issue of capacity constraints on the network links becomes relevant.

5. Conclusions

Application of formal statistical models to estimating the structure of sensitive trade networks gives encouraging results. The estimated structure of the networks is relatively insensitive to variations in the underlying gravity model parameters, the set of important intermediate countries identified is generally compact and well defined, and the frequencies with which they appear as intermediaries is statistically significant on simple tests. It is tempting to begin to apply such models directly to the assessments of technology transfer risk and the associated policy problems of country status and regime membership.

We would argue that while some applied analysis with this type of methodology is appropriate, there is also much to be done in the way of refining this approach. The concept of shortest paths is an extreme-value statistic, and may be less important than some measure of total transmissivity (over all network paths) through each country. We have used very limited data from a temporal standpoint, and have not accounted for either the stock of technology in a country or the rate at which diffusion occurs along any of the possible paths. Broadening the modeling framework to account for these features is essential to fully understanding the correlation of modeling results with real risk of sensitive technology transfer.

Traditional export control regimes are centered around supplier groups. These are, in economic terms, producer’s clubs that agree to something more restrictive than free trade in what is felt to be common self-interest in a stable world. In most cases, there is lack of agreement within a regime on whom the targets of export restrictions are or should be. The existence of a distinct, relatively compact set of probable intermediaries, which is economically positioned to efficiently transfer technology to targets of mutual concern, opens at least one interesting possibility for strengthening export controls. That possibility is simply to broaden the supplier-based regimes to include important intermediaries. Perhaps a special class of membership might be created to reflect the manner in which control should be exercised and the non-reciprocal nature of obligations that might be entailed. An intriguing aspect of such a regime is that the identity of the intermediate country set might well be better defined and/or less controversial (for technical and political reasons) than that of the ultimate targets of export restrictions. Ultimately, identifying the sensitive information network pathways and intermediaries of concern will contribute to developing and successfully implementing an overarching standard of behavior beyond existing export controls that facilitates legitimate, legal trade networks while reducing the risk of sensitive technology transfer.

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Development of an academic course in safeguards and nuclear non-proliferation at Swedish universities

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Abstract. The purpose of this paper and presentation is twofold: 1. to present a new course on nuclear non-proliferation and safeguards aming at various university programmes in Sweden. As a first step, the course will be given to civil engineering students at the Uppsala university. For these students the technical aspects of safeguards will be emphasised. Later on, students in international relations at Stockholm university will be addressed with a focus on the non-proliferation aspects. 2.to present a elaborated course material that will be used in both academic courses and for training of personel at the Swedish Nuclear Power Inspectorate and in the Swedish nuclear industry.

1. Introduction

Several national and international reports and studies have over the last decades expressed concern about diminishing and disappearing of knowledge in the nuclear technology. For example, as early as in 1990 the US National Research Council reported a strong reduction in nuclear engineering students, an extremely high age of faculty members and shutdowns of research facilities at American universities. In a survey from 2000, OECD arrived at a similar conclusion regarding the whole industrialised world. Parallell with the growing disinterest in nuclear technology among young students, it may be noted that senior scientists, industrial key persons and, in fact, significant parts of the staff in the nuclear industry are about to retire. The conclusions of the reports and studies can be summarised in one recommendation: The world needs to educate engineers and all kind of experts and leaders in the field of nuclear power in order to fill the future expected gaps. This is true, especially if we take into account emerging technologies such as the Generation IV inititaive.

Higher education in this field has been dismantled in many countries during the past few decades which today poses real challenges as the buildup of academic competence is a long-term project. Nevertheless, the academic society has to prepare for and make sure that our national industries, authories and international organisations, dealing with nuclear energy and non-proliferation issues, can employ qualified people to enable a satisfactory level of knowledge to meet the demands in terms of producing energy and protect and secure the nuclear material from illegal and dangerous use.

One important step forward to meet these challenges was the creation of the World Nuclear University (WNU) on September 4, 2005, in London. The WNU, which was founded by IAEA, the Nuclear Energy Agency of the OECD and the World Nuclear Association of Nuclear Operators and the World Nuclear Association, is a network cooperation project consisting of 29 nuclear research centres from different countries and international organisations. The main focus is to strengthen the education dealing with the use and maintenance of nuclear energy globally. In addition to this global approach, the European Commission called for a European solution to the problem of diminishing knowledge.

As a consequence, the European Nuclear Engineering Network was created which in many respects have the same objectives as WNU. In this network between European universities and research institutes, many important projects have been initiated as to improve the nuclear education and research within the EU region.

There is, however, one important ingredient missing in both the WNU and ENEN in these education ventures: nuclear non-proliferation and safeguards. For example, out of 250 courses that are given within this European network programme, not a single course was dedicated to safeguards or other areas within the nuclear non-proliferation regimes. Against this backdrop, European Safeguards Research and Development Association (ESARDA) took a decision to extend its mandate to deal with education on nuclear non-proliferation as a part of training and education programmes. As a result, a first task group consisting of teachers, researchers and experts from EU universities, IAEA and research centres was set up in 2003. Moreover, a prototype course was worked out in 2005 at the Joint Research Centre (JRC) site at Ispra, Italy. The following year, the first ESARDA course was held at the same facility in Ispra. The course will be repeated on yearly basis for students from the EU-region (http:/esarda2.jrc.it/internal_activities/WC-MC/Web-Courses/index.htm).

The authors of this paper are participants of this ESARDA working Group representing Sweden and the Swedish Nuclear Power Inspectorate and we were assigned for a project to initiate academic nuclear non-proliferation courses at Swedish universities. The purpose of this paper is to describe this planned academic course. The second purpose is to present a course compendium that has already been worked out to serve as an introduction to safeguards and nuclear non-proliferation.

2. The Planned Academic Course in Safeguards and Nuclear Non-Proliferation

During the past few years, the student interest for nuclear technology has rapidly increased at the civil engineering programs at Uppsala university, Sweden. Last year over 100 civil engineering students participated in courses with relevance for nuclear technology. In a time where the Swedish nuclear industry anticipates a high retirement frequences within the next 5-10 years, this development is followed with great interest. In some of these courses safeguards principles and nuclear non-proliferation have been a part of the curriculum. Many students have found the subject interesting and asked for a deeper knowledge in the nuclear material control management and international co-operation in non-proliferation issues. However, such education has not been possible to offer in Sweden. Therefore the Swedish Nuclear Inspectorate, Uppsala University and Stockholm University started a co-operation to work out a course material that can be used in both academic courses and for training of personal at SKI and in the Swedish nuclear industry. As a first step this course is planned to be given for civil engineering students at Uppsala university, starting in the end of 2006 in order to gain experiences and feedback. In a second stage we plan to give the course for International Relations students at Stockholm university. The compendium mentioned above is written in Swedish and is presentently being translated into English.. The compendium, entitled “Nuclear safeguards and non-proliferation” contains two main parts: 1. Nuclear Non-Proliferation – an introduction. This part is mainly dealing with a historical and legal background of nuclear material control system; 2. Safeguards and verification. This part deals with the physical principles of various detectors such as semiconducting detectors, scintillators and various neutron detectors. Also the application of these detectors for safeguards instrumentation is described together with a brief presentation of data analysis principles.

Part 1: Nuclear Non-Proliferation – an introduction.

Section A: . What is nuclear non-proliferation?

1. **Introduction.** In this chapter the general background of nuclear energy, nuclear weapons and nuclear non-proliferation are presented. Central themes are: What is a nuclear weapon and what is needed to manufacture nuclear weapons? The production of uranium, plutonium and weapons-grade nuclear material, and fission and fusion processes are explained in a non-technical language.

2. **The emergence of a gobal nuclear material control**. In this chapter the efforts to create a global nuclear non-proliferation system in the period 1942-2005 is analysed. These efforts are analyzed in different phases. The starting point is the Second World War and the race between the United States and Nazi Germany to manufacture nuclear weapons and the consequences of this race for the nuclear material control shortly after the WW II. The Cold War period, with the strivings for an establishment of a UN based international nuclear safeguards system, is discussed; i. e. the creation of the IAEA and the emergence of the NPT-regime and its effects are analyzed in the context of the bipolar conflict between the United States and the Soviet Union. The latest trends, after the Cold War until today, such as the the Iraqi secret programme to produce nuclear weapons and its implications on the NPT-system and the debate on wether Iran is preparing a manufacturing of weapons of mass destruction are also looked into and discussed.

3. **International regimes – what are they good for?** In order to understand the functioning of different control regimes within the nuclear non-proliferation field, different theoretical concepts have to be presented. Different international relations theories such as classic realism, neo-realism, neo-liberalism and liberal institutionalism are discussed in order to answer the question: What is an control regime and how does they work?

4. **Export control, physical protection and transport safety.** Different export control regimes such as the Zangger Committee and the Nuclear Suppliers Group (NSG) as well as physical protection, transportation and illegal trafficking of fissile material are important topics of this part.

5. **Safeguards/nuclear material control.** This chapter emphasises the safeguards principles and the functioning of the IAEA inspection system: Methodologies, definitions, inspections routines are described and discussed, as well as the Additional Protocol and its implication on the efforts to strengthen the NPT-system. The chapter summarises this section with a answer to the question: Is it relevant to talk about a functioning global nuclear non-proliferation system today?

Section B: The Swedish nuclear material control system in practice

Sweden is used as an pedagogical example on how the global nuclear non-proliferation works in practice for a state that is a member of the IAEA, has ratified the NPT, and have safeguards agreement (including the Additional Protocol) in function. This section includes the following chapters:

1. The Swedish nuclear profile

2. The Swedish nuclear material control system and its functioning

3. Export control in Sweden

4. Physical protection and Transport security in Sweden

Part 2: Nuclear Safeguards and verification

The main focus in this first edition of the compendium is put on the NDA and the C/S. In the next edition we plan to expand the course by including the DA and to give thorough account for relevant data analysis and uncertainty estimations of measured quantities. In this present edition, this section is subdivided into three parts:

A. An introduction where the physical background of the utilisation of nuclear power is treated. Here the fission process is covered as well as various aspects on neutronics and the corresponding implication on fuel technology. The emphasis lies on light water reactor technology. In this part a detailed account on the basics of detection of ionising radiation is carried out. Here the emphasis is put on neutron and gamma-ray detection.

B. In this part various NDA techniques are described. Both such techniques that are currently approved e.g. the FDET but also emerging techniques e.g. tomography.

C. Here containment and surveillance are covered where an account is made of the various techniques available today. Issues of a more “philosophical” nature are also discussed. Such issues can be for instance to what extent the output of a surveillance camera really reproduces what is going on in a scene. This issue is connected with the notion of performance and assurance, which is accounted for briefly.

Safeguards information challenges

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Abstract. The purpose of this paper is to provide the background and broader vision in which the various presentations and posters on information related matters, prepared by staff of the Department of Safeguards, find their place, exploring solutions to the challenges faced. At a time when the IAEA is expected to be a reference for the assessment of nuclear proliferation issues, it faces major challenges with regard to the information needed to provide the international community with independent, impartial, timely and soundly based conclusions regarding the non-diversion of declared nuclear material and the absence of undeclared nuclear material or activities in States. The drawing of such conclusions assumes that all relevant information is accessible and has been put in the relevant context. Unfortunately, because of legal constraints under which it operates, or because of limitations on resources available for collecting existing but difficult to reach facts, or simply because of the overwhelming aspect of the information needed to properly evaluate proliferation issues, the IAEA needs to enhance the way it collects, processes, analyses, evaluates and disseminates safeguards relevant information.

The search for better, broader and deeper information through dedicated collection means is an ongoing effort aimed at filling the gaps that constrain the development of the coherent picture that analysts need to be comfortable with the results of complex analyses. Implementation of well-selected information collection tools is *sine qua non* for effective IAEA verification capabilities. Beyond collection, the establishment of a proper information architecture and documented business processes is a prerequisite for ensuring that the available information is accessible to those with a need to know. At the same time, information must be sufficiently protected by appropriate security measures, particularly when hacking has become an intellectual sport and institutional security attacks may be politically motivated by those wishing to weaken the IAEA's credibility.

Ultimately, all information processing is aimed at deriving focused knowledge needed by all of the parties that contribute to the IAEA's mission, through field activities or headquarters investigations that address global concerns by appropriate response. Moreover, this knowledge must survive the assault of time, given that proliferation issues can span many decades.

1. Introduction

As recorded by Member States in the IAEA Medium Term Strategy for 2006-2011, the major objective of the IAEA's nuclear verification mandate is to provide credible assurance to the international community that States are honouring their safeguards obligations. Building on its extremely effective actions under its Security Council mandate in Iraq— understanding in detail and dismantling a past undeclared nuclear programme while implementing a credible ongoing monitoring and verification plan in the 1990s as well as providing an accurate assessment in 2003 with regard to the absence of any resumption of nuclear activities — the IAEA has become, and needs to remain, a reference for the assessment of nuclear proliferation issues. The award of the 2005 Nobel Peace prize to the IAEA is recognition of the essential role it plays on behalf of non-proliferation.

How can an organisation initially built on the verification culture of nuclear material accountancy cope with the challenges associated with the current aspects of the available relevant information? Given that the IAEA's main undertaking is to 'prove a negative' (that is, the absence of undeclared nuclear material and activities), what should be done to meet these expectations?

2. The context

Nuclear proliferation is (fortunately) a complex endeavour that requires significant assets. Everything starts with the ‘political will’, which can come from geopolitical tensions, national feelings of insecurity, or attraction to the apparent status of world power for the owner of a nuclear weapons arsenal. Proliferation also requires substantial financial resources. Although a State struggling to find adequate income to feed its population may be less likely to proliferate, history demonstrates that this may not be a valid argument. The very existence of an adequate nuclear infrastructure, comprising specific facilities, adequate energy supplies and transportation means can be viewed as an advantage for a body conducting nuclear verification activities, compared to the challenges associated with the verification of the development of chemical or biological weapons of mass destruction.

Nuclear materials of specific quality (e.g. high enriched uranium, plutonium), in the right quantity (kilogram not microgram), remain the ‘choke point’ for justifying without reservation the fact that safeguards approaches focus foremost on preventing the diversion of nuclear materials.

The scientific and technical basis of a nuclear programme is necessarily broad and diverse, on the order of magnitude of at least hundreds of workers, if not thousands, with multiple skills – physicists, engineers (e.g. mechanical, electrical), chemists, and skilled and unskilled labour. All told, this group of individuals must possess multi-facetted knowledge required to deliver what the political powers expect. Such knowledge is not trivial. Currently, the amount of information (and disinformation) available on the Internet does not allow one to move forward concretely without an appropriate and significant R&D programme. Unfortunately, in the context of the proven existence of networks of the type discovered relative to Libya at the end of 2003, that aspect needs to be continuously reassessed.

Fortunately for the IAEA as a verification body, all of the assets needed for nuclear proliferation are sources of indicators and signatures that can help to detect possible undeclared nuclear activities. For such detection, the IAEA also has at its disposal significant assets. Member States have provided the IAEA with the relevant policies, financial resources and legal instruments, such as the definition of the IAEA’s rights and obligations and their temporary reinforcement through, for instance, resolutions of the IAEA Board of Governors or the UN Security Council. In addition, Member States have assisted the IAEA in the development of staff competence, through training and technology support that reinforces the IAEA’s capabilities.

The primary asset for the IAEA as a verification body is the legal right for access in the field, whereby inspectors can enter a State to inspect installations, inventory relevant materials, monitor facilities and interview operators and other counterparts. This access represents an exclusive ‘niche’ that the IAEA possesses and that no State acting alone can possess, except in very infrequent and often politically sensitive situations. The international nuclear verification community has also developed mature measures, from techniques to methodologies, building on decades of experience and on the specificities of nuclear materials and nuclear programmes.

Although sometimes overlooked as some often equate verification to inspection, the activities conducted by the Secretariat at IAEA headquarters may represent the area where significant progress can be achieved, since these activities do not pose the difficulties associated with obtaining Member States’ unanimous support for new legal arrangements or their acceptance of additional voluntary undertakings when being inspected. The IAEA, with Member State support, has developed and continues to elaborate new safeguards concepts, improved methodologies and advanced technologies at its headquarters. IAEA headquarters is also the venue for achieving significant progress in drawing safeguards conclusions, based on analysis and evaluation of the major ‘resource’ information. Progress in information collection, analysis, evaluation and dissemination may offer valuable opportunities to improve the effectiveness and efficiency of the IAEA safeguards system.

The reason is quite simple: information management has multiple functions in the core business of verification, particularly for verification activities often referred to as ‘information driven’ safeguards. The build up of an inspector’s knowledge to an appropriate level, through properly focused and disseminated information, is a key factor for effective inspections, while the definitions of field strategies and action plans require proper decision-making supported by all appropriate information. Given the restrictions associated with field activities, an inspection body cannot verify everything ‘on foot'. The legal limitations associated with the difficulty of obtaining access to ground areas beyond those identified in formal agreements, the fact that field activities are demanding in terms of resources and must be kept for the noblest aspect, and the inherent limitations associated with budgetary constraints and the always-too-limited availability of competent staff are factors that naturally promote the development of headquarters activities. Moreover, drawing 'credible' safeguards conclusions for a State, i.e. conclusions that cannot be contradicted in the foreseeable future, must rely on the widest spectrum of information and data, assessed in terms of their credibility and quality, before being put in the right context through analysis and evaluation. All-source information analysis is the only way that the IAEA can meet its 'customers' expectations'.

3. Sources of information

International verification starts with a State's declarations. The evolution of these declarations, since the beginning of nuclear verification and the associated problems and solutions implemented, is discussed in a paper presented at this Symposium [1]. State declared information is a critical and essential component of safeguards analysis. The amount of available information has increased greatly over the past 15 years, following in particular the implementation of the Model Additional Protocol (INFCIRC 540 (Corrected)). At the same time, challenges have arisen with regard to improving the quality of the data and reducing the complexity of handling the information. While the analysis of State declared information has progressed far beyond that performed in the 1970s, more progress is needed in that traditional area, starting with enhancement of States' systems of accounting for and control of nuclear materials (SSACs). Assessment of the correctness and completeness of States' declarations remains the overall challenge for those at the IAEA charged with drawing safeguards conclusions.

The results of in-field activities, such as inspector observations, and of technical monitoring activities, such as video monitoring as part of containment and surveillance (C/S) measures, provide a wealth of safeguards relevant information for the IAEA. The implementation of new types of access that differ from the traditional verification of declared information, such as complementary access, or the flow of information derived from the use of advanced technologies, such as environmental sampling or remote monitoring, provide remarkable opportunities to reinforce the effectiveness and efficiency of the IAEA safeguards system. However, they also generate new challenges with regard to the proper handling of the information produced.

As part of the measures to strengthen the IAEA safeguards system, new information collection methods have been developed, in particular from open sources available to anyone. However, open sources can be overwhelming (vast quantity of information, multiple languages, information origins from news media to scientific and technical literature), unreliable (open sources can be based on pure political agendas and not on factual reporting), and 'expensive', such as that from commercially available satellite imagery or from scientific libraries. In some instances, such as the case of Iraq, the IAEA may be able to benefit from third-party information, albeit with the additional challenge of source sensitivity.

Conducting all-source information analysis poses a major challenge in terms of the nature of this information. Historically, paper has been an unavoidable origin that unfortunately buries more information than it displays. What can be realistically expected when a key element of information that could contribute to resolving a critical question is actually buried among metres of archived reports stored on shelves? Little can be expected when the IAEA has limited resources for allowing new staff members to develop their own research before they become fully operational on a specific topic, as it is done in the academic world before the development of a PhD thesis.

Although progress has been made in terms of receiving already computerized information (e.g. State declarations, open source 'harvest', pre- and post-inspection notes, experts' reports), textual data can be as overwhelming as paper-based information, as a result of the historical development of very specific types of independent databases, the immaturity of IT developments and the need for increased security measures. Such problems are not unique to the IAEA and are being addressed by other organisations. The fact that disconnected pieces of information, recorded for time-specific and motive-specific reasons, should be used for overall all-source information analysis presents an interesting challenge. Moreover, the non-textual data generated from cameras, sensors, site layouts, design information, satellite images, ground photos, sample analysis graphics, etc., while providing invaluable additional information, also create an additional burden for the development of new solutions to link and access these resources.

Last, but not least, is the most volatile support for information: the human mind, which is often the repository of details that can make the difference between addressing an issue in a timely and cost-effective manner, or spending an unreasonable amount of resources and time to 'reinvent the wheel' or, even worse, overlooking a problem because a few individuals are aware of the information but are unaware of its significance. The 'age pyramid' of the nuclear community and the lack of effective knowledge management are greatly impacting the population of verification specialists, including inspectors, given the spectacular turnover that the IAEA Secretariat will face in the next few years. How many individuals who have recently retired, or will be retiring in the next few years, are experts with 'eye-opening' experience, such as the case of Iraq's weapons programme discovery in 1991, that of South Africa and its voluntary nuclear disarmament and ongoing IAEA cases such as the Democratic People's Republic of Korea (DPRK) and Iran? All too soon, the IAEA will not have available the many inspectors, and other critical staff members of the Secretariat whose experience has helped to make the IAEA's nuclear verification activities so effective.

The overall characteristic of safeguards relevant information is that, on the one hand, its quality is 'weak', lacking sufficient comprehensiveness to ensure that conclusions are based fully on facts and leave no room for unwanted, opinion-based conclusions. On the other hand, the quantity of information is overwhelming. Can the information systems that will be implemented in the next few years deal appropriately with all these challenges?

4. The way ahead

As it has done over the last decade following the conclusion of Programme '93+2' for strengthening safeguards and the adoption and implementation of the Model Additional Protocol, the IAEA will continue to ensure that it makes the most of its information resources. With regard to information collection, 'better', 'broader' and 'deeper' are the key words. Better information collection involves improving the quality control of declared information, with the provision to States of enhanced declaration computerized tools, training SSAC personnel to enhance quality assurance, assessing more reliably open sources information credibility, and adding new technical expertise and information tools such as information extraction to identify 'the signal within the noise'.

Broader information collection relies on the identification of possible new sources, for example through the development of access to Web-based information in less common languages, through assistance from recruited experts, through the implementation of machine translation tools, through the use of commercial satellite imagery that can be appropriately analysed with adequate internal skills (including for high resolution radar), and/or through the growing awareness of proliferation challenges among commercial companies so as to obtain information that was simply discarded in the past (e.g. commercial enquiries ignored as soon as company ethic and national export control would prevent further action).

Deeper information collection relies on the ongoing identification of current limitations, including a gap analysis of the required expertise in relevant technologies, and improved access to information not yet fully reachable via standard tools. For instance, only a small fraction of the information posted on the Internet is actually accessible through typical search engines.

As previously indicated, all sources of information must benefit from enhanced collection means and methodologies, including those for State-declared information, inspection results, open sources and satellite imagery, and information supplied by Member States and private organizations.

From an information management point of view, the key challenge is to render available, in an accelerated manner, all existing information, on a need-to-know basis, to comply with the need to enhance dissemination and to ensure that all-source information analysis provides a solid basis for credible conclusions while strictly respecting the IAEA's commitment to its Member States for confidentiality of information. The optimal solution relies on an integrated information architecture, from the digitization of the historical paper heritage to the consolidation of a complex infrastructure, both in terms of databases and hardware. Now-mature concepts of a business driven architecture [2] (i.e. a service oriented architecture) can provide tremendous opportunities for optimising the timely dissemination of information and knowledge to those who need to know while significantly decreasing the maintenance costs of state-of-the art systems. At the same time, enhancing security policies and technical solutions will allow the IAEA to maintain the trust that States have in its ability to respect confidentiality undertakings [3]. The two papers referenced above elaborate on these concepts within the context of the overall information environment.

Owing to Member State recognition of the need for these efforts and the provision of special regular budget funds and extra budgetary funds, the IAEA has an exceptional opportunity to move forward at a much needed accelerated pace. The ISIS (IAEA Safeguards Information System) Re-engineering Project (IRP) [4] will deliver, after approximately three and one half years, a fully integrated information system that would increase both the effectiveness and efficiency of the Department of Safeguards. All developments within the next five years will be highly coordinated to ensure successful results, including the implementation of ’best practices’ for project management and the application of infrastructure programming.

Drawing credible conclusions and developing effective action plans will be the end results of the deployment of an advanced integrated information system. However, to improve the overall analytical culture, the review of current processes [5], continuous process improvements, the contribution of new skills and expertise and the implementation of new tools will have to take place. Through the acceleration of the nVision project [6] and its integration within the IRP framework, the information collection, analysis and delivery will be drastically enhanced, in particular with the aim to free the analysts from routine data processing tasks and allow more focus on matters of substance. In parallel, nuclear security [7] and new proliferation issues, such as the necessity to better understand cross-border trade networks [8] identified after the discovery of Libya’s clandestine programme, should reinforce the IAEA’s overall ability to remain a trustworthy source for the assessment of nuclear proliferation issues.

On a longer term, the information challenge to be addressed is not only to ensure the generation of knowledge needed to deliver the expected products (e.g. timely conclusions) but also to guarantee the preservation of knowledge through the years to come. Proliferation issues often brew for years before making the news headlines. The IAEA’s ability to be proactive and act more as a prevention body rather than as an effective crisis fire-fighter will require enhancements of its efforts toward knowledge management. Ensuring that today’s knowledge of apparent details remains available tomorrow, consolidated with other details to come, the development of timely understanding may then be possible. And, as previously highlighted, increased efforts for proper knowledge management will help to cope with the tremendous turnover rate of staff of the IAEA Secretariat. In this context, information technology plays an important role, but overall management activities must also systematically include a component of knowledge management.

5. Conclusions

Information is at the heart of modern nuclear verification. Over the past 15 years, in particular since the discovery of Iraq’s clandestine programme that highlighted the need for focusing additionally on the detection of undeclared activities, the IAEA has moved significantly away from its traditional role of nuclear material accountancy ‘auditor’. The challenges posed by safeguards-relevant information and its collection, analysis, evaluation and dissemination have been complicated by the high profile of international security in today’s world. As a result, the IAEA has intensified its efforts to remain ‘ahead of the race’, through the implementation of processes, measures and tools commensurate with the expectations of the international community. Nevertheless, efforts will have to go beyond those of the IAEA Secretariat to ensure the existence of an efficient and effective IAEA safeguards system.

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Safeguards information analysis: Progress, challenges and solutions

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Abstract. While the IAEA's authority to verify the correctness and completeness of a State's declarations under its comprehensive safeguards agreement derives from the agreement itself, it is only with the provisions for broader access to information and locations available under an additional protocol that the IAEA is able to draw the safeguards conclusion regarding the absence of undeclared nuclear material and activities in the State. Under the State level concept, all relevant information about a State's nuclear activities is assessed to obtain as complete a picture as possible of the State's current and planned nuclear programme. The array of sources for information evaluation is both broad and diverse. Mainly, it encompasses information provided by States, information obtained from open sources, commercial satellite imagery, and inspectors' measurements. All the information is checked for internal consistency and consistency with information gathered during inspections and visits in the field. The 'information driven' approach has required an expansion of knowledge, expertise, information and analytical/evaluation skills.

As the IAEA's corporate knowledge in exploiting new types of information has increased, so too has its capability for detecting proliferation indicators. In all areas concerned (open source, satellite imagery, consistency and trend analysis, nuclear trade analysis, environmental sampling, as well as State-declared information) remarkable improvements have been made with regard to methodologies, tools, expertise and skills. Although each of these areas has proven invaluable for the detection of certain undeclared activities/material, it is obvious that the strength of these methods is in their integration. Further qualitative and quantitative improvements will require the acquisition of additional specialist expertise, knowledge and technologies. The combination of all (new) technologies will be important for enhancing the IAEA's ability to detect proliferation indicators, including the existence of proliferation networks. Inevitably, the latter will become an even more serious challenge in the future as nuclear technologies, and the capabilities to develop them, become more readily accessible in a more globalized economy supported by widespread and rapid industrial expansion.

1. Introduction

The IAEA carries out verification activities aimed at providing credible assurance that States are adhering to their safeguards obligations. This includes the provision of assurances that a State's declarations are correct and complete, i.e. that there was no diversion of declared nuclear material and that are no undeclared nuclear material or activities. To be able to judge whether a State's declarations are correct and complete, the IAEA has developed a variety of methodologies to analyse information obtained from various sources. The methodologies include trend analyses and consistency checks of accountancy data, nuclear flow assessments, comparison of States' declarations with other information obtained by the IAEA, analyses of information obtained from open sources, analyses of satellite imagery, nuclear trade analysis, and the evaluation of environmental sampling.

The IAEA is in the unique position of having a wide variety of information sources available for State evaluations: information declared by the State (nuclear material accounting reports, additional protocol declarations, import/export notifications, etc.); information obtained in the course of implementing safeguards (e.g. inspection activities); information from open sources and commercial satellite imagery; non-safeguards IAEA information; and a number of ancillary sources. In order to evaluate this extensive and diverse amount of information and thereby to enhance its detection capability, the

IAEA is improving its information collection, planning, processing, analysis and evaluation tools and methodologies. This paper provides an overview of current activities in the Department of Safeguards in the area of information analysis and evaluation, identifies potential improvements and proposes how improvements could be implemented.

2. Sources and methodologies

The primary focus of the analytical function within the Department of Safeguards is the State Evaluation Report (SER). State evaluations are conducted to determine the level of compliance with safeguards agreements. The evaluation considers all elements of a State’s nuclear fuel cycle and associated activities, including scientific, technical, economic and commercial activities, nuclear-related companies, and assessments of political and security issues. All key sources of information are compared with information declared by the State, to determine whether the State’s nuclear activities are consistent with the declared nuclear facilities and nuclear material holdings.

The results of the annual SERs are documented in the annual Safeguards Implementation Report (SIR), which is then issued to the IAEA Board of Governors. Given that the Department of Safeguards will conduct, on average, more than 100 State evaluations annually, an efficient system for collecting, processing and analyzing data is required to ensure that safeguards conclusions are based on sound information.

The SER is a product of the State Evaluation Working Group, which includes contributions from both the Safeguards Divisions of Operations and the Support Divisions. The purpose of the group is to bring together a wide range of analytical skills and methodologies for a detailed assessment of a State’s nuclear activities. The appropriate Safeguards Division of Operations is responsible for coordinating contributions from its division, from the support divisions and from the Office of External Relations and Policy Coordination and the Office of Legal Affairs. Specialized units of the safeguards support divisions are responsible for the collection, processing, analysis and transmittal of information to the State Evaluation Working Group. These units comprise the information management unit (e.g. responsible for State declared and supplied information pursuant to safeguards agreements and additional protocols, nuclear material accounting information); the information analysis unit (e.g. responsible for open source analysis, illicit trafficking information and relevant information in travel reports); the satellite imagery analysis unit (e.g. responsible for information obtained from commercial satellite imagery); the nuclear trade analysis unit; the material balance evaluation unit; and the unit dealing with environmental sampling.

While the SER process and the resulting conclusions reported in the SIR are key ‘products’, there are a number of additional analytical studies undertaken and reported on annually within the Department. Many of these are conducted to assist the planning of verification activities in the field. A trigger for these studies is often related to proliferation issues concerning undeclared nuclear activities and facilities. New information sources have been increasingly developed over the last decade to increase the IAEA’s ability to detect undeclared nuclear materials or facilities. Information from the industrial base, open sources and satellite imagery often identify locations outside declared nuclear facilities that may warrant that a State provide further clarification or that the IAEA conduct complementary access in the State. Environmental sampling may also be applied to detect the presence of undeclared nuclear material within facilities, which can also contribute to the detection of undeclared facilities.

2.1. Information provided by the State

One of the most, if not the most, important source of information is the State. Under the provision of a comprehensive safeguards agreement, the State system of accounting for and control of nuclear material (SSAC) provides the IAEA with the accountancy reports for its respective facilities and locations outside facilities (LOFs). These reports include Inventory Change Reports, Material Balance Reports and Physical Inventory Listings. In addition, the SSAC provides design information about facilities and LOFs, as well as other relevant information, such as information about neptunium and americium inventories, nuclear loss and nuclear production.

The State may also report (where applicable) under the Voluntary Reporting Scheme on, inter alia, the export, import, production and inventory of nuclear material and the export and import of specified equipment and non-nuclear material. This information is provided voluntarily in addition to the relevant information required under the existing safeguards agreements and under the provision of INFCIRC/549 on plutonium and high enriched uranium inventories; the latter information is provided only by five of the nuclear weapons States and by four of the non-nuclear weapons States. All of this information is checked for internal consistencies, and also for consistency with other sources such as inspection reports.

The information is also used for analysis and evaluation of nuclear material processes, inventories and flows. These include the material balance evaluation (MBE) of bulk handling facilities with an inventory of more than one significant quantity (SQ) and trend analyses of those components in the material balance equation that could be used to mask a diversion of nuclear material in a particular material balance area (MBA). Furthermore, trend analyses and flow assessments are being performed for sites and for the State as a whole. In specific cases, very detailed flow assessments combined with process assessments are carried out for processes such as conversion and reprocessing. Through the above analyses/evaluations and evaluation of inspectors’ measurement data, the IAEA can determine whether the quantities and processes (activities) of nuclear material have been correctly declared. By extending the evaluation process, the IAEA is also able to determine the completeness of the declarations provided by the State. For a State with an additional protocol in force, the SSAC will provide periodical declarations on nuclear related activities to the IAEA, as defined in Articles 2 and 3 of the Model Additional Protocol (documented in INFCIRC/540 (Corrected)).

The information contained in these declarations should give a complete picture of a State’s nuclear activities, including information on nuclear material activities, nuclear related activities not involving nuclear material, source material transfers and holdings, site descriptions, equipment transfers and future plans for development of the nuclear fuel cycle. This information is checked for internal consistency, as well as for consistency with other information obtained from the State and from other sources. The latter include publicly available databases, commercially available satellite images and measurement data collected during inspections.

Other information provided by or through State authorities (which could include bodies other than the SSAC) includes information on imports and exports of nuclear and dual use material and equipment and information on trade activities related to nuclear goods and services. The latter could, in co-operation with the relevant State authorities, also have been obtained from (private) companies. Again, the outcome of the analyses of this information is compared and cross-checked with the outcome of the analyses of information from other sources.

2.2. Information obtained from publicly available databases

The IAEA basically reviews and analyses two kinds of open source information: information in the form of documents in the broadest sense, and information from satellite images.

2.2.1. Open source information in the form of (text) documents

The collection of open source information is a very different task compared to that of processing State-supplied information or information from inspection activities. Open source information tends to be text data that is amorphous and of varying quality, reliability, and credibility. It may include information on the economic and political status of a State, its imports and exports, and its nuclear research programme, as well as information on specific facilities.

The IAEA’s open source acquisition and analysis effort began in the mid-1990s. Since then it has expanded and become a core component of the information analysis process. Open source information is compiled for the SER, ranging from background materials on the State’s economic and political status to detailed fuel cycle information. It is the latter information that poses the greatest challenge. While general information on a State is usually readily available, more specific information on facilities, research and nuclear materials requires a more specialized effort and the focused collection and evaluation of materials. Especially important is the acquisition and analysis of scientific and technical information, which may come from multiple sources and in many languages. This information can provide important insights into the R&D capabilities of States and their industrial infrastructure, and is particularly useful in evaluating additional protocol declarations.

The IAEA obtains the relevant information, inter alia, from databases that contain non-restricted information and that have been developed by institutes specialized in the field of nuclear non-proliferation. In addition, the safeguards analysts extensively search the web for other relevant data, including Government reports and reports from non-governmental organizations and academia. The data selected can be in the form of newspaper articles, scientific papers, and economic, social, and political overviews, etc., and should provide relevant information on past, on-going and new nuclear proliferation and safeguards issues (including illicit trafficking and trade in nuclear goods and services).

The analysis results are used to check the additional protocol declarations and other information provided by the State and that obtained from other sources. All the results are compiled in a State file which forms the basis of the input to the SERs and other (special) reports. Furthermore, the analyses results are used to obtain an overview of the nuclear activities and capabilities of a State. They could, among others, provide information about experiments, institutes and scientists involved and connected with other institutions within or outside the State. Together with the evaluation results of other sources and methodologies, these analyses provide valuable information about a State’s nuclear programme.

2.2.2. Commercially available satellite imagery

Currently the IAEA has contracts with several commercial suppliers of imagery data. This guarantees access to the sensors needed for safeguards purposes; it includes optical, thermal, and radar data. The contracts also provide an opportunity to recover imagery data from archives, where available and when needed. The imagery data are usually provided in a timely fashion.

Normally, the satellite images are ordered on request from, and in close consultation with, the respective Safeguards Division of Operations. The imagery is analysed, inter alia, to check the additional protocol declarations (e.g. those for large sites) to detect changes in sites, buildings and processes, and to obtain information on possible undeclared activities and facilities. These could range from introducing new processes, or increasing capacities, to preparations for nuclear tests. The evaluation results are cross-checked with other available relevant information.

2.3. Information obtained from inspector verification activities

The most effective way to obtain an overview and insight of the nature and extend of a State’s nuclear programme is to conduct verification activities at the sites involved in the programme. These activities may include inspections, design information verification (DIV) visits and complementary access, and are normally carried out by safeguards inspectors. When deemed necessary, the inspectors may be accompanied by analysts or experts in specific areas.

The aim of these on-site activities is basically to confirm that the activities taking place and the related throughputs are as declared by the relevant State. Important tools to arrive at soundly based confirmations are performing measurements and collecting samples. The measurement and analytical results are evaluated by safeguards evaluators. The evaluation results provide information of a quantitative and a qualitative nature. The quantitative results give a statement about the amounts of nuclear material present in a certain location or stratum. They also determine the physical properties and chemical composition of the material involved. The results are used for material balance evaluation (MBE) purposes and for the assessment of flows between facilities and sites within and between States, i.e. they confirm (or not) the declared inventories and flow of nuclear material.

The qualitative results (e.g. from environmental samples or other samples taken for other purposes than MBE or flow assessment) provide a statement about possible undeclared activities. For instance, environmental sampling results could determine whether undeclared enrichment or reprocessing activities are being performed or have taken place, and results (including information about impurities) from waste samples could indicate from which processes the waste in question is stemming from. Again, all the outcomes are compared with all other relevant information.

3. Challenges and solutions

The challenges the Department of Safeguards is facing with respect to enhancing efficiency and effectiveness can be traced mainly to the following factors: the large amount of information to be analysed/evaluated, the structural location of the various evaluation units in the Department, and the availability of resources. One way of tackling the problem is to divide the areas in which the enhancement of the efficiency and effectiveness could be achieved into three parts: the evaluation process, the input to the process, and the output to the process.

3.1. The evaluation process

The information analysis/evaluation is performed in six organizational units, each of which obtains the necessary data from various sources. The units are currently as follows: the accountancy unit dealing with State reports, additional protocol declarations, etc.; the Section for Open Source Analysis and Satellite Imagery within the Division of Information Technology; the Nuclear Trade Analysis Unit attached to the Safeguards Deputy Director General’s Office; and the unit performing MBEs and the unit for evaluation of analytical results of environmental samples — the latter two units are currently located in the Section for Statistical Analysis in the Safeguards Division of Concepts and Planning.

Often there is synergy through cross-unit collaboration. And there are countless ways in which such collaboration could occur. For example, nuclear material accounting information can provide knowledge of very small amounts of uranium hexafluoride holdings, whether natural, depleted or enriched. These amounts may seem innocuous in terms of the significant quantity of enriched uranium required for nuclear weapons.

However, the analysis of proliferation often requires investigating less obvious and subtle indicators. Small amounts of uranium hexafluoride, even in gram amounts, in a State without a declared uranium enrichment capacity could raise a question as to why the location has such material. What is manufactured in the State or imported? If manufactured domestically, then why? Was it for conversion experiments? Checks of nuclear material accounting reports would also be performed to confirm if the material had been imported, when and from where. Were any enrichment activities (or conversion activities) included in the additional protocol declarations? Questions such as these could trigger extensive searches in open sources for enrichment research or nuclear cooperation with another State that has mastered enrichment technology. If the material was imported, open sources would be searched to determine the past and present nature of any nuclear cooperation, including the technical capabilities of the exporter.

Deeper investigations may be undertaken if there are any signs of research on developing dual-use technology that could be applied to enrichment research. Similarly, a search may be undertaken for commercial imports of equipment or materials with a direct or indirect usage for enrichment. Satellite imagery, including archive material, of the location could be obtained to identify if any features of particular importance may exist, such as uranium hexafluoride containers.

Environmental samples of the location and environs could be used to determine if there is any evidence that might suggest that there had been nuclear material with enrichment levels different from the levels declared in accounting reports. These types of information and the issues raised could be compared with inspectors’ observations and activities in the field. Anomalies at the facility would be taken into account, and the historical material unaccounted for (MUF) may warrant further detailed analysis, even if there are no statistically significant deviations.

From the above illustrative example, there are clear benefits to close collaborative analysis, especially on high priority issues. However, the current organizational spread is not conducive to an optimal communication between the evaluators. Therefore, the situation would be improved if all the evaluation units and sections were brought together into one organizational entity.

Such a new organization would improve communication between evaluators and allow the production of a comprehensive consolidated evaluation report, which then could be given to the respective Operations Divisions for inclusion in the SERs. This could be done by the appointment of individuals who would serve as a unique contact point for the inspectors, thereby streamlining communication between evaluators and inspectors. Furthermore, the evaluation process could be considerably improved if, where applicable, the software tools are harmonized and, where needed, more specialized and sensitive analytical tools are purchased. Examples of the latter are tools to allow searching in the "deep web" and tools to deal with multilingual information collection. Several of these improvements are planned to be implemented in the context of the ISIS Re-engineering Project (IRP), whereas others could be implemented presently.

Obviously, the quality of the evaluation process depends not only on the tools, but also on the availability of skilled evaluators. Therefore, it is essential that there is a sufficient number of highly qualified staff members available to deal with the large amount and variety of information associated with the approximately 140 State evaluations conducted annually. In this regard it should also be stressed that the handling of this data and information requires a smooth functioning process, and a well-functioning knowledge management system — for example, State files have to be created in such a way that all information used during, and resulting from, the evaluation process should be readily available for all parties involved now and in the future.

3.2. The input

As mentioned earlier, the information originates from the State, publicly available databases, commercial companies and from inspectors. On occasion, questions have been posed by States about what information should be provided to the IAEA and how they should send this information (e.g. in which format and how frequent). To clarify these and other issues, the IAEA organizes regional seminars and also holds bilateral meetings and consultations, when necessary. In several cases, there has been the need to improve, through the State's authorities, the channels of communication with private companies. The latter is particularly important for issues related to nuclear trade analysis.

Moreover, the quality of the analysis of satellite imagery could be improved if States would provide on a regular basis additional information, such as geo-spatial datasets and geographic information. This could be implemented by giving the IAEA access to relevant databases or parts thereof.

In general the input for the evaluation of open source information and information from inspectors is smoothly provided. It requires, however, that updated and extended tools are made available to the IAEA.

3.3. The output

Currently, each of the various evaluation units/sections prepares an output report that serves as input for the SER. If and when the above-mentioned suggestions were to be implemented, all of the evaluators would produce one consolidated report. This would enhance the efficiency and the quality of the output. Moreover, the establishment of central contact points (for each Safeguards Operations Division, or for each section or for each State) would improve the communication between inspectors and evaluators.

4. Conclusion

The importance of the joint contribution of the evaluators to the SER, and thereby to the process of drawing conclusions on the nature of the nuclear programmes of States, is well recognized.

Over the years, the inspector’s profile has changed as a result of changes in the nature of the inspector’s work. The evaluator’s work is also subject to continuous change. Therefore, there is a need to train evaluators and to procure up-to-date tools.

Although the IAEA’s achievements in the area of information analysis are remarkable, it is obvious that there is room for improvements. Some of the improvements are of an organizational nature, and probably would not require extensive investment. Others, such as the procurement of state-of-the-art tools and equipment and the employment of highly skilled personnel, would depend on the willingness of the IAEA to invest considerable financial resources. In sum, the success of the IAEA in this area depends on its commitment, and that of its Member States, to further development and improvement.

Open source research and nuclear safeguards

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Abstract. The paper assesses the use of open source research for enhancing the collection and analysis of information related to nuclear Safeguards. It examines the following sets of issues: the various ‘types’ of open sources applicable to enhancing the analysis of nuclear Safeguards issues; the ‘utility’ of open source research; the ‘challenges’ involved in conducting open source research on Safeguards issues; and the potential ‘problems’ associated with researching open sources related to Safeguards. The paper concludes that open source research has a valuable role to play in enhancing the collection and analysis of information related to Safeguards. Its utility stems primarily from its ability to provide essential contextual awareness for better understanding data collected during site visits and information provided as part of state declarations. Moreover, open source research also has a proven ‘tip off’ capability in the Safeguards field and can be invaluable when responding to situations that have gone predicted.

1. Introduction

Research of open sources undoubtedly has a significant role to play in terms of enhancing Safeguards. The utility of open source research stems primarily from its ability to generate contextual awareness, both as part of the general verification process associated with Safeguards, but also in terms of responding to events and developments that may not have been predicted or are totally unexpected. Open source research can also, on occasion, provide information and/or generate insights capable of prompting specific Safeguards-related investigations. Indeed, expanding both the sources and coverage of open source information offers a cost effective means to improve the collection and analysis of Safeguards-related information.

This paper begins by defining ‘open source’ and examines the various types that may be applicable to the analysis of issues related to Safeguards. The paper then examines the ‘utility’ of open source research by focusing on those areas in which it can enhance the analysis of Safeguards-related issues. This is followed by a consideration of the problems and challenges associated with conducting open source research in this field.

2. Defining open sources

Open sources can be defined as publicly available information that can come in print (hard-copy), electronic and verbal formats. Open sources do not contain classified information of any type and their distribution can be relatively widespread or very constained in nature. [1]

At the most basic level, open sources consist of literature that is published in print and/or electronic formats including periodicals (e.g. newspapers, magazines and journals) and reports released by governments, international organisations and other entities such as companies, societies, universities and so on. These types of sources obviously come in a wide variety of languages and cover a wide array of topics both technical and political in focus. Moreover, the revolution in information connectivity in recent years has resulted in an explosion in the availability of open sources via on-line databases (both subscription- and non-subscription-based), specialist websites and through Internet searching.

Some open sources are referred to as 'grey literature' because of the greater difficulties associated with identifiying and obtaining them, in some instances because they are not part of the standard publishing process. While grey literature is still publicly available information, its existence may not be widely known about and its distribution is by definition more limited in nature; as a direct result such sources can be expensive to acquire. Examples could include papers and poster presentations given at symposia and conferences like this one at the IAEA in Vienna, materials on display at exhibitions and workshops, in-house company reports, and the like. Such literature can contain a surprising depth of informaton on technical issues such as nuclear technology and capabilities in any given context.

Open source information can also be derived from direct interaction with subject matter experts (SME). Examples of SMEs could be officials representing, or consultants working for, governments, international organisations and companies. Journalists, particularly investigative journalists and those with specific thematic or regional expertise, can also be valuable sources of information and analysis. For example, it has been estimated that journalists with specific country expertise publish less than 10% of what they actually know.[2]

Finally, open source information can be derived in a much more technical fashion using satellite imagery which can be acquired, usually for a not insignificant cost, from various commercial providers such as Digital Globe.[3]

3. Utility of open sources

The identification, collection and analysis of pertinent open source information can provide a cost effective means through which to supplement Safeguards-related information derived by inspectors from site visits, and to assist in verifying the authenticity of information declared to the IAEA by states subject to Safguards Agreements including those that have concluded Additional Protocols. Indeed, the expanded mandate given to the IAEA vis-à-vis states that have concluded an Additional Protocol has increased the Agency's requirement for accurate and timely information on a much wider array of activities than purely nuclear materials and their associated facilities. A pertinent example would be a company that, while not actually working with nuclear materials, is involved in the manufacture of the sub-components required for gas centrifuges used in the uranium enrichment process.

Open source research and analysis are probably most useful in terms of providing Safeguards analysts with contextual awareness in relation to specific countries, particularly in terms of assessing information gathered during inspections or when provided as part of state declarations.

At one level, this awareness can involve political-strategic and economic issues which provide the broader context within which to understand national nuclear policies, activities, ambitions and decision-making, both in the civil field and potentially with regard to military intentions in the nuclear area. Indeed, it has been noted in the military sphere that open sources can provide a fast and inexpensive means of orientation for planning purposes and this can involve assessments of national intentions, cultural attitudes and strategic generalisations.[11]

Open sources can be particularly useful in terms of understanding national threat perceptions and strategic relationships between countries, both of which could potentially provide important indicators of intent in the nuclear area. Security-, defence-, intelligence- and regionally-focused periodicals, both practitioner and academic in focus, would constitute such sources. Solid examples would be, among many others, the Jane’s Information Group, the Nuclear Threat Initiative, the International Institute for Strategic Studies and the Al-Ahram Centre for Strategic Studies. Moreover, resource and other economic indicators from sources such as the Economist Intelligence Unit, and the Energy Information Administration of the US Department of Energy, can provide important context for evaluating whether or not a given country actually needs to develop nuclear energy for civil power purposes.[4]

Beyond the *de jure* and *de facto* Nuclear Weapon States open sources will rarely contain specific information on national intentions regarding national acquisition of nuclear weapons. The lack of pertinent information in this respect stems primarily from the extreme political sensitivity associated with the issue and the desire of most governments in the Non-Nuclear Weapon States to ensure that it is not publicly aired in any great detail, even if there is an underlying current of domestic opinion that may favour nuclear weapon acquisition.

Beyond the political-strategic and economic spheres, open source research can provide relatively accurate and important context vis-à-vis the various elements of the nuclear fuel cycle in any given country as well as peripheral support areas such as the speciality metals and chemical industries and associated research centres. It is possible using open sources to develop a fairly detailed picture of the various elements of the fuel cycle, and peripheral support areas, currently in existence or being developed in most countries. Official sources can be a particularly useful reference point in this respect including publications and websites associated with national nuclear organisations and government ministries with responsibility for issues such as energy, mineral exploration, and science and technology.[5]

While a considerable amount of information is often available from national atomic agencies, it is seldom clear at first glance how comprehensive this information actually is. Indeed, the information provided by national nuclear organisations through reports, articles and websites can be used to develop a baseline assessment upon which to generate a more complete picture by adding information derived from other open sources; and against which to assess openness and potential undeclared activities. In this respect technical and scientific sources can often provide very specific information of high relevance to nuclear Safeguards.

Relevant technical and scientific information can be accessed via a range of different sources including international scientific databases [6], national educational and scientific databases[7], university websites[8], company publications and websites[9], and material and websites associated with professional societies.[10] In some instances, such sources may not be readily accessible and will require a subscription or membership of the relevant institution (grey literature).

On occasions when information is required urgently to inform a situation that has not been expected or predicted, open sources may offer the only solution to meet such a requirement at least in the short term.[12] The suddenness of Libya's announcement on its nuclear programme in December 2003 is probably a good example of such a situation because it took by surprise all parties outside of a small group of Libyan, British and American officials. Moreover, open source research has a 'proven value' in terms of providing a 'tip-off' capability for developments in the technical, military or political fields.[13] In the Safeguards area, open sources already have a proven track record in this respect in at least one instance and possibly others.[14]

Open sources will rarely, of course, provide the complete picture. This is particularly the case with regard to the most sensitive aspects of the fuel cycle potentially associated with nuclear weapons development -- uranium enrichment, plutonium separation and weaponisation -- which are subject to the greatest secrecy whether the country is a closed or open society.

4. Challenges and problems associated with open source research

The process of identifying, collecting and analysing open source information pertinent to nuclear Safeguards is not free of challenges. Notable challenges include: avoiding the pitfall of information overload; overcoming the difficulties posed by both open and closed societies; effectively utisiling non-English language sources; and harnessing multidisciplinary expertise in the research process.

Information overload is primarily a result of the rapid and immense growth of the Internet and web-based open sources over the past decade; it is perhaps the most difficult challenge to overcome particularly if time is the most valuable resource for the Safeguards analyst. So much information is now available on the Internet that even the 'surface web' generates vast quantities of sources. For example, a basic search on the Google search engine for 'uranium enrichment' produces some 3.7 million returns. In such circumstances, the signal-to-noise-ratio is so high that it is almost impossible to distinguish between irrelevant and useful sources. Obviously, more precise and targeted searches reduce the number of returns significantly, but a certain amount of information overload is inevitable whenever the Internet is involved. One sure method of avoiding information overload is to develop lists of proven and reliable source materials upon which to rely, at least initially, in any Safeguards related research project. This approach will help to conserve resources and avoid swamping the end-user with information that may not be wholly pertinent.

When searching for open source information on specific countries the transparency of the society involved -- whether this can be best described as open or closed -- also poses certain challenges. In open societies, for example, particualrly those that are technically advanced like Germany or Canada, such a great deal of information is likely to be readily available that it will be difficult to identify and concentrate on the most important and informative sources. A system of filtering information in order to search only for certain institutions or technical capabilities can mitigate this challenge somewhat.

In closed and non-transparent societies, such as North Korea, the challenges of conducting open source research will obviously be much greater. Notably, there will be a lack of key open source documents from the country itself with a significant level of detail on the issues under consideration. Those sources that are available via the national print and broadcast media will be subject to state control and propaganda.[15] Moreover, there may not be an official website associated with a particular country's national nuclear programme or, if there is, it may offer a very limited amount of information. Such circumstances will necessitate searching as widely as possible across all pertinent topics – the fuel cycle and peripheral support areas – in order to piece together various and multiple open sources with the aim of creating as complete a picture as possible with the data available.

A challenge that may not be immediately apparent involves the considerable amount of Safeguards relevant information that is available only in non-English language sources. Websites from non-English language speaking countries may often have part, or a summary, of their site available in English, but more information is generally available in the original language version. Two difficulties arise in this context. The first involves the actual identification of the information if the focus is purely on searching English language sources. The second difficulty involves translating such information once it is uncovered. Although several machine translation and search systems do exist, they do not yet appear to be sufficiently accurate to cope with such sources, particularly because the language and is likely to be of a very technical type requiring specific scientific vocabulary. If these sources are to be fully accessed, therefore, then professional translators may need to be used and this can be an expensive option.

Beyond these challenges it is also important when using open sources to take into account the potential that they will contain inaccurate and irrelevant information, biased perspectives and even disinformation. For example, bias in open sources can obviously be a product of cultural, personal or political outlooks.[2] Disinformation can, of course, be fed into open sources by governments and non-state actors. There is an onus on Safeguards analysts, therefore, to rigorously assess all open sources for their accuracy and lack of basis and to do so systematically and on an on-going basis. This itself necessitates the inclusion of experts in the open source loop; from the identification and collection of sources through to their analysis and potential presentation in the form of reports, assessments and so on. As Safeguards related research is a truly multidisciplinary process it is important to have experts that possess the requisite array of technical, regional, political and language skills.

5. Conclusion

Open source research has a valuable role to play in enhancing the collection and analysis of information related to nuclear Safeguards. Its utility stems primarily from its ability to provide essential contextual awareness for better understanding data collected during site visits and information provided as part of state declarations. Moreover, open source research also has a proven 'tip off' capability in the Safeguards field and can be invaluable when responding to situations that have not been predicted.

Open source research is not, however, without its problems and challenges. The challenges relate primarily to the need to avoid information overload, to overcome the difficulties posed by attempting to generate accurate information on closed societies, and to effectively utilise and exploit non-English language sources.

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[6] For example: the ISI WEB OF SCIENCE database contains references for approximately 8,700 journals, http://scientific.thomson.com/products/wos/; The ELSEVIER SCIENCE DIRECT database claims to contain over 25% of the world's science, technology and medical fulltext and bibliographic information, http://www.sciencedirect.com/

[7] For example: the Australian government's 'Bright Sparcs' database (http://www.asap.unimelb.edu.au/bsparcs/bsparcshome.htm) is a register of people involved in the development of science, technology, engineering and medicine in Australia, including references to their archival materials and bibliographic resources.

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Advanced information analysis technologies for safeguards

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Abstract. The implementation of information-driven safeguards requires the deployment of an array of advanced information technologies to support the collection, analysis and evaluation of information about States' nuclear activities.These technologies are essential for managing the extremely large (and growing) volume of information utilized for evaluation purposes. The large volume has arisen partly because in-depth evaluation reports have to be prepared annually for (currently) about 100 States; this number will continue to rise. Also, the IAEA now has greater access to relevant safeguards information. To convert even a fraction of this information into knowledge is a time-intensive, manual process for analysts. The limited amount of human resources available within the IAEA for State evaluations significantly constrains the amount of information that can be effectively scrutinized and restricts the extent to which the knowledge 'signal' can be extracted from the vast information 'noise'. As a result, useful interrelationships and patterns that may exist within large volumes of information may be not identified within the short deadlines to which analysts typically work.

In this regard, the Department of Safeguards has acquired a number of advanced tools. This paper explains the methodologies and plans for their use and the experience accrued. It describes their effectiveness for evaluating the large volume of information derived, for example, from the Internet and specialist databases including those available in-house. It also looks at how unstructured and loosely structured information can be automatically structured, semantically mined and visualized and how large volumes of non-English language text can be rapidly translated with good quality. Finally, it argues that advanced information tools are critical for information analysis and the IAEA safeguards mission. Further, it points out that that the key specialist skills for exploiting many of these tools are currently unavailable at the IAEA and must be imported as a priority.

1. Introduction

One of the most crucial lessons learned from the discovery of Iraq's clandestine nuclear programme is the importance of all-source information analysis. This has given rise to information-driven safeguards as a guiding concept for evaluating a State's compliance with its safeguards obligations. In this context, the IAEA aims to evaluate all available information in order to draw annually soundly-based conclusions concerning the non-diversion of declared nuclear material and the absence of undeclared nuclear material and activities in States.

The design of an approach for the evaluation of all available information poses a tremendous challenge for the IAEA. It is particularly challenging for the evaluation of the vast repositories of open source information available to the IAEA, especially through the Internet. Successfully tapping this information can give analysts valuable perspectives that would have been difficult to conceive a decade ago.

The current information technologies used by the Department of Safeguards, based largely on keyword searches, have helped to manage this deluge of information. They have proven to be valuable in finding information when it is very specific and the question is well defined. Yet the keyword search approach, by itself, can have severe limitations and require considerable analytical time. With the increasing availability of massive volumes of electronic information, even the keyword search produces grossly excessive numbers of 'hits'. The excessive burden on the analyst will continue to worsen as the volume of accessible information continues to rise sharply, at least for the foreseeable future. Furthermore, there may be information of value to an analyst but the specific question has not been asked – in most cases because the relevant question is not known.

Fortunately, new information technologies are emerging which offer highly advanced methods for assisting analysts in the evaluation of massive volumes of open source information. Deployed properly, these tools will help analysts to retrieve more information of value which they could not have achieved before. The availability of these tools will open up more possibilities for analytical investigation and will save time, enabling analysts to retrieve high quality information for analysis.

2. Project nVISION

Project nVISION was established in January 2005, within the Department of Safeguards, Division of Safeguards Information Technology, Information Support Unit (SGIT/IIS). Its creation followed a successful workshop held in October 2004, in New York, at the request of the IAEA to review advanced information tools. The workshop was organized by the United States Support Program to IAEA Safeguards (USSP).

Project nVISION will be a key component supporting a larger IAEA project — the IAEA Safeguards Information System Re-engineering Project (IRP) — that seeks to create the information environment for all-source analysis in the Department of Safeguards, particularly the fusion of all key data sources. In addition to open source information, the IRP will cover information provided by Member States and that collected from IAEA verification activities.

2.1. Strategy and objectives

The strategic goal of project nVISION is to determine whether and how state-of-the-art information tools could be used for the analysis of large volumes of information, particularly textual documentation. A guiding criterion is that the technologies should support both infrequent and frequent users of information analysis tools. Many information analysts in the Department of Safeguards that are infrequent users of information search and retrieval systems do not have the time to become intensive users of information tools. They need tools that are straightforward and efficient for retrieving relevant information. Most information collection tasks should be satisfied in this straightforward manner. Frequent users, however, will be able to use more highly advanced tools, such as clustering, visualization and links analysis, to capture deeply buried information or to identify patterns and interrelationships that are difficult to discern.

Deployed properly and with appropriate training for users, the complete range of project nVISION tools will help analysts to retrieve more information of value than they could achieve before. These tools will open up more possibilities for analytical investigation. Also, they will save time, enabling analysts to retrieve higher quality information for analysis.

This strategic goal of project nVISION has been translated into eight objectives:

(a) Current events and awareness raising;
(b) General information searching;
(c) Automated expansion of access to information;
(d) Automated division of large volumes of information into clusters of interest;
(e) Automated extraction of entities and semantics from text;
(f) Automated links analysis;
(g) Visualization; and
(h) Information management.

These objectives will be discussed in detail in Section 3.

2.2. Expertise, skills and support

Project nVISION comprises a small team of information specialists and analysts. Project membership is informal and voluntary. Furthermore, many of the critical skills and expertise needed to achieve the project objectives are currently not available in the Department of Safeguards. Therefore, the team's first goal has been to climb the learning curve and to assess various tools. A two-year horizon has been set for achieving this goal.

Since the start of the project in 2005, the team has gained considerable experience with commercial versions of advanced technologies. Project nVISION also established a Statement of Work with the Joint Research Centre at ISPRA, for assistance and expertise. An International Experts Group (IEG) was also set up to support the work of the project. The IEG comprises prominent information analysts, including experts proficient in the use of advanced information tools. The first IEG workshop, held in 2005, was an important validation of the directions being taken by project nVISION.

Based on this experience, a system approach is being designed which will substantially improve the capabilities of analysts to exploit open source information, according to the objectives of project nVISION. The critical challenge is to embed analytical intelligence into the tools and system. This requires analysts to transfer their knowledge, experience and analytical thinking processes to the system. The emphasis of most of the work so far and for the foreseeable future is set in an analytical paradigm, rather than software code writing for which there is little need since most of the technologies are commercial products.

For the success of the project it is essential that analysts are able to define their processes. Since analysis is more an art than a science, analysts should be able to define the choice of tools, configuration and outputs. Past experiences where analysts have largely played a passive role in key decisions and choices have resulted in tools that were not well utilized.

2.3. Enhanced open source information analysis architecture

For project nVISION, a first step is to design an enhanced information analysis architecture that takes into account the varied needs of different users (e.g. inspectors, proliferation analysts and managers) who have roles to play in the evaluation processes. This will require the deployment of user-friendly tools that can and will be used and that are readily maintainable. However, specialist services will be needed for applying essential tools that are not user-friendly.

The general architecture will be structured around the following main activities:

(a) Information collection and monitoring;
(b) Information analysis; and
(c) Information dissemination.

Information dissemination is not addressed in this paper, since this is part of the IRP's broader mandate for defining the architecture for the sharing of all information within the Department of Safeguards.

2.4. Information landscape

The IAEA faces unique analytical challenges in supporting its safeguards mission. Each year, it must determine whether States are complying with their safeguards agreements. The 2006 schedule includes the analysis and evaluation of safeguards information for over 100 States to determine whether they are complying with their safeguards agreements. This number will continue to rise. This represents an enormous challenge for information collection, analysis and dissemination.

Given the need for evaluation of such a large number of States, the open source system requires considerable flexibility. Most analysts, especially inspectors who spend long periods in the field, do not have the time to become intensive users of information tools. They need tools that, while advanced, are straightforward to use and efficient for capturing key information without overload. Analysts who travel much less will be able to use more highly advanced tools to capture deeply buried information or to identify patterns and inter-relationships that are difficult to discern.

2.5. General requirements

Any solution for synergising the analytical performance of a group of analysts, through the use of information tools, must take into account the variation in skills, experience and available time. It is important to build an information system that can be used by all. This is especially important at the IAEA, where many users engaged in the analysis of information for safeguards State evaluations have limited time for learning how to use advanced information tools. They need quick answers with little effort.

The information system must also cater to the medium- to heavy-end users and those who need to use more advanced technologies. It must enable those analysts who are infrequent or medium-end users to step up a level through creating the incentive and confidence to learn tools and search strategies at the next highest level.

Key requirements for the information system are summarized below:

— The system should have a mechanism that helps analysts to run their search queries against the most probable information sources that can satisfy the search; if the result is not obtained, the system should offer the next most probable group of sources and so on.

— The system should support an environment of information exploration and knowledge discovery – i.e. analysts should have confidence that the tools either directly satisfy the search query or generate new lines of enquiry which are worthwhile to investigate.

— The results should be easy to understand and the selection of tools and search strategies for any follow-on questions should be readily identified; analysts at any level should not feel ‘lost’ or deterred when searching because of unsatisfactory results or because of uncertainty about tools and search strategies.

— Users at each level (infrequent, medium and heavy) should become experts at using the tools; this will require the availability of support for analysts in the provision of sources and training in tools and search strategies.

— The overall system should be designed such that information sources that can satisfy a query are likely to be ranked high in the results list; this will help to ensure that the analyst spends minimal time reading clutter.

— The tools should be capable of navigating and extracting useful results from high ‘noise’ sources, especially the Internet.

— The system should be designed such that the user can readily access SGIT/IIS experience and knowledge gained from successful information searches.

— If insurmountable difficulties are encountered in finding the necessary information, the system should enable the user to task SGIT/IIS analysts who have substantial experience in complex searches.

— The system should allow for communication of open source information and results to be shared among users.

3. Project nVISION portal

The tools will be brought together and accessed from a centralized location, delivered as a web-based portal or 'dashboard'. As the portal will comprise many levels of complexity, there may be a need to define roles that would allow for matching technologies and information sources to particular needs.

Selected features of the project portal are explained below relative to the project's eight objectives mentioned in section 2.1.

3.1. Objective A: Current events and awareness raising

An important feature of knowledge discovery is an environment of awareness raising. Analysts should be able to effortlessly keep abreast of general and specific news and developments in their areas of interest. This can be achieved through the automated monitoring of key information sources. This monitoring system would be customized to individual needs. At the broadest level there would be global breaking news of nuclear related and more general events, including video bulletins. The next layer would be for receiving breaking news that is more specific to the analyst's countries or topics of interest. A third layer would be for the automated monitoring of any changes in an analyst's preferred websites. A fourth layer would automatically and continuously monitor high value databases, including scientific and technical ones, to detect critical events and activities. The technologies for developing such an awareness-raising system exist. SGIT/IIS has a cooperative programme with JRC-ISPRA for developing such a service.

3.2. Objective B: General information searching

Access to all the valuable sources of information, including reference material and favourite websites, would be through the portal. Querying would be through keyword searches. However, a key conceptual feature of the portal design is to ensure that analysts are guided to the sources most likely to contain the information they want or need.

3.2.1. Core sources

The majority of information searches are relatively straightforward for supporting information collection tasks. Many search queries can be satisfied through the interrogation of a few open source databases and some specific open source publications. Therefore, a collection of common, high value databases and other specific open sources would be a key feature of the portal. These are the 'core' sources on the portal. This core collection would cover key proliferation and security events, scientific and technical research in the nuclear fuel cycle, nuclear-related plans, nuclear issues of safeguards concern and nuclear-related imports and exports. Should the required information not be retrieved from the core sources, the portal would offer a three-tiered approach for directing the analysts to the most probable sources on the Internet.

3.2.2. Tier 1 internet

To minimize the information noise, the first step would be to subdivide the Internet into categories (or concepts) of specific interest, e.g. scientific and technical information, commercial information, companies, safeguards issues. Under each category, there would be a collection of valuable websites chosen by SGIT/IIS analysts. These websites would contain information not expected to be available in the core sources. They are the by-product of SGIT/IIS experience in information searches, and the list would be added to as new ones are discovered.

To further minimize the results list, SGIT/IIS has studied de-duplication technologies, together with the JRC-Ispra. These technologies offer the possibility to subtract the results from a blank search of the core sources from those retrieved using Tier 1 sources (assuming identical search queries in both cases). The removal of such duplicates would save time for the analyst by removing the documents already evaluated when processing the results of the search of core sources.

3.2.3. Tier 2 Internet

If searches on the core and the Tier 1 sources both draw blanks, then a search of Tier 2 sources would be the next avenue. Tier 2 establishes broader collections of websites for each category in Tier 1. It is premised on filtering the Internet into the same categories through query terms (invisible to the user). This would be accomplished through pre-processing the query 'under-the-hood'. This in-built 'intelligence' would be based on SGIT/IIS experience in information searches. The analyst would only have to select a category of interest and perform a keyword searches. The analyst's keyword would then be automatically added to the hidden keyword string. This would allow the analyst to search these higher value Internet collections without having to know or develop the highly complex search queries that define the categories. De-duplication, as described for Tier 1, could also be applied to minimize the results needed for evaluation.

3.2.4. Tier 3 internet

If blanks are drawn from the Tier 2 search, the analyst would still be able to resort to a full Internet search through the portal. However, more advanced nVISION technologies, such as clustering, would be needed to help in the evaluation of the large numbers of results.

3.2.5. Federated searching

The system would allow the analyst to query multiple sources and databases with a single query. Sources can be internal (e.g. databases, K2 indexes, LiveLink) or external (e.g. websites, online databases). This parallel or federated technology for querying would save significant time by minimizing the need to sequentially search each source or database. The user may choose any combination of sources to include when performing a query. SGIT/IIS has acquired a tool which can perform this type of federated search.

3.2.6. Portal user assistance

It is worth mentioning that this part of the system will help new staff members who have not been exposed to information collection and analysis, especially on nuclear issues, to develop an improved analytical mindset. The user friendly, intuitive 'help-always-at-hand' concept that is inherent in the design is intended to create an environment that will allow analysts to benefit from knowing that they can obtain useful search results with minimal effort. And the constant exposure to relevant awareness-raising information will improve an analyst's contextual knowledge. This type of knowledge is important for high quality analysis.

3.3. Objective C: Automated expansion of access to information

As mentioned above, most information searches are likely to be readily satisfied by the keyword and category searching of high quality sources and databases. Yet the remaining gap has to be bridged. This is a major challenge (and goal) of project nVISION – to significantly increase the probability of retrieving information of value from very large volumes of information, most of which has a very poor signal-to-noise ratio.

Objective C may seem paradoxical. It would actually increase the already overwhelming volume of information available to the analyst. There are three components for achieving this, as discussed briefly below:

— Mining the web. Firstly, technology has to be deployed for mining the deep or invisible web. Even the best Internet search engines can only index about 30% of the web. Project NVISION is currently investigating deep web miners to determine whether more information can be readily accessed from the web.

— Machine Translation. Secondly, there is a need to rapidly translate large volumes of non-English information. Much useful non-English information cannot be processed because of the extensive time and human effort involved. However, significant progress has been made in automated machine translation. The translation quality is expected to be of a suitable standard for analytical purposes.

The project intends to deploy two systems for machine translation, including an advanced statistical translation system that automatically learns and thus improves its ability to increase the quality of translations. The objective is to enable analysts to rapidly translate any volume of information to a quality sufficient for understanding the gist and identifying critical information. A Junior Professional Officer at the IAEA will be devoted to the machine translation part of project nVISION; and

— Name Variation Generator. Thirdly, information about individuals of interest to an analyst can be missed because of variations in name spelling. Project nVISION has acquired technology that accesses a database with almost one billion name variants. This technology also utilizes phonetics. When a name is entered, the tool can retrieve a list of spelling variations. These variants can be automatically combined into a single search query, rather than the analyst having to enter each name individually into a search engine.

3.4. Objective D: Automated division of large volumes of information into clusters of interest

The linguistic structure and semantic content of textual documents are machine analyzed. They are reduced to a mathematical signature. Those signatures that are similar can then be clustered. These clusters can be represented either visually or collated in document collections under machine-generated textual descriptions that represent the common themes of each cluster.

Current technologies cluster in a wholly unsupervised way – i.e. they identify common themes among documents. However, project nVISION's experience shows that this unsupervised process can create many clusters which may not be of particular value for the analysis clustering objective.

Project nVISION aims to create a clustering system which can be supervised or made hierarchical. In such a system, unsupervised clusters would also be mined to identify and weigh keywords of specific value, such as words with a high correlation to reprocessing, enrichment, conversion and weaponisation. These clusters would then re-position themselves on the user's screen according to the level of correlation to the fuel cycle. This would give the user a relevant way of choosing which clusters to interrogate rather than just selecting blindly. The project is preparing for a collaborative project with a laboratory for building a hierarchical clustering prototype. The project will also create the necessary taxonomies and lexicons required for supervised clustering.

Another clustering objective is to provide instantaneous summaries of clusters so that an analyst can quickly get the gist of the semantic themes. This would enable an analyst to quickly determine if a cluster requires deeper investigation.

3.5. Objective E: Automated extraction of entities and semantics from text

The content of the overwhelming majority of textual documents is essentially unstructured. There is limited classification or categorization of content. This lack of a regular structure is a major impediment for the analysis of large volumes of documents. The identification of patterns and inter-relationships — a fundamental requirement for high quality information analysis — can be more readily achieved within groups of structured documents.

Entity extraction — i.e. the extraction of, inter alia, the names of persons, places, and company names, as well as materials, technologies, dates — can rapidly identify critical categories or types of information contained in large volumes of information. The project has acquired a system for entity extraction. A priority goal is to develop natural language rules which will help to improve the efficiency of entity extraction. Higher efficiency in extraction will improve the analytical worth of unstructured document collections. Analysts will be required to help develop these rules. In addition, the project will employ an expert to oversee the process.

3.6. Objective F: Automated links analysis

Once information is in a structured form, it is possible to use links analysis tools to determine interconnections between entities or documents. This can include person-to-person connections, organisation-to-organisation, person-to-organisation, person-to-equipment, and so forth.

Project nVISION aims to make entities linking as automated as possible. Results from tests on information that already has some structure have been encouraging. The project has determined that scientific databases can be automatically parsed and uploaded into a links analysis tool and entity connections displayed and manipulated with relative ease. Importantly, the tests have shown that this process can be used to rapidly navigate large volumes of structured information and has revealed inter-connections between entities that would be virtually impossible to detect in a reasonable timeframe using a keyword search approach.

The objective now is to identify structured databases of value for information analysis so that they can be utilized for automated links analysis. Furthermore, in order to expand the application of automated links analysis, it is necessary to determine how best to exploit information that was initially unstructured (free text) and then automatically structured to a certain extent by entity extraction tools.

3.7. Objective G: Visualization

Links analysis and clustering tools provide the main form of visualization required for project nVISION. However, there are many products available which can also represent information visually in different and useful ways. The project will continue to review more visualization products and determine their applicability to analytical tasks.

During the tests of visualized information it became clear that users without a knowledge of proliferation issues could readily exploit the tools to frame and solve proliferation-related problems.

3.8. Objective H: Information management

Search results can be selected for further analytical processing or stored for future analysis. When saving the selected results, a ‘snap-shot’ of the data (e.g. website) will be stored, along with meta-data, such as source name, date and original url. Search results selected for immediate use can be put directly into an ‘analysis basket’; from here further processing of the documents can be done, e.g. clustering, summarization, de-duplication, visualization.

Alternatively, results stored in the structured folder can be readily grouped into sets, and the user can choose how to organize the sets into a hierarchical structure (similar to a file folder structure). Default folder structures (e.g. the fuel cycle) can be established for each user role; the individual user will have the ability to add to and modify this structure. Results from any of these folders could also be transferred to the analysis basket for further processing.

Users will be able to collaborate by sharing individual results or sets with other users and groups. Notification of a shared item can be accomplished through alerts, including email reminders and visual notification on the portal. Users will belong to certain groups according to their role, and may have the ability to ‘subscribe’ to other available groups, so as to have data pushed to them that they feel is relevant. Collaboration can also take the form of ‘publishing’, which would involve pushing document(s) to a more permanent location such as the Virtual State File.

4. Milestones

— Design and Implementation: precise dates on some aspects, but not all, depend on arrival of a cost free expert (CFE).

— March 2007: evaluate all applications and gather user analytical requirements.

— June 2007: design architecture including hardware choice.

— November 2007: test initially developed system.

— June 2008: implement improvements and optimize system.

— December 2008: complete full system to be deployed to users.

5. Conclusions

The concept of information-driven safeguards and the requirement that the conclusions concerning States' compliance with their safeguards obligations are based on evaluation of all available information means that the Department of Safeguards must deploy leading edge information systems and analytical strategies. The IRP and project nVISION are critical to these objectives.

Since its inception in January 2005, project nVISION has undertaken an ambitious mandate to expand the Department of Safeguards' knowledge of advanced technologies, with a view to making strategic decisions on defining a robust system to support the analysis of open source information. Project nVISION members have moved appreciably up a steep learning curve. Consequently, the project has been able to specify a system that encompasses open source analytical tools suitable for the novice through to advanced users.

In addition, project nVISION will help to foster the development of an analytical culture by continuously streaming awareness-raising information and alerts to key developments. Information will be customized for the individual user, allowing access to more relevant information rather than clutter. Project nVISION will also increase accessibility of a much greater volume of high quality information. It will enable automated translation of non-English documents. Document clustering, entity extraction, linking and visualization will allow analysts to effectively mine large volumes of information that has low signal-to-noise ratios.

Many of the key project nVISION technologies are complex, and completing the system requires critical skills that are not available in the IAEA. These skills need to be imported through recruitment, but it will be absolutely essential for these skills to become firmly established within the Department of Safeguards for the long term.

Early tests have shown that properly designed tools enable users at any level of proliferation knowledge, including those without such knowledge, to readily frame and answer proliferation-relevant questions and issues. It is a natural self-education and knowledge enhancing environment for all users.

Indicators and signatures*

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Abstract. The goal of this presentation is to give an idea of the methodology used to deal with proliferation problems. It can be useful for chemical, biological, balistical proliferation. Here, we underline nuclear proliferation scenarios. Nevertheless, the overall approach is also similar to activities related to terrorism.

Everyone knows that to strengthen the NPT/IAEA safeguards and similar treaties verification protocols, the organisations in charge need to build strong capabilities to assess known situations and also to prepare themselves to unknown, or undeclared events and activities.

To accomplish this, to collect, analyze, build ad hoc knowledge, organisations have to select the information, to manage the enormous amount of available data. Rather recently, the emergence of new crisis has confirmed the central and vital role that information processing plays at each levels of the international or national non-proliferation community. It is why looking for indicators and signatures is so important, to focus on pertinent information, that could mean something from a nuclear proliferation perspective. This allows people dealing with nuclear proliferation not to be overwhelmed by tons of paper or Gbites of memory.

<u>A strong need for expertise</u>

Identifying, select and following indicators or looking for signatures is not an easy task. It requires strong expertise. From the development and maintenance of its nuclear deterrence, France acquired expertise in the design, production of fissile material, manufacture and testing of nuclear weapons. There is also in France a long history of nuclear achievements, with small or large scale facilities, both in civilian and military fields; each step of the nuclear fuel cycle can be very precisely described.

French nuclear technical assessment relies on Commissariat à l'Energie Atomique (CEA, i.e. Atomic Energy Commission). Since 1958, CEA laboratories are in charge of nuclear civilian and military applications. Other expertises exist in nuclear fuel cycle operator like AREVA, COGEMA, in reactor knowledge in EDF, in balistic field in etatic or private laboratories. This very dense and comprehensive expertise network is a key element to feed, validate and strengthen non-proliferation activities. This expertise network, inside and outside CEA, allows to contribute to technical analyses for a very large range: from the mine, the reactor, to the weapon and the missile, from the modelisation to the production and the testing.

<u>An iterative and never-ending process</u>

An iteractive process is created and favorished between experts and non-proliferation teams. With their knowledge, experts help to choose the data, when they are available like open sources, satellite imaging, environmental analysis, equipment acquisition,... or help to define and develop new technics if needed. This very important work is guided by handbooks, keeping the knowledge, and list of indicators or signatures, specifically linked to one step of a proliferation scenario. It presents some similarities in the goals with the "Physical model" developped and run by the AIEA.

* Only an abstract is presented here, as the full paper was not available.

Indicators are of numerous nature, they can be results of bibliometry, of equipments interests, of specific analysis or visual observations. Every time it is possible, new technologies for survey or monitoring are implemented and compared to old ones and tested against known proliferation scenarios. If the conclusions are positive, it is to say that with this new technology or method the assessment could have been better, the technology is kept, if not, an other approach is needed to find the indicator or to measure the appropriate signature.

The process must be always an iterative one between analysts, nuclear experts, tools experts, like bibliometry, analysis, imagery for instance to allow tu use the best technic available for the good indicator; the process must also be fed and tested by events or facts that make nearly each day the proliferation News.

So indicators and signatures are far from being written in marble; they are continually adjusted and optimised to curb and prevent nuclear proliferation.

A fuzzy logic decision support system for open source information analysis in a non-proliferation framework*

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Abstract. In the strengthened safeguards framework, information analysis has gained renewed attention. With the entry into force of the Additional Protocol, increasing amounts of information need to be analysed in order to assess States' compliance with their safeguards undertakings. The analysis basically consists in cross-checking States' declarations, inspection results and open source information at State level. Here, a decision support system is proposed, which provides a synthesis of open source information. It is comprised of three modules: an input module, where information is translated in a computable form; an aggregation module for its synthesis following an original fuzzy logic methodology; and the output module giving an overall view of the proliferation capabilities reached at State level. The aggregation methodology builds upon IAEA proliferation indicators and takes into consideration both the reliability of the information sources and the relevance of the indicators.

1. Introduction

Since the transition to the strengthened safeguards regime, the International Atomic Energy Agency faces the necessity of handling a larger and increasing amount of information in order to assess States' compliance with their safeguards undertakings. Besides traditional safeguards agreements, the Additional Protocol requires that States provide declarations pertaining a wider range of information including materials, equipment, know how and sites, directly or indirectly related to the nuclear fuel cycle. Through the State evaluation process, analysts check these declarations against inspections' results and other information sources, including open sources, and draw conclusions about the absence of undeclared nuclear materials.
A decision support system (DSS) is proposed for the synthesis of open source information. The DSS builds upon a comprehensive collection of proliferation indicators, the IAEA Physical Model [1], and makes use of fuzzy logic. An original methodology has been developed for the aggregation of information, taking into account both the relevance and the reliability of the information. It will be outlined and illustrated hereafter. Finally, attention has been paid to produce a readable output, giving the analyst an immediate representation of the State nuclear capabilities according to open sources.

2. The DSS

2.1. Premise

Proliferation indicators are here considered as the smallest piece of knowledge in which pertinent information can be portioned. Once relevant information has been identified, experts assign proliferation indicators values that represent the level to which the corresponding indicators are deemed to be present according to that source. Linguistic values are employed, allowing seizing the peculiarity of open source information which can be uncertain and imprecise.

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* excerpt from PhD thesis [3]

Fuzzy sets have been identified as the most appropriate mathematical framework for computing with words [7]. Differently from classical sets, fuzzy sets have smooth boundaries and can be overlapping, so that one element can belong to a certain extent to more than one set. Here the universe of discourse, i.e. all the values that a proliferation indicator can assume, has been portioned into five fuzzy subsets, and each of them has been associated with a linguistic label as follows: DN, definitively not [present], RN, rather not [present], UN, uncertain, RY, rather yes [present], DY, definitively yes [present]. Formally, the subsets are described through Gaussian-type membership functions (Figure 1); semantically, they are associated with possibility distributions, meaning that each of them represents the distribution of the possibility that a given indicator is present or not in a given country [8].

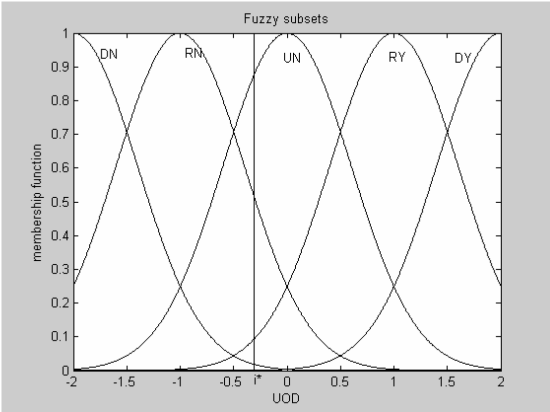


Figure 1. The subdivision of the UOD in five fuzzy subsets and the corresponding linguistic values.

Two characteristics of information play a significant role in the design of the DSS: relevance and reliability. Relevance is related to the importance an indicator assumes in a proliferation path. The Physical Model is a detailed description of all the elements potentially present in the path towards the acquisition of sensitive nuclear material. Each indicator is characterised by a *strength***,** so that an element can be a strong, a medium or a weak hint of the presence of a given technology: e.g. a *centrifuge* is a strong indicator of the presence of the UF6 enrichment process by gas centrifugation, a *molecular pump* is a medium indicator and *titanium alloys* is a weak indicator [1].Clearly, relevance is an intrinsic characteristic of the indicator.
Differently, reliability is related to the source providing the information. For that purpose, information sources have been ranked by experts who have assigned them a reliability level in a range of three: high (HI), medium (ME) or low (LO) [2].

Lastly, three principles have been stated in order to compensate to some extent the intrinsic arbitrariness of the methodology: (a) transparency of methodology, that allows sharing and eventually adjusting the aggregation mechanism; (b) multiplicity of sources: all information sources are considered, each one with its reliability level, (c) traceability of information: at each step of the aggregation process it must be possible to go back to the original input information.
Schematically, the DSS is made of three modules: the input module (2.2), where open source information is transformed into indicators' values; the aggregation module (2.3), where indicators' values are combined and synthesis indicators' values are obtained; and the output module (2.4) presenting the synthesis values in an overall view of a State's proliferation capabilities.

2.2. The input module: from open sources to indicators' values

The aims of the input module are a) to identify and index relevant documents retrieved within accredited sources, b) to analyse the relevance of the information contained in documents and c) to assign values to proliferation indicators according to the contents of the documents.
Most of the steps are necessarily performed by experts, i.e. ranking the information sources, identifying potentially useful documents, reading and analysing them. However, a commercial software tool has been applied for the indexing of the documents, and specific tools have been developed for the search of documents and for the assignment of values to indicators [2]. A topic tree has been designed, allowing for a semantic search rather than a Boolean search by keywords. Based on IAEA Physical Model, the topic tree

performs search operations on elementary pieces of information, such as a single proliferation indicator (a leaf of the tree), or on higher levels, e.g. the *UF6 enrichment process*, which drags into the search all the lower levels, including the UF6 enrichment technologies described in the Physical Model (centrifuge, diffusion, molecular laser and aerodynamic) with all the related indicators.
When relevant documents are found, the expert reads and interprets them, then assigns linguistic values to the matching proliferation indicators. Each linguistic value (DN, RN, UN, RY, DY) corresponds to the degree to which it is possible that a given indicator is present in a country, according to the document retrieved. This is a first analysis step discriminating relevant information. Moreover, for the multiplicity principle, all accredited information sources are taken into consideration, each with its reliability level. Therefore indicators can be assigned multiple values, not necessarily coherent with each other, but each of them being associated with a reliability level. A graphical interface has been developed which supports this step [2].
The yield of the input section is for each State a set of proliferation indicators'. Each value is associated with a reliability level, and each indicator is associated with a strength. All these data are stored in a multiple linguistic values database.

2.3. The aggregation module: the synthesis of information

In the second module, proliferation indicators are aggregated in order to produce synthesis values giving a concise description of the nuclear capabilities reached by a given State.
The Physical Model is the description of over 900 indicators of the nuclear fuel cycle, at three levels: the phase level (e.g. enrichment), the process level (e.g. enrichment of UF6, UCl4 or Umet), and the technology level (for UF6 enrichment, e.g., centrifugation, diffusion, aerodynamic, molecular laser). Synthesis values are first obtained at technology level and subsequently aggregated at process level.
The aggregation methodology developed provides the combination of multiple linguistic values into numerical synthesis values, taking into account both the reliability and the relevance of the information [3].The methodology addresses two logical systems, both based on fuzzy sets: the Possibility Theory, allowing for a mathematical aggregation of information through combination operators [4] and Approximate Reasoning [5], where fuzzy inference systems make use of fuzzy rule bases for a knowledge driven aggregation. The aggregation proceeds in two phases, each subdivided into 2 steps.

2.3.1. First aggregation phase

In the first phase, aggregation is performed according to reliability. The aim is to obtain one single value for each indicator, by aggregating multiple values, and taking into account the respective reliability levels. The process is a 2-step process.

- step 1: for each indicator, the values provided by information sources of equivalent reliability level are combined. As a result, three components, i.e. a high-reliability component, a medium-reliability component and a low-reliability component are obtained for each of them.
  A mathematical combination is suitable, and combination operators are selected according to the characteristics of the information. It is assumed that all the information is to be taken into account, that sources of equivalent level are interchangeable, but that no conclusion can be drawn about their independency. In Possibility Theory, one operator corresponding to these requirements is the *minimum* operator.
- step 2: for each indicator, the three components obtained at step 1 are combined, giving priority to those of higher reliability. The result is one single value for each indicator.
  Aggregation is performed considering only information common to all the components, giving higher priority to higher reliability components. The corresponding operator is *priority maximum*, which fully considers the information provided by the most reliable source, whereas the less reliable components are taken into account only as far as they are in agreement with the former.

2.3.2. Second aggregation phase

In the second phase, for a given technology described through a set of indicators, the single values obtained in the previous phase for each indicator are aggregated taking into account their respective relevance. The result is a synthesis value of the presence of that technology in a given country. As for the first phase, it is a 2-step process.

- step 3: for each technology, indicators with equivalent strength are aggregated in order to obtain a strong, a medium and a weak synthesis component (or “evidence”) of the presence of that given technology.
  As for step 1, all the information is assumed to be useful and interchangeable, but here redundant information enhances the precision of thr result and a reinforcement effect is necessary. Therefore a *product* operator is selected to formalise the operation.
- step 4: for each technology, the three components obtained at the previous step are aggregated into one synthesis value of the presence of that given technology.
  The combination of the three evidences obtained prevously is not performed through a mathematical operator anymore. It is assumed instead that a knowledge driven process is more suitable and that experts can evaluate how the combined presence, to different levels, of high, medium and weak evidences can result in a final synthesis value of the presence of the technology. A fuzzy inference system is adopted supported by a rule-base written with the contribution of experts [6].
  The output of this last step is a fuzzy set membership function which, after defuzzyfication, reduces to a numerical value. This corresponds to the possibility that a given technology is present in a given State.

For a given State, the aggregation methodology is applied to each technology, yielding to synthesis numerical values at technology level that are further elaborated by the output module.

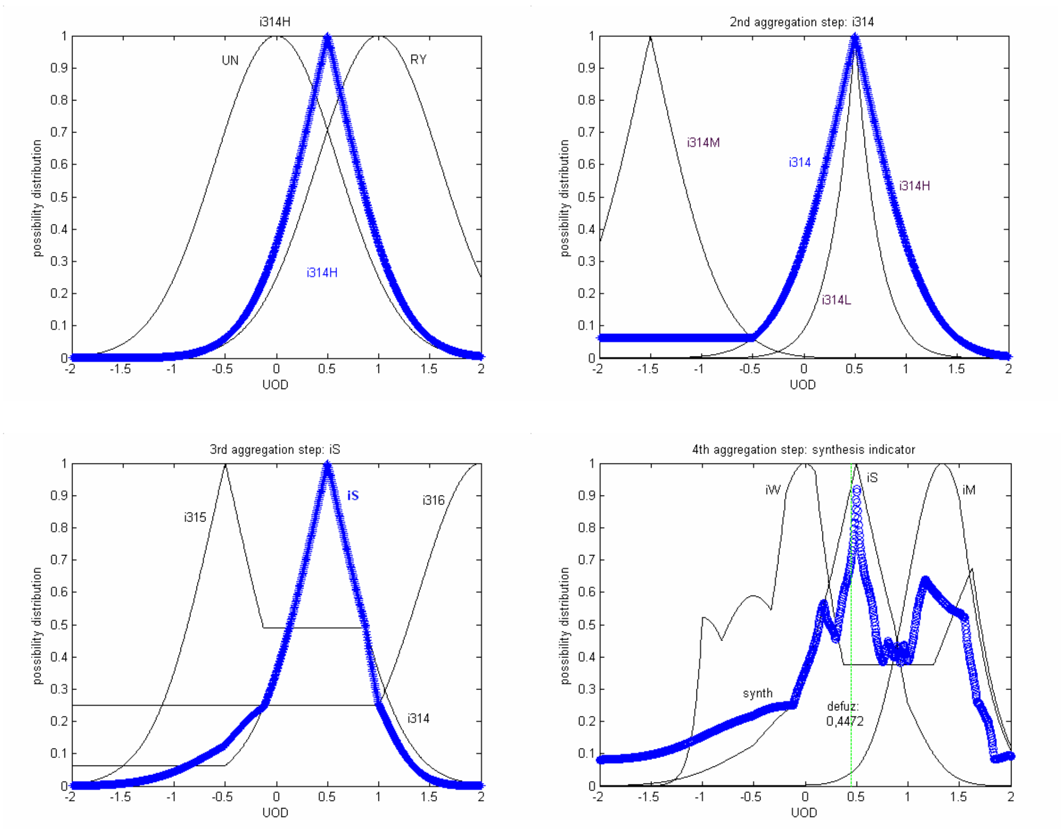


Figure 2. Combination tools for the four steps of the information aggregation process.

2.3.3. Example

The aggregation methodology is briefly illustrated through a simplified case. We assume that a given technology is entirely described by 9 indicators, and that information reported in 8 information sources allow assigning them values, as shown in Table 1.

		HIGH reliability			MEDIUM reliability		LOW reliablity		
		S1	S2	S3	S4	S5	S6	S7	S8
STRONG indicator	i314	UN	RY	UN	DN	RN	UN	RN	DY
	i315		UN	RN	RY	UN	DY	DN	
	i316	DY			UN				DN
MEDIUM indicator	i200		RY	DY	DN	UN	RY	DY	
	i201	RY			RN		UN		
	i202		RY			DY		UN	
WEAK indicator	i150	DY	DY	DY	UN	RY	DY	DY	UN
	i151		DN	UN	RN	UN	UN		DY
	i152	DN	UN	DN		RY		RY	

Table 1. Input values for the simplified description of a given technology.

In figure 3 the linguistic values are translated into Gaussian-type membership functions, and the overall aggregation process is illustrated. In the first three columns, the input data for high, medium and low reliability sources respectively are represented; coloured curves represent the result of the first aggregation step. In the fourth column, the red curves represent the result of the second aggregation step. The fifth column reports the 3 evidences obtained after the third step, and the graph in the sixth column corresponds to the last step, aggregation through the fuzzy inference system. The value resulting after defuzzyfication is 0,4472, corresponding to UN, uncertainly present. Some simple sensitivity tests have been performed that confirm the self consistency of the system [6].

2.4. The Output module

The previous module drastically synthesizes information and produces numerical values associated with the possible presence of nuclear technologies at State level. The output module is intended to present to the final user a more readable output, in a graphical form. The support is the Physical Model flow chart (Figure 3), where each box represents a process. The chart is made interactive in such a way that the underlying technology level can be accessed through the higher process level. The synthesis values produced by the aggregation module are translated into a warning colours scale, allowing for an immediate visualisation of the critical points where, according to open sources, technical capacities are reached.

3. Conclusions

The present work focuses on the development of a decision support system for open source information analysis as a contribution to enhance the effectiveness of the information analysis process in the nuclear non proliferation framework. It confirms the feasibility of the approach, which makes use of proliferation indicators as atoms of information and of fuzzy sets as their computable counterpart. The methodology outlined for the indicators' aggregation process proves self-consistent after simple sensitivity tests. Particular attention has been paid to a most readable presentation of the final result in the form of a physical model flow chart.
Further developments include the study of an updating procedure, since open source information is undoubtedly changing at a rapid pace. One more improvement would be the adjunction of the quantitative dimension to indicators relating to materials. Finally, the present work concentrated on the aggregation of the proliferation indicators described in the IAEA Physical Model and on open information sources. It can be foreseen to enlarge the set of indicators to other fields: in nuclear proliferation control, political or economical indicators could contribute to a broader assessment of a given country; or in other proliferation

control frameworks, such as for chemical or biological weapons of mass destruction. The system can also apply to information sources other than open sources: States' declarations or inspections' results can be similarly synthesised, thus providing homogeneous and directly comparable results.

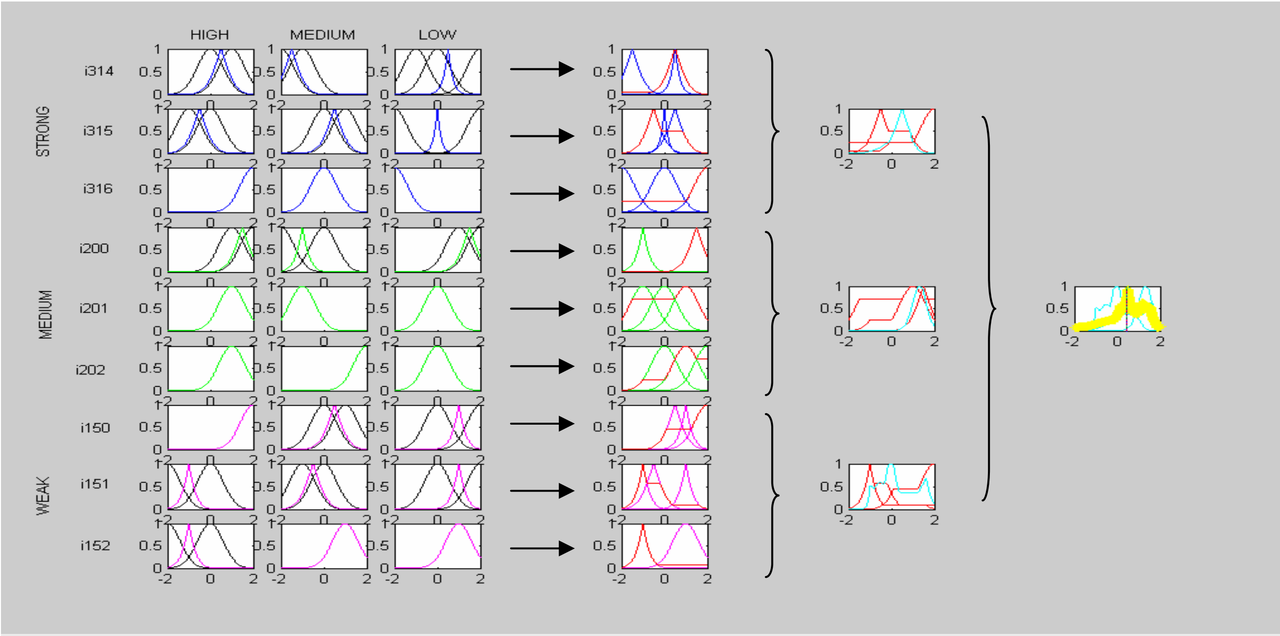
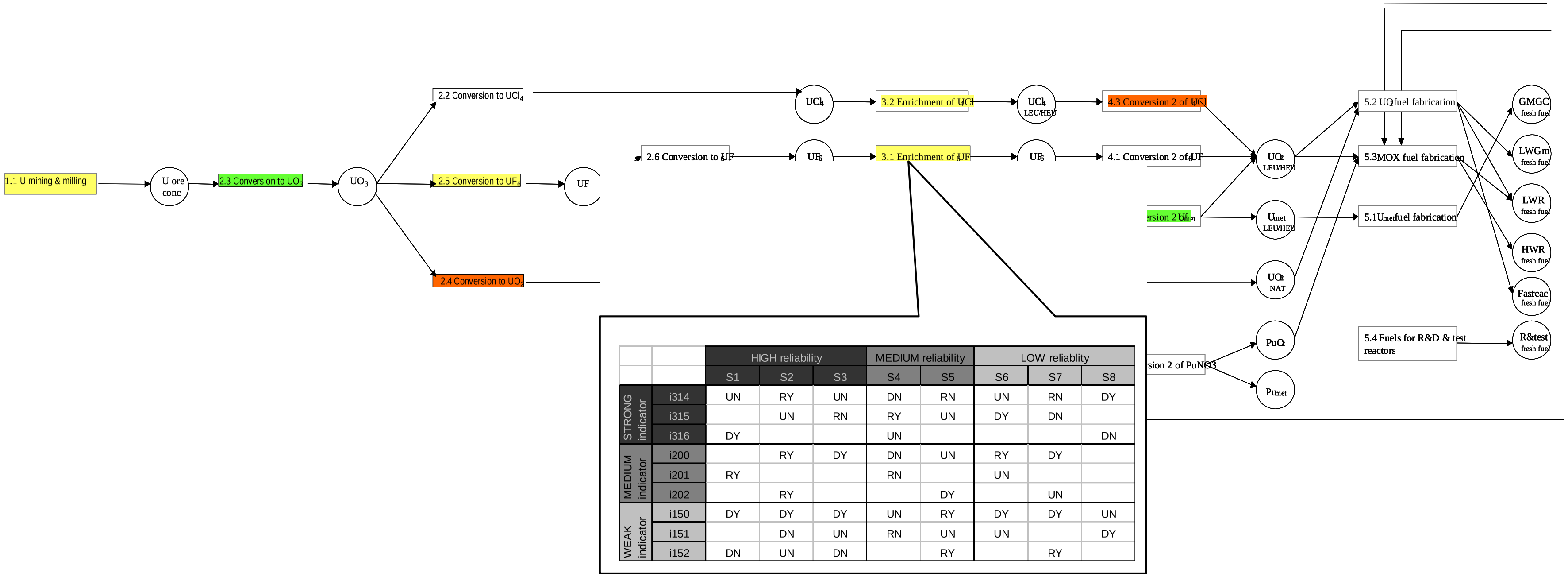


Figure 3. The overall aggregation process referring to input data in Table 1.



		HIGH reliability			MEDIUM reliability		LOW reliablity		
		S1	S2	S3	S4	S5	S6	S7	S8
STRONG indicator	i314	UN	RY	UN	DN	RN	UN	RN	DY
	i315		UN	RN	RY	UN	DY	DN	
	i316	DY			UN				DN
MEDIUM indicator	i200		RY	DY	DN	UN	RY	DY	
	i201	RY			RN		UN		
	i202		RY			DY		UN	
WEAK indicator	i150	DY	DY	DY	UN	RY	DY	DY	UN
	i151		DN	UN	RN	UN	UN		DY
	i152	DN	UN	DN		RY		RY	

Figure 3. Physical Model flow chart (front end) with hypothetical synthesis results and link to input data.

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Future safeguards verification tools

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Abstract. The IAEA operates a large diversity of equipment to verify Member States' declarations about nuclear materials made under their safeguards agreements, pursuant to the Treaty for the Non-Proliferation of Nuclear Weapons (NPT). Verification of nuclear materials is a cornerstone of the safeguards system. The variety of physical and chemical properties of nuclear materials, including their storage environment, requires an arsenal of instruments. IAEA needs are changing and ongoing equipment development is needed to provide its inspectorate with the verification tools for performing the verification activities. This paper discusses new and improved technologies to enhance and adapt safeguards verification tools. Verification activities range from the measurement of declared nuclear material inventories and flows, to the search for indicators of undeclared nuclear materials and undeclared nuclear activities such as enrichment and reprocessing.

The IAEA is facing increasing demands to perform remote verification of nuclear material flows utilizing unattended monitoring systems. Additional analytical capabilities for environmental sampling and effective non-destructive assay (NDA) methods are indispensable for the IAEA's ability to detect undeclared nuclear materials and activities and to support Member States in the detection of, and response, to illicit trafficking.

1. Introduction

Safeguards are applied by the IAEA to verify that commitments made by States under safeguards agreements are fulfilled. The safeguards system is designed to verify the declarations made by States about the exclusively peaceful use of their nuclear material and activities. With the additional aim of deterring any diversion through the risk of early detection, the IAEA independently verifies the non-diversion of declared nuclear material and the absence of undeclared material and activities in a State. A wide variety of tools are used to verify nuclear material declarations in a timely manner, including NDA equipment, remote monitoring systems, destructive analysis (DA), environmental sampling and containment and surveillance (C/S) systems. The results of these verification activities, combined with the results of other safeguards measures such as information analysis and evaluation, form the basis of the annual safeguards conclusions drawn for States about the correctness and completeness of their declarations.

The IAEA's needs are identified in the Strategic Equipment and Technology Development Plan for 2006–2011 [1][2]. The following paragraphs discuss these equipment needs and consider the technical vision required to help the IAEA meet its safeguards objectives.

2. Future inspection schemes

The IAEA needs improved detection capabilities and novel tools to meet the growing number of safeguards challenges, such as the detection of indicators of undeclared nuclear materials and activities. The future goal is to possess new and improved equipment/technology that will address present deficiencies, have higher reliability, enhance the efficiency and effectiveness of verification activities and ensure the security of information that can be provided in a timely manner.

The advancing implementation of integrated safeguards — the optimum combination of all safeguards measures available to the IAEA — will significantly reduce the number of routine inspections and physical inventory verifications (PIVs). However, the number of unannounced inspections and complementary access visits, which are a cornerstone of integrated safeguards approaches, will increase sharply. Instruments for those specific types of verification activities need to be made available to inspectors at very short notice. Such equipment must be multipurpose, easy to operate and portable in order to allow the inspector to perform various activities within the short time span of the verification activity, including the search for indicators of undeclared nuclear materials and activities. New plant designs with complex operational schemes will require the development of new, design-specific safeguard systems.

3. Improvement of current detection systems

Advancements in data treatment capabilities will provide additional instrument functionality, such as location stamped information and improved nuclear material identification, through the use of information organized in embedded databases. Other advancements in detector technologies (e.g. NaI, CdZnTe, LaBr3 and HPGe) and signal processing hold the promise of increasing the resolution and efficiency of radiation detectors [3]. Current high resolution gamma spectroscopy (HRGS) systems are based on low temperature cooled HPGe detectors, which sometimes limit their application in the field. Less expensive, smaller HRGS systems with reduced power consumption could be designed provided that the detectors can work at room temperature. Table 1 summarizes the currently available detector types and their potential use for safeguards purposes.

Miniaturization will continue to be important, facilitating the conversion of bulky equipment into smart portable equipment designed for field applications. Such equipment should be easy to use, robust and able to provide multipurpose detection functions. A device is under development consisting of a wireless control unit, neutron (He-3 tubes) and gamma detectors (CdZnTe) packed in a customized lightweight body pack. The spectral capabilities of the gamma detector would identify the presence of radioisotopes while the neutron response would be indicative of nuclear materials. The design includes wireless transmission of the sensor data which would allow the inspector to perform freely other activities while the body pack continuously screens for radiation.

Table 1. Available detectors for safeguards purposes.

Detector Type	ER (1)	RE (2)	Application	Description
HpGe	H	-	U – Pu isotopics (low energy range)	Best resolution (~ 550 eV at 122 keV); LN2 cooling required, electrically cooled systems available but expensive and subject to degration in energy resolution
CZT500	M	Ref.	Attribute U-Pu-SF	Medium sensitivity, no cooling, medium resolution, widely used
CdTe	M	=	U – Pu isotopics (>100 keV range)	Medium sensitivity, Peltier cooled (possible replacement for HpGe).
NaI(Tl)	L	+	Attribute U-Pu (e.g. HM-5) Isotope identification, Enrichment	High sensitivity, no cooling, low resolution
LaBr3:Ce LaCl3:Ce	M	+	Attribute U-Pu Isotope identification	High sensitivity, no cooling, medium resolution (possible replacement for NaI detectors)
HgI2	M	-	U – Pu isotopics (>100 keV range)	Solid state detector with low sensitivity, no cooling, medium resolution (insufficient for possible replacement for HpGe detectors)
LiSi:Ce glass	L	+	search for NM by neutron/gamma detection	High sensitivity, no cooling, low resolution, inadequate neutron-gamma discrimination.
LiI:Eu	L	+	search for NM by neutron/gamma detection	High sensitivity, no cooling, low resolution, adequate neutron-gamma discrimination, further testing needed.
Si		-	Gross gamma detection for counting of SF (e.g. VIFM)	Silicon photodiode used at room temperature
Remarks: (1) ER = Energy resolution is quoted H-M-L (typical values: H 550 eV, M 2.8 keV, L > 8 keV) (2) RE = Efficiency relative to CZT500 (3) Different CZT detectors (CZT 1500,CPG A2250,CPG A2317, CZT CAP 2157, CZT CAP 2156) are used/tested for customized SG applications (4) GaAs, CdWO4, BGO, Xe have low ER and currently no SG use foreseen				

4. Tools for detection of undeclared materials and activities

Confirming the absence of indicators of undeclared nuclear materials and activities demands enhanced instrument capabilities [4]. For example, complementary access requires detection devices to search for non-traditional elements/isotopes (such as americium, neptunium, beryllium, and tritium) that could indicate the presence of a clandestine nuclear weapons programme.

A portable instrument for HF gas monitoring is also being developed, designed for easy operation in airborne and ground-based mobile searches for enrichment activities. The IAEA has great interest in technologies for the detection and quantification of enrichment and reprocessing activities. At present, the IAEA lacks an NDA method for detecting or locating undeclared enrichment or reprocessing activities at a distance. Several promising technologies exist, however. For example, tuneable laser diode systems have the potential to determine ^{235}U enrichment in UF_6 gas and to indicate the presence of HF gas, a by-product of enrichment activities. The application of differential absorption light detection and ranging (LIDAR) techniques to measure the on-site trace level of elemental or chemical compounds, such as volatilized solvents used during reprocessing (Tributylphosphate (TBP)), could be developed as a possible technique to confirm the absence or presence of reprocessing or enrichment activities. The potential use of noble gas monitoring and sampling for the detection of reprocessing activities is also being considered.

Design information verification (DIV) is an important measure to confirm that existing facilities are used as declared by an operator or the State and to detect the presence of undeclared design features and hidden facilities which could indicate undeclared nuclear activities or the diversion of nuclear material. Environmental sampling has proven to be an excellent tool to determine past and present usage of nuclear materials. Its unmatched sensitivity using particle analysis is used to detect enrichment or reprocessing activities. However, the complexity of the analysis, the costs involved and the time lapse between sampling and actual results are limiting its routine potential. Alternative methods such as commercially available laser ablation spectrometers are being considered to reduce the increasing number of environmental samples on a case-by-case basis. Instruments using optical stimulated luminescence (OSL) will soon be commercially available and can be applied to investigate the past presence of radiation emitting substances.

Among several geophysical methods, ground penetrating radar (GPR) [5] has been selected as an approved technology for the detection of hidden objects and structures. However, the immediate and unequivocal interpretation of the resulting images is very difficult, and additional measures are needed to reach a final conclusion.

5. Improved verification techniques for enrichment plants

The safeguarding of enrichment plants is a high priority task for the IAEA and the number of enrichment plants will significantly increase in the coming years. New approaches for safeguarding enrichment plants are under development, which call for better technology for in-situ verification measurements and monitoring systems. The development of in-line NDA instrumentation will permit the monitoring of flow and/or enrichment levels, i.e. provide confirmation that the level of enrichment does not exceed the declared maximum and the absence of any high enriched uranium (HEU) production. Furthermore, any new approach should guard against excess production of enriched uranium from undeclared feed materials that could be subsequently used as feed for a much smaller, clandestine enrichment facility.

The IAEA therefore needs instrumentation to quantify all materials in the various process stages, including those in vessels connected to the process, in order to strike an effective balance among feed, products and tails. This entails authentication of the various operators' load cells (process and accountability) to derive a real-time balance of feed and withdrawal of UF_6. A laser item identification system (LIIS) is under development to continuously track the flow of UF_6 cylinders in the facility by their unique surface identification [6]. Diode laser absorption spectroscopy is an established analytical technique and is being developed for UF_6 measurements, with the aim to partly replace the need for DA and thereby improve timeliness and reduce inspection resources.

6. Improved verification techniques for plutonium handling facilities

Recently, a large reprocessing facility has commenced its active test and the safeguards instrumentation in place has set a new technological standard in terms of networking and integration of unattended verification systems and C/S systems covering the complete process and storage areas. Although the verification systems deliver data consistent with international target values, the combined measurement uncertainties in the verification of the plutonium strata may easily exceed the limits set by the safeguards goal of detecting the loss of one significant quantity. Additional assurance through measures such as timely access to operational data and checking of plant specific parameters may be required.

A new class of monitoring systems for in process materials operating on a real-time basis can only be realized through sophisticated software. Furthermore, the ability to perform Monte Carlo simulations for the design and calibration of the monitoring systems will be indispensable. Future reduction in inspection efforts can only be realized by the full use of an integrated network of the various verification and monitoring systems and intelligent data evaluation packages. This requires the highest level of reliability of the systems in use and quickly available resources for repair if components of the network fail.

7. Improved verification techniques for spent fuel

The quantitative and qualitative verification of spent fuel in wet and dry storage areas continues to be of key interest to the IAEA. The number of spent fuel assemblies (SFAs) is rapidly increasing and, as reprocessing capacities are not developing at a commensurate rate, this number is starting to exceed the storage capacity of most spent fuel ponds at nuclear power plants. As a consequence, spent fuel is transferred to dry storage casks for intermediate or long-term storage.

7.1. Spent fuel in wet storage

NDA methods are the only means of verifying spent fuel. The most advanced verification tools for spent fuel in wet storage are the digital Cerenkov viewing device (DCVD) and the safeguards MOX Python (SMOPY) device [7]. SMOPY combines gross neutron counting with gamma spectroscopy to verify and distinguish irradiated MOX fuel from low enriched uranium (LEU) fuel and to confirm burn up. This device will soon be commercially available. Further development of the DCVD is still ongoing to provide additional functionality for partial defect testing and optimised usability.

Three major problems still exist for the verification of spent fuel:

— The IAEA does not have a universal partial defect method for verifying SFAs. Monte Carlo simulations of the partial defect capability of NDA equipment suggest that a package of instruments (FDET, SMOPY, DCVD), combined with weighing and visual inspection of the fuel assembly, is required for the detection of partial defects.
— Most measurements are very intrusive from the perspective of the facility operator who is generally reluctant to move and isolate fuel for safeguards purposes. Therefore, such measurements could only be made at the time of the fuel transfer.
— Measurements involving the installation of underwater equipment are typically manpower-intensive and require significant inspection resources.

A passive gamma emission tomograph, currently being developed with the help of several Member States Support Programmes (MSSPs), is projected to be able to detect defects at a pin level.

7.2. Spent fuel transfers

As spent fuel is moved from wet storage to difficult-to-access dry storage casks, the IAEA maintains continuity of knowledge while the casks are transferred to and positioned in a permanent storage location. Significant inspection resources are presently needed to verify the loading of a spent fuel cask with FSAs, which become virtually inaccessible after the cask has been closed. The development of unattended verification systems to monitor the loading of SFAs is the only means of decreasing inspection resources.

Such systems have been recently developed for CANDU reactors to monitor the complete process of loading, transferring and storing of spent fuel in dry storage silos. This process starts underwater in the spent fuel bay with the loading of spent fuel bundles into a basket. Once dried and welded, the basket is transferred to the dry storage facility inside a shielded flask. The final step consists in lowering the basket from the transfer flask into the concrete silo. The quantification of loaded bundles is performed with a VXI fuel monitor (VIFM) timer-counter system (Figure 1) with accompanying underwater surveillance measures. Continuity of knowledge during transfer is maintained by using a mobile unit neutron detector (MUND) mounted in a waterproof enclosure fixed and sealed on the upper part of the transfer flask. The MUND is a small, battery-operated device for up to eight weeks of continuous operation, allowing quick service by swapping the unit with a spare one. The silo entry gamma monitor (SEGM) monitors the final silo loading. SEGM is constituted by a pair of gross gamma silicon detectors inserted at different levels into the verification tubes available for each silo providing direction sensitive verification of the silo loading (Figure 2). The detectors can be easily relocated to empty silos by inspectors when the permanent sealing of a full silo is completed.

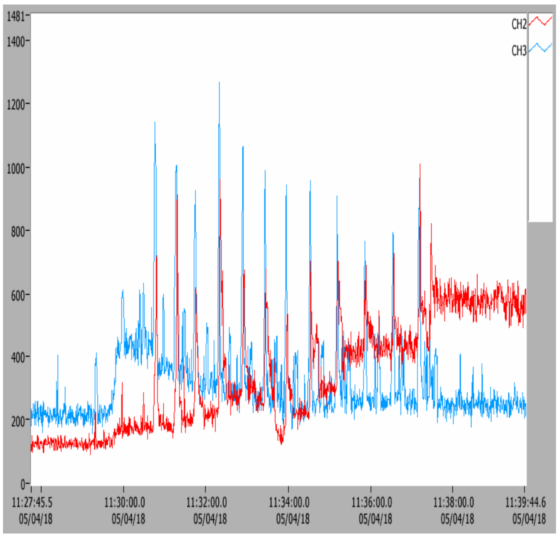

FIG. 1. Typical VIFM results.

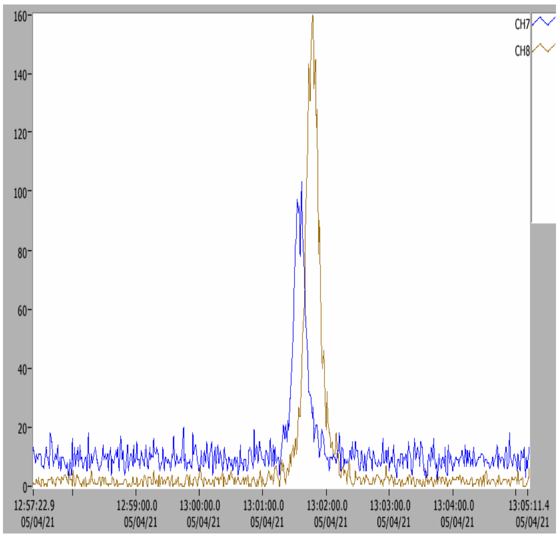

FIG. 2. Results from SEGM.

7.3. Spent fuel in dry storage

C/S systems will be the principal safeguards mechanisms used to safeguard spent fuel in dry storages. In the event of a loss of C/S data or as a part of periodic routine requirements for re-verification, the IAEA requires methods to demonstrate that the cask content has not changed.

Two possibilities exist to re-verify spent fuel in dry storage containers: (a) to quantify the content of the container by an independent NDA measurement, or (b) to compare a neutron/gamma ‘fingerprint’ taken at the time of loading with a fingerprint taken at the time of re-verification. The quantitative option has proven to be difficult to achieve due to the limited penetration of radiation from the inner assemblies and the interference of neighbouring casks when measuring neutrons. The gamma scan of the canister should be unique for the loading of a given canister. However, non-nuclear radioisotopes can also generate gross gamma rates and a further distinction using a neutron measurement is currently being developed. In addition, gamma signatures typically only originate from the outer layer of the spent fuel load. The maintenance and/or restoration of continuity of knowledge of a spent fuel dry storage container by a reproducible fingerprint require a systematic management of fingerprints over a long period of time. A database for storage and evaluation of fingerprints is under development to secure and effectively compare fingerprints, taking into account the decay and changes of the measurement hardware configuration. An assessment of the sensitivity level for the detection of the removal of spent fuel items is underway.

8. Improved verification of waste, hold-up and in-process inventories

The content of nuclear material in waste drums is usually very low and not homogeneously distributed in the drum. Representative sampling is therefore unfeasible, and the IAEA needs to employ NDA methods instead. Bulk handling facilities such as reprocessing and fuel fabrication plants often accumulate a huge number of waste containers that together contain significant amounts of nuclear material.

Although several NDA methods exist to tackle the problem of hold-up accounting in bulk facilities, the related measurement uncertainties often exceed the specified goal. The in-situ object counting system (ISOCS) — a commercially available germanium-based spectroscopy system with embedded computerized numerical calibration capability — verifies nuclear materials contained in hold-up and waste. Specialized remote, often physically large, hold-up counters based on neutron coincidence techniques are being developed to quantify and monitor plutonium inventories in the process of an MOX fuel fabrication plant on a real time basis.

9. Improved detection capability of illicit trafficking

The IAEA continues to enhance and develop equipment to counter illicit trafficking of nuclear materials, capitalizing upon the synergies between safeguards equipment and instruments used to detect radiation at borders, terminals and other venues.

Such NDA systems must provide a quick analysis in order to cope with the enormous traffic of goods and people. Integration of nuclear material and other radioactive source monitoring systems with monitoring systems for other hazardous and sensitive materials (e.g. explosives) would minimize the intrusiveness of the control measures. In order to reduce the number of follow-up control measures and to avoid possible evacuation measures, the number of false alarms should be minimized. Active detection methods for shielded nuclear materials (e.g. prompt gamma activation analysis (PGAA)) are being considered for possible development. In addition to fixed installed portal monitors, the IAEA needs to further improve its portable equipment to quickly identify any radioactive material, including possible radiological dispersion devices (RDD).

10. Verification tools supporting the disposition of excess nuclear weapon materials

The IAEA is exploring technology to verify nuclear materials declared as excess by nuclear weapons States (NWSs). The challenge is to provide verification tools to draw soundly based safeguards conclusions, without disclosing and knowing the characteristics of the disposed nuclear material. An attribute verification system with information barrier for plutonium, with classified characteristics utilizing neutron multiplicity counting and high resolution gamma spectrometry (AVNG), is under development by two NWSs. The multiplicity counting will be performed with liquid scintillators, while a HRGS system is used for the isotopic composition. The system will provide a final result without providing raw data.

11. Conclusions

Numerous NDA instruments and components for measuring nuclear and physical properties are available. Emerging challenges to verify and detect nuclear material demand the ongoing adaptation of existing instrumentation and the development of new equipment.

The IAEA places high priority on improving its verification techniques for enrichment plants and pays particular attention to the development of new, alternative technologies that could be used in the search for undeclared facilities, nuclear material and activities.

The verification of spent fuel remains a focus of safeguards activities.

Possessing limited R&D capabilities, the IAEA relies mainly upon the support of its Member States, which should also ensure the involvement of R&D organizations operating outside of the traditional safeguards community.

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Novel technologies for the detection of undeclared nuclear activities

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Abstract. Implementation of international nuclear safeguards has changed considerably over the past 15 years. Early detection of undeclared facilities, activities and materials requires new approaches, supplemented by technologies that differ significantly from those used traditionally for on-site verification. Within this framework, the IAEA developed the project 'Novel Techniques and Instruments for Detection of Undeclared Nuclear Materials, Activities and Facilities' to identify specific needs in this area and initiate the necessary R&D of techniques and instruments that will be used for the implementation of additional protocols, including the conduct of complementary access. Techniques based on the detection of emanations associated with nuclear processes are being investigated. These include the sampling and monitoring of specific solid, liquid and gaseous materials to provide new methods and approaches for the detection of undeclared nuclear activities from distances ranging from hundreds of metres to many kilometres. For example, laser-based methods have demonstrated real-time monitoring of atmospheric materials, on-site chemical analysis and the capability of detecting a previous exposure to ionising radiation. On the basis of IAEA priorities and resources, a limited number of projects have been selected and cooperation with Member States has been initiated to take technologies to the maturity level needed for use in-field.

This paper introduces the Novel Technologies Project and describes some of the techniques and instruments currently under consideration by the IAEA for the detection of undeclared nuclear materials and activities.

1. Introduction

The IAEA works to maximize the contribution of nuclear technology to human endeavours, while verifying its peaceful use. The IAEA's mission is addressed by science and technology, mobilizing peaceful applications of nuclear science and technology to developing countries; by safety and security, protecting people and the environment from harmful radiation exposure; and by safeguards and verification, preventing the further spread of nuclear weapons. In the area of safeguards and verification, the IAEA carries out inspection activities that include confirming a State's declared nuclear material (including plutonium and enriched uranium) and maintaining vigilance for evidence of undeclared nuclear material and activities. In exceptional circumstances, the IAEA may also be granted special responsibilities under United Nations Security Council resolutions, allowing it to search for and uncover covert nuclear weapons programmes (e.g. following the 1991 Gulf War), or to conduct ongoing monitoring of disarmament (e.g. monitoring the freeze on reprocessing plutonium under the 1994 framework agreement with the Democratic People's Republic of Korea (DPRK).

In 2004, the IAEA General Conference called upon the Secretariat to examine innovative technological solutions to strengthen the effectiveness and improve the efficiency of IAEA safeguards. Member States also agreed to provide appropriate assistance to facilitate the exchange of equipment, material and scientific and technological information for the implementation of additional protocols. The project Novel Techniques and Instruments for Detection of Undeclared Nuclear Facilities, Material and Activities (known as the Novel Technologies Project) was established in 2005 to identify specific needs and initiate the necessary R&D of techniques and instruments that will be used for the implementation of additional protocols, including the conduct of complementary access.

The IAEA Strategic Objectives for 2006- 2011 [1] include the enhancement of the IAEA's detection capabilities through the development of new or improved safeguards approaches and techniques, and the acquisition of more effective verification equipment. The following goals are applicable to the Novel Technologies Project:

— Improve current detection capability;
— Pursue R&D for the development of novel technologies for detection of undeclared activities;
— Utilize, *inter alia*, Member States Support Programme (MSSP) mechanisms as well as internal resources and expertise; and
— Optimise safeguards equipment and technology.

2. Development and implementation of safeguards methods and instruments

Implementation of effective and efficient safeguards has increasingly relied on the development and deployment of methods and instruments meeting specific functional and technical requirements. Accordingly, equipment development has complemented the safeguards implementation approaches. For example, early safeguards equipment was developed in support of on-site verification of materials and activities at declared locations.

After the 1991 Gulf War and the discovery of a clandestine nuclear weapons programme in Iraq, safeguards approaches were enhanced to include additional methods and techniques, providing the IAEA with further tools by which it could better detect undeclared activities. These included environmental sampling, information analysis, export monitoring, satellite imagery and new technologies such as ground penetrating radar. New technologies were also developed in support of additional protocols activities, including those for complementary access.

By their very nature, clandestine weapons programmes take place at undeclared locations or at declared locations which may be used as a 'cover' for an undeclared process being carried out. The location of such activities requires appropriate equipment that can detect unique characteristics related to the particular activity. The Novel Technologies Project aims to broaden the range of techniques and instruments available to the IAEA, including emerging techniques and instruments that enable the IAEA to detect undeclared activities in undeclared locations (e.g. small industrial areas, universities, workshops).

3. The Novel Technologies Project

In 2005, the IAEA Department of Safeguards solicited suggestions and proposals through its MSSP system. Broad requirements based on safeguards needs were prepared and sent to all MSSPs and other international bodies. Over 60 proposals, covering a wide range of techniques, were received and reviewed by the Safeguards Department. Techniques regarded as 'new'[1] were forwarded to the relevant organizational unit in the IAEA for further consideration. Those regarded as 'novel'[2] methods or instruments addressing a particular safeguards problem were selected for further development and evaluation within the Novel Technologies Project. Interestingly, many were based on emerging laser and other forensic techniques.

3.1. Project tasks

The following proposals, meeting specific safeguards needs for both on-site and away-from-site detection of undeclared activities, have been selected by the IAEA for further development and evaluation. (See Figures 1–12.)

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[1] New technologies are defined as those for which the methodology is already understood and implemented by the IAEA for safeguards applications. Examples include the next generation surveillance and sealing systems.-
[2] Novel technologies are defined as those for which the methodology has not been applied previously by the IAEA for safeguards applications.

3.1.1. Optically stimulated luminescence (OSL)

Need: To determine an undeclared location that has been used previously for storing radiological material.

FIG. 1. An undeclared location is used for the storage of undeclared materials.

FIG. 2. The materials are removed and the location is subsequently 'disguised'.

Proposed Solution: Use OSL to measure the radiation-induced signature retained in many common building materials.

FIG. 3. An IAEA inspector collects samples of the surrounding building materials.

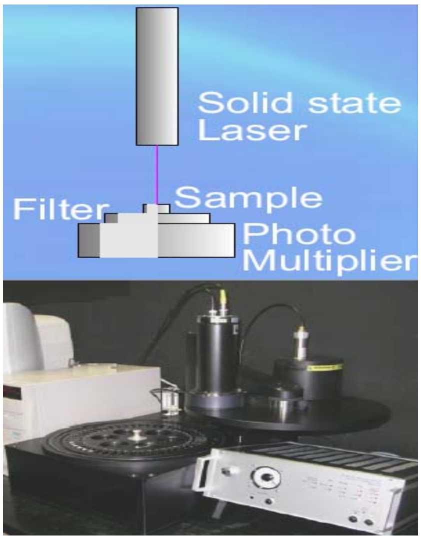

FIG. 4. The collected samples are analyzed for residual nuclear activation, indicating the previous presence of stored nuclear materials.

3.1.2. Laser-induced breakdown spectroscopy (LIBS)

Need: To determine the nature and history of compounds and elements found on-site.

FIG. 5. Unidentified materials found during an on-site complementary access inspection.

Proposed Solution: Use on-site LIBS to determine the nature and history of compounds and elements.

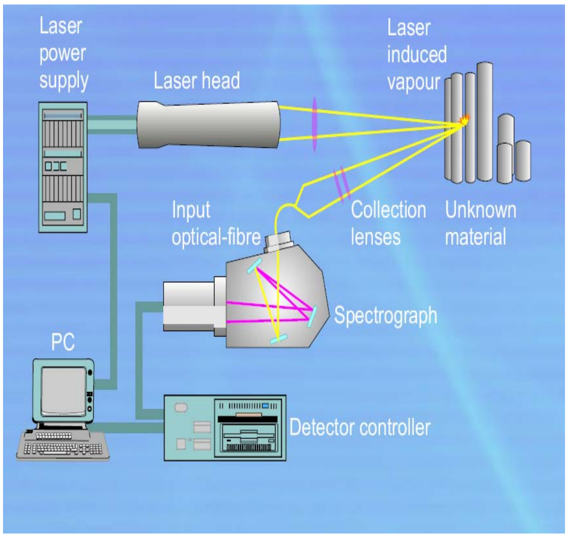

FIG. 6. LIBS comprises (i) a laser system to ablate the material surface to create a micro-plasma, and (ii) a spectrometer to generate a spectroscopic profile of the micro-plasma's constituent components.

FIG. 7. A trained IAEA inspector operates the LIBS unit on-site. The spectroscopic profile is compared to those in its library to determine material's make-up and history.

3.1.3. Light detection and ranging (LIDAR)

Need: To detect the presence and nature of nuclear fuel cycle process activities at suspected locations.

Proposed Solution: Use a mobile LIDAR laboratory in the vicinity of a suspected site to detect the presence of characteristic gaseous compounds, emanating from nuclear fuel cycle (NFC) processes into the atmosphere.

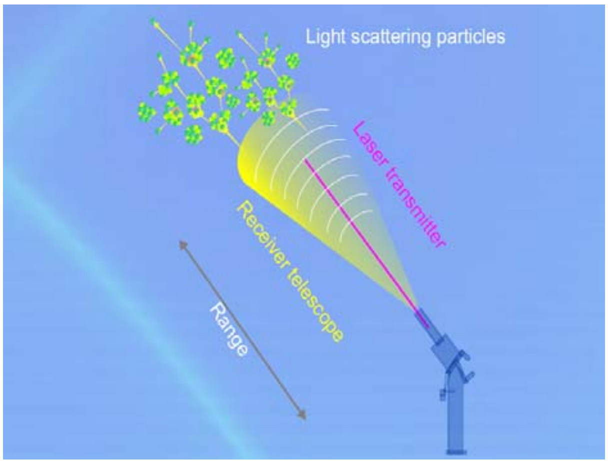

FIG. 8. LIDAR methods are used routinely by environmental monitoring agencies to determine the presence of pollutants in the atmosphere.

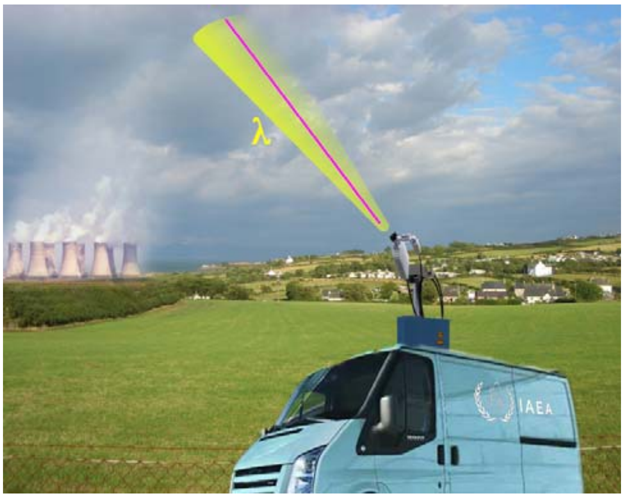

FIG. 9. A mobile LIDAR travels to the vicinity of a suspected location engaged in undeclared NFC processes. A laser, tunable to precise wavelengths (λ) selectively stimulates specific airborne molecules emanating as gaseous compound from the process. A light-sensitive telescope scans the atmosphere, detecting the presence of the stimulated molecules.

3.1.4. Sampling and analysis of atmospheric gases

Need: To detect the presence and nature of nuclear fuel cycle process activities at suspected locations.

Proposed Solution: Use on-site laboratory to determine the atmospheric composition of gaseous mixtures.

FIG. 10. A mobile on-site laboratory samples and concentrates atmospheric-borne pollutants. Local meteorological conditions and the GPS location are also recorded.

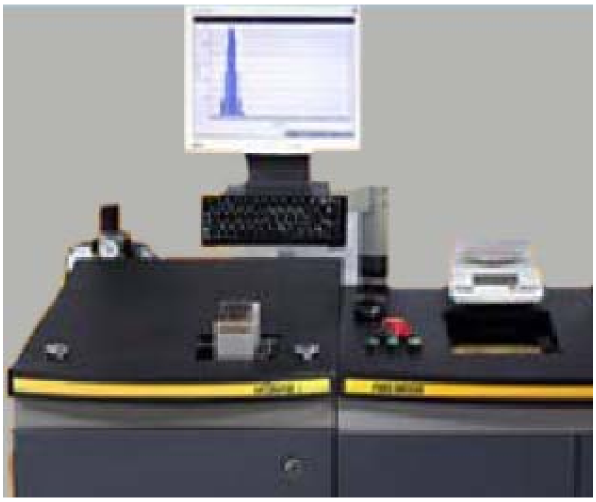

FIG. 11. Samples are brought to a field laboratory for analysis.

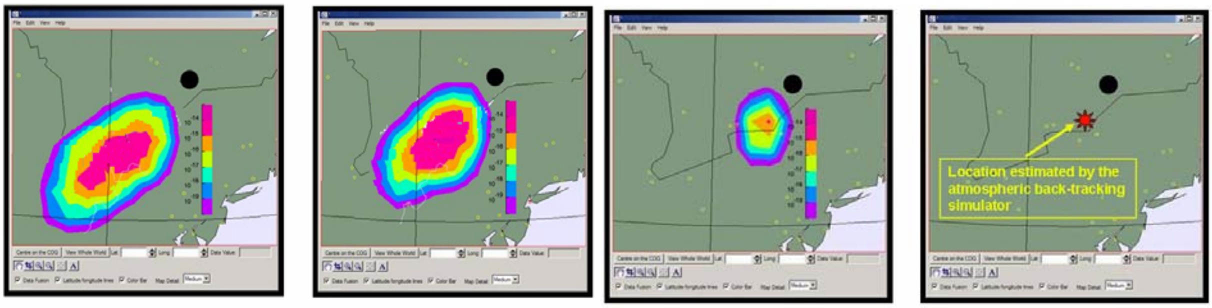

FIG. 12. Airborne material is identified through sample analysis, and the data are combined with meteorological information in a suitable atmospheric computer model to provide an estimate of the source direction and probable location.

3.2. Project activities

In parallel to pursuing the tasks outlined in section 3.1, the Project has also convened specialist technical meetings on techniques for the verification of enrichment activities [2], noble gas sampling and analysis [3] and laser spectrometry techniques [4]. Further specialist meetings covering novel technologies are being planned. Additionally, the Project has been active with the support of Member States in establishing contacts with international R&D organizations and with experts engaged in a wide range of sensor and detection technologies. MSSPs have also been supportive, agreeing to assist the Project by facilitating technical exchanges with both private and government-operated R&D laboratories and by providing access to experts for short-duration tasks, facilitating attendance at technical meetings, advising on novel methods and instruments, conducting field tests and providing supplementary funding.

The Project is also developing a secure technical database to handle relatively large volumes of technical information. The database will also provide non-sensitive information on the Project's tasks and activities on a dedicated website, to further raise the profile of this work to the international R&D community.

3.3. Project planning

The Novel Technologies Project was established to develop and evaluate effective techniques that meet IAEA needs and that can be incorporated within safeguards approaches for detecting evidence of undeclared nuclear fuel-cycle activities, particularly at undeclared locations. To that end, the Project will continue to conduct surveys to identify safeguards needs that cannot be met with available techniques, broaden technical collaboration with other non-proliferation organizations and the international R&D community and, where required, initiate further tasks that will lead to safeguards-useable methods and instruments. The basis of these initiatives will be a review and analysis of the nuclear fuel cycle processes, the identification of the most safeguards-useful activity indicators[3] and emanating signatures[4] that can 'travel' from the source location and be detected with a high level of confidence and accuracy. Indicators and signatures will be information, matter and/or energy associated with a particular NFC process. Once identified, methods useful for the detection of promising indicators and signatures will be assessed by experts to determine if suitable methodology or instruments are available. Where none exists in a safeguards-useable form, then the Project will define appropriate technical and procedural requirements, initiating the necessary R&D and testing regimes.

4. Conclusions

The establishment of the Novel Technologies Project has provided a mechanism for the IAEA to address the technologies required for emerging and future inspectorate needs. Moreover, it has facilitated the IAEA's access to a greatly expanded range of methods and instruments, thereby allowing safeguards planners the opportunity to develop novel verification and detection approaches.

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[3] Indicators are defined as entities that go into making the process operative. Examples are resources, required materials, facility design and related R&D.
[4] Signatures are defined as entities produced by the nuclear fuel cycle process when it is in operation. Examples are produced material, process by products and energy emanations.

Uranium isotopic assay instrument

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Abstract. The isotopic assay instrument under development at Pacific Northwest National Laboratory (PNNL) is capable of rapid prescreening to detect small and rare particles containing high concentrations of uranium in a heterogeneous sample. The isotopic measurement concept is based on laser vaporization of solid samples followed with sensitive isotope specific detection using either uranium atomic fluorescence emission or uranium atomic absorbance. Both isotopes are measured concurrently, following a single ablation laser pulse, using two external-cavity violet diode lasers. The simultaneous measurement of both isotopes enables the correlation of the fluorescence and absorbance signals on a shot-to-shot basis. This measurement approach demonstrated negligible channel crosstalk between isotopes. Rapid sample scanning provides high spatial resolution isotopic fluorescence and absorbance sample imagery of heterogeneous samples. Laser ablation combined with measurements of laser-induced fluorescence (LALIF) and through-plume laser absorbance (LAPLA) was applied to measure gadolinium isotope ratios in solid samples. Gadolinium has excitation wavelengths very close to the transitions of interest in uranium. Gadolinium has seven stable isotopes, and the natural ^{152}Gd:^{160}Gd ratio of 0.009 is in the range of what will be encountered for ^{235}U:^{238}U isotopic ratios. LAPLA measurements were demonstrated clearly using ^{152}Gd (0.2% isotopic abundance) with a good signal-to-noise ratio. The ability to measure gadolinium abundances at this level indicates that measurements of ^{235}U/^{238}U isotopic ratios for natural (0.72%), depleted (0.25%), and low enriched uranium samples will be feasible.

Introduction

The paper describes PNNL's development of an uranium isotope assay instrument specifically designed for detection of undeclared uranium enrichment and reprocessing activities. This investigation capitalizes on newly available violet diode lasers that provide significant flexibility in uranium isotope measurements. With external cavity diode lasers, ^{235}U can be excited at 404.3 nm with fluorescence at 394.4 nm, while ^{238}U can be excited at 415.4 nm with fluorescence at 493.3 nm. The approximately 100-nm difference in fluorescence wavelengths is trivial to spectrally separate using optical bandpass filters. Likewise, for LAPLA measurements, the two-laser absorption channels can be separated easily with optical filters or diffraction gratings. While these lasers are not as small and low-powered as those in CD-ROM drives, they are small enough to be integrated suitably into a compact, field-portable instrument.

The measurement approach and instrument configuration is suitable with a broad range of safeguard verification missions. A variety of deployment options are possible including: field-portable applications, where swipe samples can be collection and analyzed in real-time; wide-area environmental sampling inconjunction with an aerosol particle collection system; and analytical laboratory applications, where the high spatial resolution and rapid scanning features provide large volume sample prescreening.

Isotopically selective excitation of uranium vapor was the basis of the Atomic Vapor Laser Isotope Separation (AVLIS) program at Lawrence Livermore National Laboratory for uranium enrichment. LALIF and LAPLA have been used on a variety of samples that include various steels and glasses [1].

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Reported elemental detection limits typically range between 1 and 100 ppm with 1 to 1000 ng of material ablated and mass detection limits of 0.1 pg for single-shot measurements [2].

Instrument Development

The prototype laboratory assay instrument design, shown in Figure 1, uses a commercial neodymium-doped yttrium aluminum garnet (Nd:YAG) laser source (Continuum Surelite I, Model SL1-20) for sample ablation. The ablation laser has an emission wavelength of 1064 nm, a repetition rate of 10 Hz, and an incident energy of 0.5 mJ/pulse. The ablation laser beam is focused to a 30-µm spot size. Laser ablation was performed in a custom vacuum chamber, which contains the ablation laser entrance port, sample translation system, fluorescence collection optics, and laser absorbance exit ports. Optimum ablation plume conditions were obtained using a cover gas of 99.95% pure argon at approximately 2 torr.

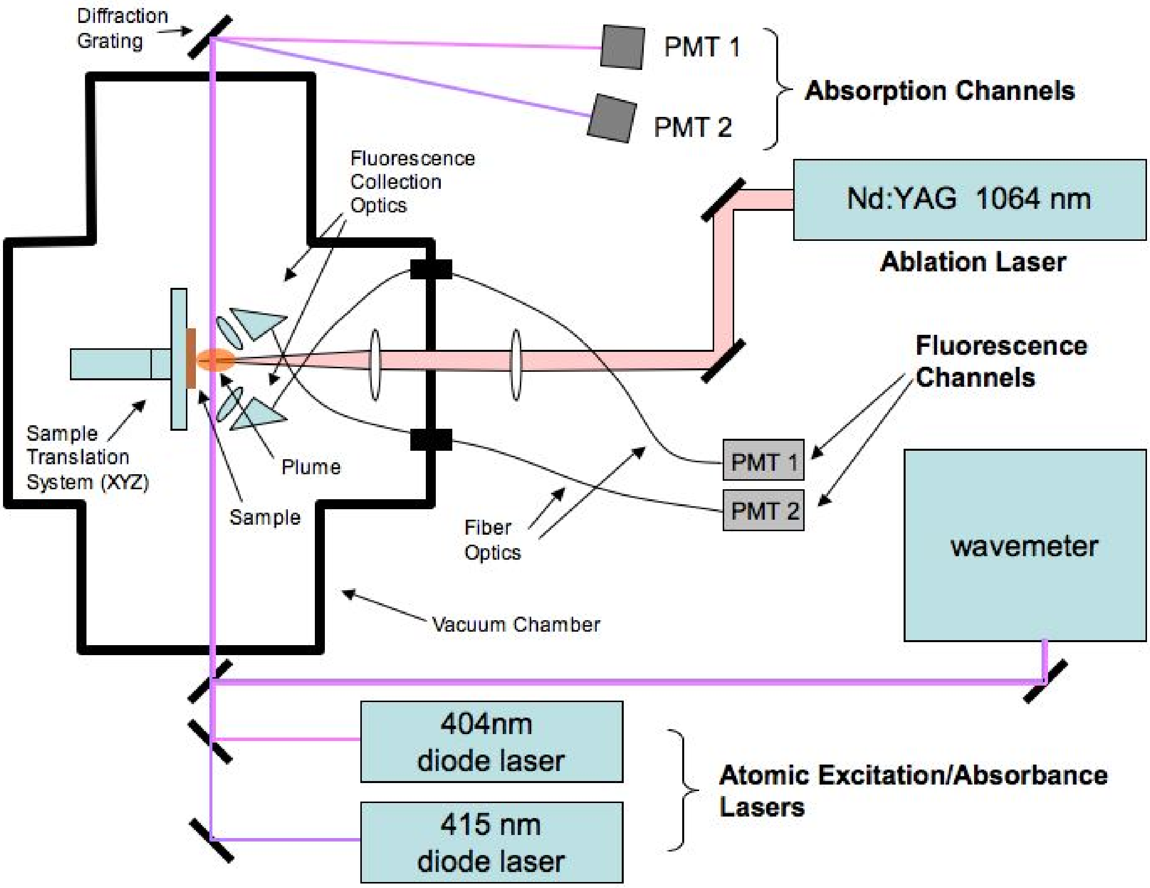

FIG. 1. Schematic diagram of LALIF and LAPLA isotope assay instrument.

Two external-cavity violet diode lasers (Toptica Model DL100-LT-01067) are used to either excite the atomic fluorescence (LALIF) or probe the through-plume absorbance (LAPLA). Laser beam-combining optics combine the tunable wavelengths of the two lasers (~404 nm and ~415 nm, respectively) into one beam outside the ablation chamber, as shown in Figure 2. The combined beam then is injected into the ablation chamber through the laser input port. The wavelength emission from each violet laser is monitored using a Burleigh Model WA-1500 wavemeter.

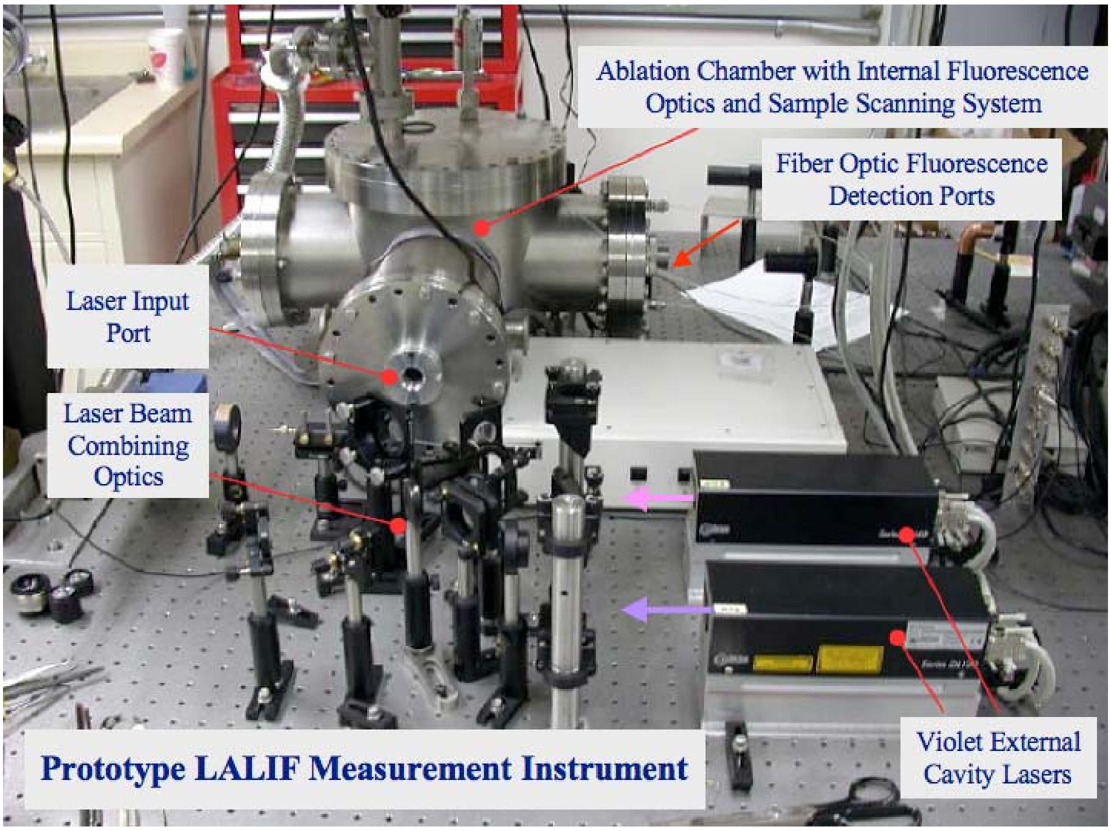

FIG. 2. Prototype Laboratory Assay Instrument showing the two violet external cavity lasers, beam combining optics, and ablation chamber.

Two fluorescence detection channels, one for each isotope, are used to measure the LIF signals. Inside the vacuum chamber, two sets of collection lenses are positioned above the sample. These lenses collect the LIF signals from the laser ablation plume. Each fluorescence channel emission is refocused into an optical fiber that passes through the vacuum chamber to the external fluorescence detectors. The detector modules consist of a laser line optical rejection filter, an isotope bandpass filter, and a PMT detector. Small monochromators also used during testing enabled rapid switching between different LIF lines and evaluation of background ablation plasma emissions.

Directing the coaxial violet laser beams through the ablation plume makes the laser absorption measurements possible. The laser light that passes through the plume exits the vacuum chamber through a window port and then is wavelength-split using a diffraction grating. Each laser line is then directed to an absorption detector, as shown in Figure 3.

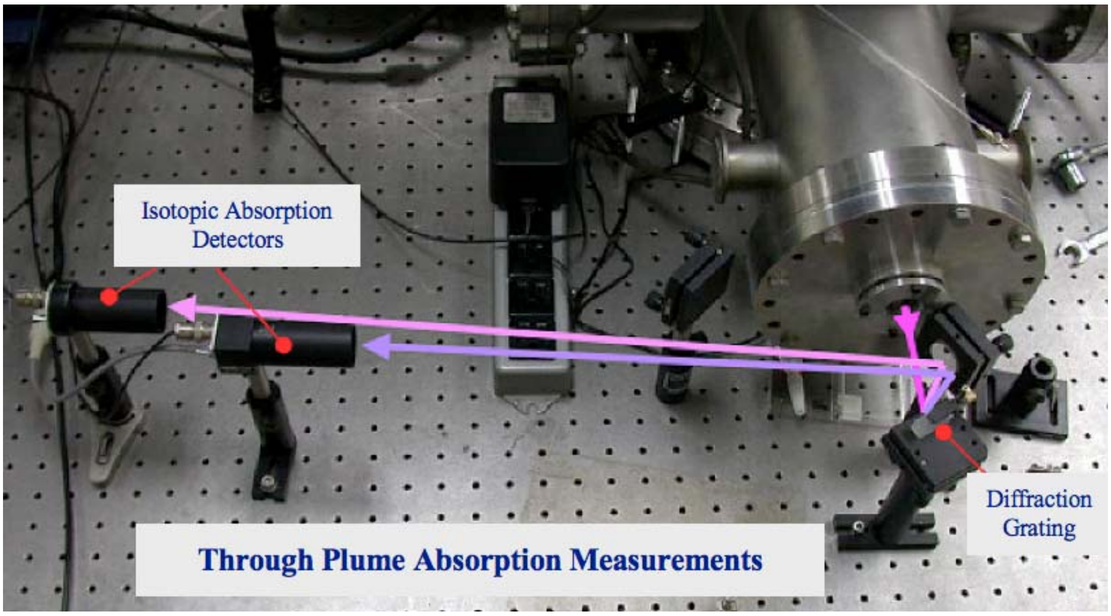

FIG. 3. LAPLA measurement arrangement.

The measurement process is controlled using a combination data acquisition and instrument control hardware and custom LabVIEW software from National Instruments Corporation. This hardware and software combination controls the ablation laser trigger, sample scanning, fluorescence or absorbance signal acquisition, and isotopic signal processing and display. To wavelength-stabilize the violet diode lasers, the Burleigh wavemeter is monitored using the LabVIEW software, which, in turn, generates a proportional wavelength-tuning signal that is fed back into the diode laser controller.

Sample Characterization

An important feature of the external-cavity diode lasers for fluorescence excitation is that transition lines for other elements can be excited. This feature makes possible development and testing without the use of radioactive materials and their corresponding contamination issues. Gadolinium has excitation wavelengths very close to the transitions of interest in uranium. Table 1 shows the spectral lines of neutral gadolinium and uranium, whose lower energy (E_{low}) levels are either in the ground or first excited states. Gadolinium has seven stable isotopes, and the natural ^{152}Gd:^{160}Gd ratio of 0.009 is in the range of what will be encountered for ^{235}U:^{238}U isotopic ratios. The isotope shifts for the ^{152}Gd:^{160}Gd pair are approximately 10 times the room-temperature Doppler widths of 0.72 GHz. In contrast, the uranium isotope shift is 17 to 25 times the room-temperature Doppler width for the transitions of interest.

Table 1. Transition Pairs for Simulating LALIF and LAPLA Measurements of Gadolinium and Uranium[(a)].

Wavelength (nm in air)	E_{low} (cm^{-1})	E_{upper} (cm^{-1})	A_{ij} (s^{-1})	Gamma (s^{-1})
Gadolinium I (neutral Gd atom, 7 stable isotopes)				
403.5389	215.124	24988.884	4.78E6	
416.7270	999.121	24988.884	6.57E6	
405.4722	0	24655.639	4.26E7	
409.0412	215.124	24655.639	2.34E7	
405.8220	215.124	24849.514	6.15E7	7.81E7
419.1622	999.121	24849.514	1.30E7	7.81E7
404.5009	0	24714.841	4.00E6	7.41E7
408.0528	215.124	24714.841	5.75E6	7.41E7
413.4164	532.977	24714.841	2.65E7	7.41E7
Uranium I (neutral U atom)				
404.2750	620.323	25348.977	5E7	
394.3816	0	25348.977	6E7	
404.7611	620.323	25319.274	2.4E7	
394.8442	0	25319.274	1.7E7	
415.3971	0	24066.566	1.17E7	
493.3060	3800.829	24066.566	1.0E6	

(a) Source: Morton (2003, Tables 5 of Paper II and Paper III). E_{low} and E_{upper} are, respectively, the energies of the upper and lower states; A_{ij} is the radiative rate from the upper to the lower state, and gamma is the total radiative rate out of the upper state.

Using 405.8-nm laser excitation, the first characterization measurement was performed on a gadolinium metal foil sample. The results of these measurements are shown in Figure 4. The upper plot shows the through-plume absorbance (black and blue) and LIF signals (red) for gadolinium isotopes. The lower trace shows monitored excitation laser power during the wavelength scan used to collect the isotopic assay. Continuity of the signal is used to check for continuous single-mode operation over the more than 1-cm^{-1} scan. The blue absorbance trace at 100x magnification clearly shows the 0.2% ^{152}Gd isotope with good signal-to-noise ratio. The ability to measure abundances at this level supports measurements of natural, depleted, and slightly enriched uranium. Currently the LIF configuration is not producing comparable signal levels, but its optimization has the potential to produce similar results.

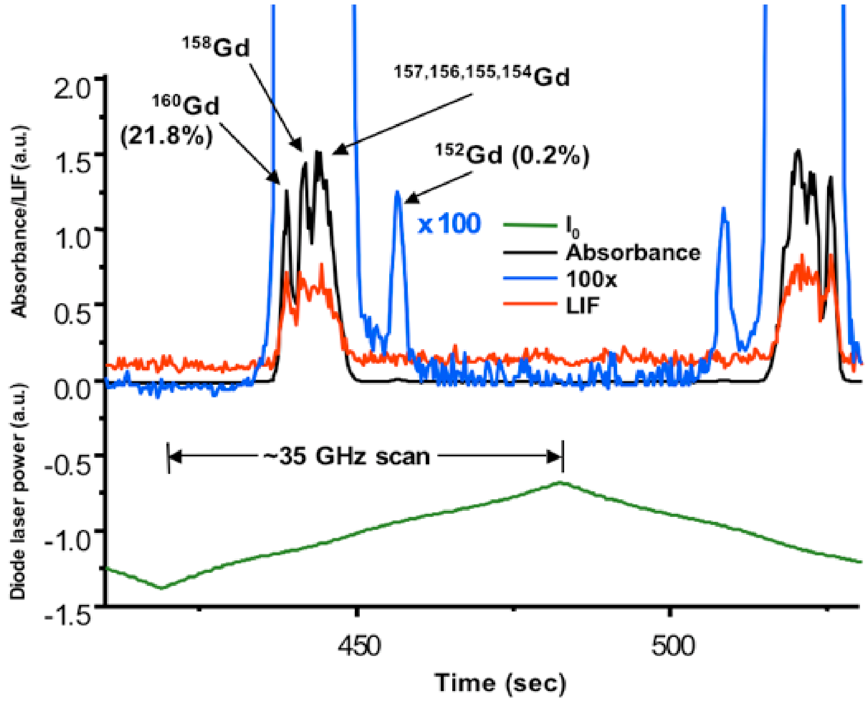

FIG. 4. Gadolinium Isotope Measurements Using LALIF and LAPLA. The Y axis (Diode laser power) is proportional to the emission wavelength. Both the upper and lower plots share the X axis (Time).

To evaluate the LAPLA detection sensitivity, gadolinium samples were prepared by dissolution in nitric acid. 10-μl liquid drops of 0.86 mg/ml gadolinium were applied to a titanium foil and then allowed to dry. The sample was then raster scanned with LAPLA measurements performed point-by-point using the ^{160}Gd isotope (Figure 5). These measurements yield a 20 mm square atomic absorption image of ^{160}Gd, with the absorbance strength color encoded as shown in the legend to the right of the chart. The single-shot detection limit (ssdl), with signal-to-noise ratio of 5, is estimated from

$$ssdl = 5 \times a_{160} \times c_l \times v_l \times \left(d_L / d_d\right)^2 \times \left(\sigma_b / A\right) \approx 2 \times 10^{-11} g$$

where a_{160} = 0.218 is the ^{160}Gd isotopic abundance, c_l and v_l are the concentration and volume of the liquid drop (0.86 mg/ml, 0.01 ml), and d_L/d_d is the ratio of focused ablation laser diameter (~30 μm) to the dried spot diameter (~3 mm)—the square gives the fraction of the drop sampled by each ablation laser shot. (σ_b/A) is the ratio of background noise (~2 × 10^{-4} absorbance, evaluated over the "no-spot" background areas), to the (average) "on-spot" absorbance (~0.01). If this roughly 20-pg detection limit represents the minor isotope, then the expected isotopic abundances of about 100:1 (for either ^{160}Gd:^{152}Gd or ^{238}U:^{235}U) dictate that a few nanograms of the major isotope need to be in the laser-sampled area for single-shot isotope ratio determinations. Thus, we expect real isotope ratio measurements can be performed for single particles of about 10 μm in diameter.

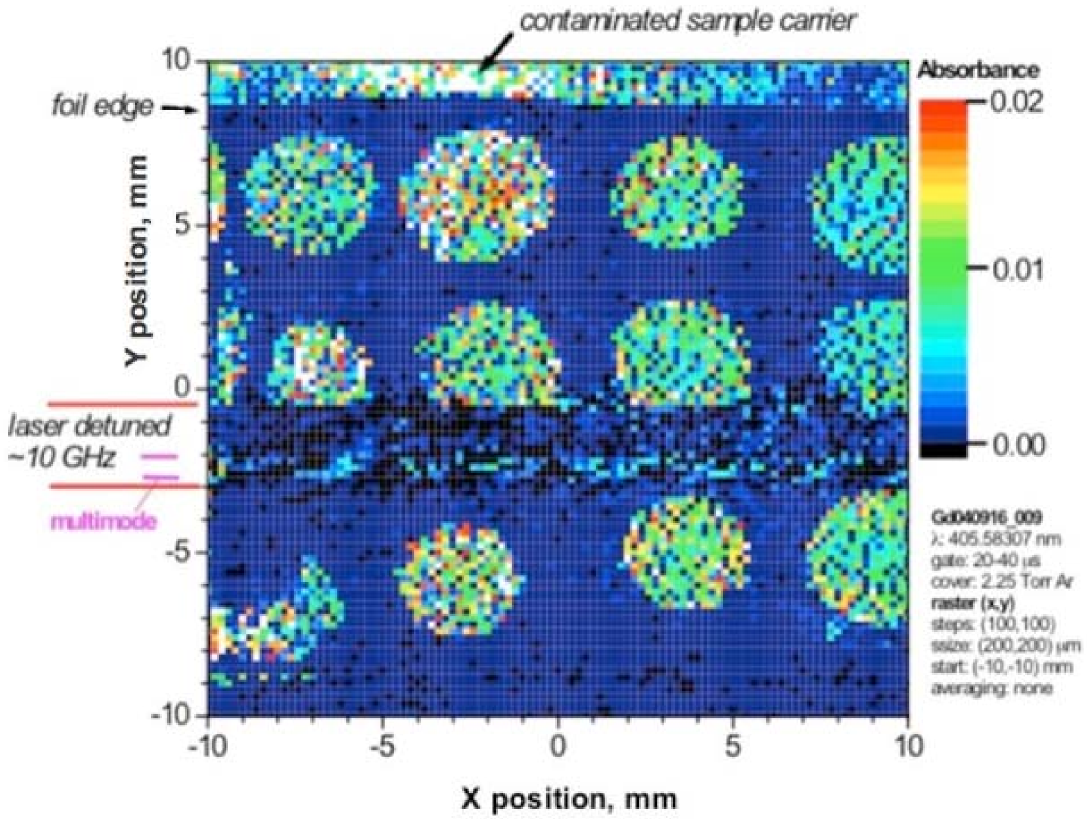

FIG. 5. LAPLA Measurements of a Gadolinium Sample Prepared by Dissolution in Nitric Acid. Here, LAPLA signal is lost when the excitation laser is detuned from the atomic transition resonance. No effort was made to normalize the pulse-to-pulse ablation laser energy accounting for absorbance variations observed in the gadolinium droplets.

The next gadolinium foil measurements demonstrate the LAPLA isotopic ratio measurement potential, as shown in Figure 6. The upper left plot shows the 405.8-nm absorbance image of the ^{152}Gd minor isotope, and the lower left plot shows the 413.4-nm absorbance image of the ^{160}Gd major isotope. The regions labeled B and C were acquired with the 405.8-nm absorbance laser detuned +1 and -1 GHz from the ^{152}Gd line center, respectively. As expected, only a weak wing absorbance signal is present when the lasers are detuned from resonance. The region A was acquired with the 413.4-nm laser blocked, confirming that large changes in the major isotope signal do not influence the minor isotope channel. The upper right plot shows the ^{152}Gd/^{160}Gd absorbance ratio. Note that in region A, the ^{160}Gd absorbance is undefined ($I_0 = I_t = 0$), and an artifact of the signal processing results in a noisy signal about zero. In the ratio plot, these undefined points, as well as ones with a ^{160}Gd exact value of zero, are plotted in grey. The areas without gadolinium have ^{160}Gd absorbance values scattered about zero and cause the point ratio to be either underscale (black) or overscale (white).

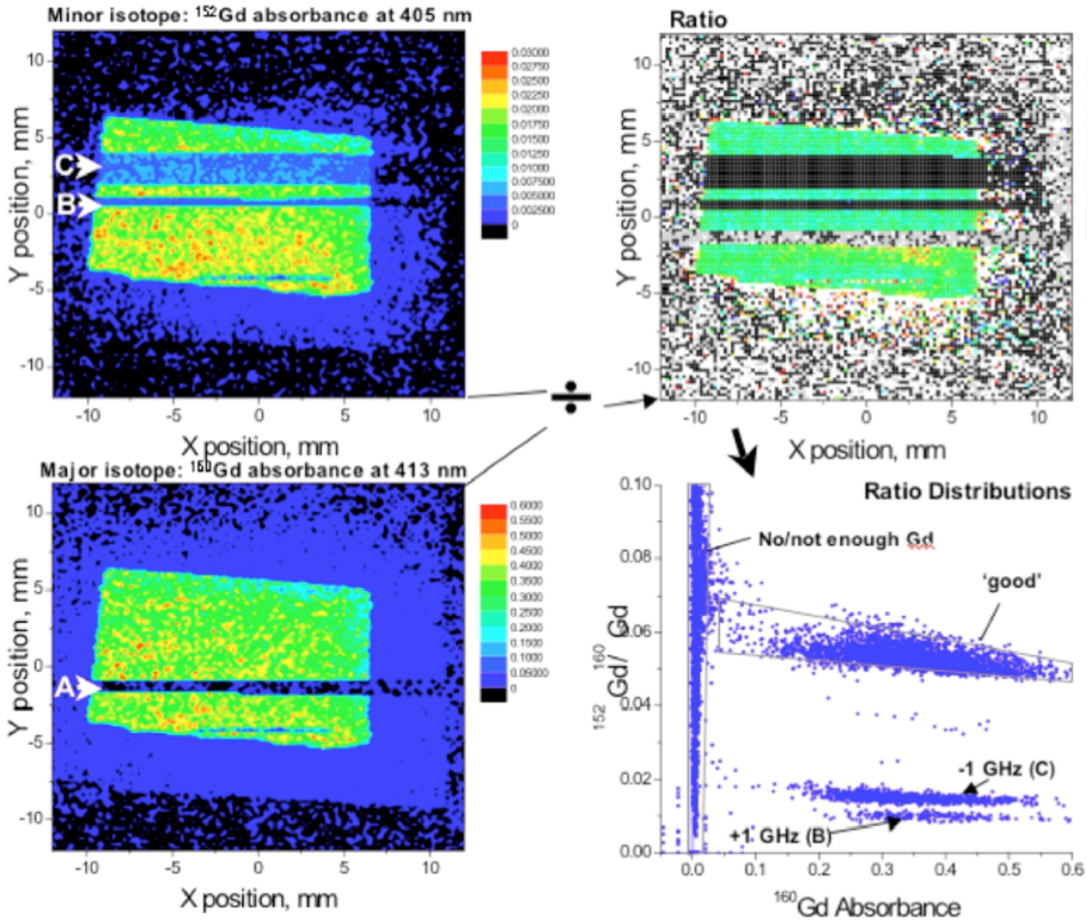

FIG. 6. LAPLA Measurements of ^{152}Gd and ^{160}Gd. Note that 1) blocking the major isotope laser (marked A) has no crosstalk into the corresponding minor isotope measurement; 2) the isotope ratio is not corrected for the relative strengths of the two transitions (see text); 3) in the ratio distributions, it is suspected that the slight downward slope is caused by differential hole-burning in the optically thick ^{160}Gd absorption, but this needs to be checked experimentally; and 4) no effort was made to normalize the pulse to pulse ablation laser energy. This accounts for some of the observed absorbance variations in the gadolinium foil region.

The lower left plot shows the ratio distributions. The vertical distribution near zero for the ^{160}Gd major isotope represents ratio results outside the gadolinium foil region. The detuned distributions of +1 GHz and -1 GHz represent measurements without a transition resonance for the ^{152}Gd minor isotope. The distribution labeled "good" represents valid isotope ratio values. The wide horizontal distribution (^{160}Gd absorbance) reflects the variation in ablation yield with pulse energy and sample morphology. The much smaller vertical distribution (isotope ratio) reflects the ability of the dual–wavelength, simultaneous absorption measurements to correct these variations on a shot-by-shot basis. Although the ^{160}Gd yields for the "good" data vary by more than a factor of 10, the resulting ratios are reproducible with a relative standard deviation of 5%.

Note that the plotted ratios of approximately 0.06 are simply the ratio of absorbances for the two isotopes in the different transitions. Correction to the actual isotopic abundance ratio of 0.009 still requires normalization for the relative strength of the two transitions as well as the statistical and thermal populations of the different starting states, which may be calibrated experimentally simply by measuring ^{160}Gd in both transitions. The use of two different transitions also provides dynamic range expansion by using the stronger transition for the rarer isotope.

Conclusions

Diode laser LALIF and LAPLA spectrometry was applied to measure gadolinium and gadolinium isotope ratios in solid samples by laser ablation. The prototype instrument is designed to provide simultaneous LALIF and LAPLA measurements. Both isotopes are measured concurrently using two external-cavity violet diode lasers. The simultaneous measurement of both isotopes will enable the correlation of the fluorescence and absorbance signals on a shot-to-shot basis and thus may lead to improvement of precision and accuracy for this technique. Single-shot detection sensitivity approaching the femtogram range and isotopic ratios with relative precision less than 10% have been demonstrated with measurements on surrogate materials. These results demonstrate the feasibility of making useful isotopic ratio measurements on uranium. The measurement approach and instrument configuration is suitable for in-field swipe sample analysis or analytical laboratory prescreening for the detection of undeclared enrichment and reprocessing activities.

ACKNOWLEDGEMENTS

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Development of an atmospheric ^{85}Kr automated sampler and analyzer

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Abstract. One of the potential interests of atmospheric ^{85}Kr measurement is the detection of clandestine reprocessing activities as 10-35 TBq of ^{85}Kr are emitted per kg of reprocessed Pu. To this goal, a system for Krypton sampling, purification, concentration and ^{85}Kr measurement is currently under development. It has been designed to be fully automated, to perform Krypton trapping at room temperature, and to obtain at least 0.5 cm^3 of Krypton for a 6-hour operating cycle. The prototype system is composed of four distinct stages that perform successively the following steps: i) air sampling, first purification and pre-concentration; ii) purification and concentration; iii) further concentration; iv) Krypton volume and ^{85}Kr activity measurements. Thanks to a permeation membrane, the first-step purifies air from O_2, H_2O, CO_2, and pre-concentrates Krypton by a factor of three. Air equivalent processed volume in 6 hours is about 0.6 m^3. The second step consists in Krypton purification by room temperature trapping - high temperature (around 200°C) desorption on high specific area active charcoal beds in three columns. Choice of the active charcoal is of utmost importance. The further concentration stage allows reduction of the elution gas volume thanks to three in-line activated carbon columns. Final transfer into measuring cell is carried out by hot desorption of the accumulated Krypton for each 6-hour cycle. For the activity measurement, we modified a commercial proportional counter to adapt it to low-level ^{85}Kr measurement. Counting efficiency is close to 70% and background is about 5 counts·minute^{-1}. We tested the automated process, without the proportional counter, for many operating cycles. First tests show that we collect about 0.7 cm^3 of stable Krypton (STP conditions). The whole system will be able to detect moderate ^{85}Kr activity variations, typically a few tenth of Bq·m^{-3}, above 1.4 Bq·m^{-3} background.

Foreword: Since a few years, the IAEA has become aware of growing challenges, including the cover acquisition and clandestine operation of sensitive nuclear fuel cycle technologies able to produce nuclear material necessary to manufacture nuclear weapons, among which reprocessing techniques. This awareness has been fuelled by the announcement by North Korea that it has reprocessed spent fuel previously under AIEA monitoring and the chance of new reprocessing campaigns to occur. Iran is also developing troublesome Plutonium production capabilities. The measurement of the atmospheric concentration of ^{85}Kr could improve the IAEA capabilities to detect undeclared reprocessing activities. This Technique proposed by France on the base of its CTBT noble gas monitoring experience, has been selected as one of the seven novel techniques and instruments for detection of undeclared nuclear facilities, material and facilities to be developed in the framework of the project SGTS SG-08[1]. To carry out the research and development on prototype equipment, a task has been proposed to France (06/TDO-005) which is still in reviewed[2].

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[1] Research and Development Programme for 2006-2007/IAEA Department of Safeguards/January 2006
[2] Summary of Decisions and Agreed Action Resulting from the French Support Programme Annual review Meeting,, Vienna, 19 June 2006.

1. Introduction

Krypton-85 (τ½ = 10.76 yrs) is a fission product that exists in nuclear fuel for years after the fuel has cooled. The specific content of ^{85}Kr per ton of fuel varied between 50 and 340 TBq depending on the type of reactor and the operational history of the fuel rods [1]. In a similar way, the content of Plutonium in a fuel rod may vary by about an order of magnitude. In the course of the production of "weapon grade" Pu (≤ 7% ^{240}Pu), 10 to 35 TBq of ^{85}Kr are released per kg of Plutonium [1]. Over the last several years, interest has arisen in the use of ^{85}Kr as a tool for detecting signs of clandestine reprocessing [1-4].

However, ^{85}Kr atmospheric background is relatively high. According to Steinkopff [5], the mean ^{85}Kr activity for weekly samples collected between 2001 and 2003 amounts to 1.5±0.5 Bq·m^{-3} for the Offenbach site (Germany), and 1.6±0.6 Bq·m^{-3} for the Freiburg site (Germany). Moreover, it exhibits strong variations [5-7] that are mainly due to a combination of releases from reprocessing facilities and atmospheric dilution. Although sample variability is high, the maximal value reached "only" 3.5 Bq·m^{-3} (in September 2001), and only a few weekly samples were between 2 and 3 Bq m^{-3} [5]. Thus, even in these locations, influenced by releases from La Hague, France, and Sellafield, UK, weekly detection of ^{85}Kr at levels above 5 Bq·m^{-3} may potentially be detected which give indication on the capability to detected clandestine activity.

Moreover, the amplitude of these variations strongly depends on the geographical area and on distance from the main reprocessing facilities. Hundred of kilometres away from these facilities (for instance in the Southern Hemisphere), available measurements show that relative variations are relatively low (a few %) and close to the uncertainties of the measurements [6,7]. Then, in these areas, detection of moderate increases of the ^{85}Kr activity (a few tenth of Bq·m^{-3} above background) may be an indication of a clandestine reprocessing.

In the past decades, some laboratories developed systems for low-level atmospheric ^{85}Kr measurements especially for purposes of environmental survey of atmospheric radioactivity and global modeling of atmospheric transport [6-10]. However, these systems are manual, and used liquid Nitrogen to increase the efficiency of Krypton trapping on solid adsorbents. Therefore, such systems require presence of skilful staff and a well-equipped laboratory. Generally speaking, techniques for collecting and separating ^{85}Kr from the atmosphere are similar, however, to those described for radioactive Xenon isotopes, but are less efficient for Krypton than for Xenon. Unfortunately, no commercial system exists to collect and analyze ^{85}Kr at environmental concentrations.

The aim of this work is to develop an automated system for Krypton sampling, concentration, purification, and for ^{85}Kr measurement. For this work, we benefit from experience gained during development of the Spalax™ Noble Gas Equipment for CTBT implementation [11]. Indeed, some features of this equipment are similar to those of the "Spalax™" equipment.

This system was designed according to the following technical specifications: i) fully automated sampling, treatment (concentration and purification), and measurement; ii) trapping at ambient temperature (no cryogenic cooling); iii) a 6-hour duty cycle; iv) ^{85}Kr measurement thanks to a proportional counter; v) obtention of at least 0.5 cm^3 of Krypton. This volume theoretically provides a detection capability of a few tenth of Bq·m^{-3} of ^{85}Kr over background, assuming given performance of the proportional counter (efficiency, background) and given precision over Krypton volume determination.

2. Description of the equipment

The prototype system is composed of four distinct stages that perform successively the following steps: i) sampling, first purification and pre-concentration; ii) purification from radon and concentration; iii) further concentration; iv) ^{85}Kr activity and of Krypton volume measurements. We give a diagram of the whole system in the Figure 1 below. Homemade software based on National Instrument acquisition card allows to control and command the whole system, and to register useful data.

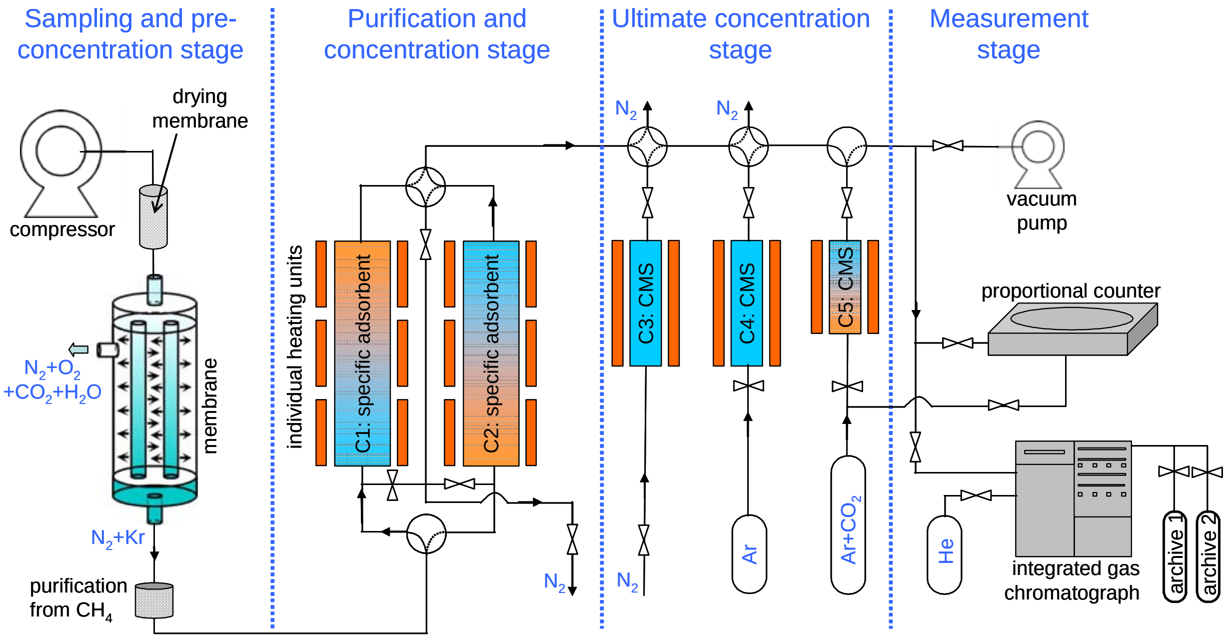

FIG. 1. Diagram of the atmospheric ^{85}Kr automated sampler and analyzer.

2.1. Sampling, first purification and pre-concentration stage

This first stage must achieve three simultaneous goals. The first one is to sample ambient air with a high flow rate. The second is to produce a gas mixture convenient for adsorbent like activated charcoal, i.e. to eliminate most of O_2 and CO_2 that are poisons for the adsorbent involved in the next step of the process. The third one is to pre-concentrate Krypton. We achieved all these goals by using a permeation membrane originally designed for pure Nitrogen production. We tested several commercially available membranes were tested for their capacity to enrich Nitrogen with Krypton. The most efficient one provides an enrichment factor of 2.9 i.e. the Krypton concentration, measured at the Nitrogen outlet, is close to 3.3 ppm instead of 1.14 ppm in ambient air [12]. In addition to purification and pre-concentration, this arrangement present also the advantages to be a passive (no power supply required), continuous (does not require any regeneration time) and robust device.

It is necessary to compress the sampled air under about 8 bars in order to obtain optimum performance of the membrane. Compressed air is then dried by means of a drying membrane (-40° dew point). The flow rate at the inlet of the Nitrogen generator is about 4 $m^3 \cdot h^{-1}$. In order to obtain optimum Krypton enrichment, the Nitrogen flow rate, at the outlet of the Nitrogen generator, is restricted to about 100 $L \cdot h^{-1}$. Finally, the gas mixture (mainly dry Nitrogen enriched with Krypton) is oxidized on catalytic Platinum (400°C) in order to remove traces of Methane. Therefore, air equivalent processed volume in 6 hours is about 0.6 m^3. Therefore, the global Krypton recovery yield η is relatively low at optimum enrichment value. η is given by:

$$\eta = 100 \cdot E \cdot \frac{\phi_{output}}{\phi_{input}}$$

Where E is the Krypton enrichment factor (2.9) and ϕ_{input} and ϕ_{output} are respectively the input and output flux (respectively 4 $m^3 \cdot h^{-1}$ and 0.10 $m^3 \cdot h^{-1}$). Thus, we obtain: $\eta \approx 7.25\%$.

2.2. *Purification and concentration stage*

The second stage of the process is dedicated to Krypton trapping and concentrating and Radon removal. Room temperature continuous adsorption of Krypton is achieved by using a dual-bed system of 1m long, 30mm diameter thick Copper column (named C1, C2). Both are divided in three sub-columns (named C11, C12, C13 for the first column and C22, C21, C23 for the second column). Each of them is filled with 100 grams of "molecular sieve" carbon and located in a tubular oven. The breakthrough time of each 1m-column is fitted to be close to 1 hour, i.e. the time the oven spends to cool down to ambient temperature

Each C1 sub-column is then regenerated successively by raising the temperature up to 300°C and flushed by means of a moderate Nitrogen flux. For example, Krypton and Nitrogen mixture delivered downstream from the sub-column C11 is added (or adsorbed again) on C12 column and so on. Ultimately, the whole Krypton retained in the 1m-column is adsorbed on its last third i.e. C13 sub-column. That desorption procedure allows to extract all Krypton by means of one third of the Nitrogen volume needed to extract Krypton at one go from the 1m-column.

Radon is a naturally occurring radioactive gas in air whose excessive presence in the counter leads to a severe increase of background. As Radon desorption is more difficult and slower than Krypton desorption, most of the Radon remains in the "molecular sieve" carbon at the level of the two first heating units whereas Krypton is at the level of the third heating unit. Thus, this arrangement allows an efficient Radon / Krypton separation. To remove Krypton from the adsorbents, the system uses pure Nitrogen produced by the process itself. However, considering the poor adsorption capacity of most of the current adsorbents towards Krypton, we performed an adsorbent screening to choose the best one for Krypton at ambient temperature. We selected the adsorbent for its ability to trap Krypton at ambient temperature (20°C) and to provide the best resolution with respect to Nitrogen at ambient temperature [12].

2.3. *Ultimate concentration stage*

This stage achieves the ultimate Krypton concentration prior to gas introduction into the measurement stage. It consists in three in-line columns (named C3, C4 and C5), inserted in small-size furnaces. C3 is similar to C1 and C2, but its thickness is reduced. C4 and C5 are made of ¼" stainless steel tubing. These columns are filled with "molecular sieve" carbon. C3 elution and C4 adsorption are performed with pure Nitrogen, whereas C4 elution and C5 adsorption are performed with Argon. This allows to decrease as much as possible the Nitrogen concentration in the final mixture, whose composition must be compatible with the functioning of the proportional counter, i.e. Nitrogen content as low as possible. These two elution/adsorption steps allow an additional reduction of the elution gas volume by a factor of about 50. Final transfer from C5 into measurement cell is carried out without use of elution gas. The heated C5 column is directly connected to the air evacuated β-measurement cell. Desorbed gases expanded naturally into it. Then, C5 column is flushed with Ar+CO_2 (90% Ar – 10% CO_2) to transfer all Krypton in the measurement cell. First tests show that about 0.7 cm^3 of stable Krypton is collected (STP conditions) after thermal desorption of the last column. As the total volume of the measurement cell of the counter is 245 cm^3, the Krypton final concentration is about 0.3%. Therefore, the Krypton concentration factor for the whole process is about 2,500.

2.4. *Measurement stage*

Some authors report the use of liquid scintillation counting [13] and gamma spectrometry [14] for ^{85}Kr activity measurement. However, gamma spectrometry is not sensitive enough for low-level ^{85}Kr measurements and it will be a hard task to integrate liquid scintillation counting into an automated system. Proportional counting has the potential for low-level measurements of β-emitters in solid samples and many laboratories use them routinely. Nevertheless, we do not find any commercial "off-the-shelf" proportional counter that fits our needs for activity measurements of gas isotopes. Therefore, in close cooperation with a manufacturer (Canberra, Loches, France) we alter a "Mini 20" commercial proportional counter to adapt it to low-level measurement of ^{85}Kr. The principle of this measurement is to add the Kr produced by the previous stages of the equipment to the gas normally used to fill the counter (see Figure 2 below). Thus, gas mixture obtained after desorption of the C5 column (mainly Ar, N_2 and Kr) is mixed with a larger volume of 90% Ar + 10% CO_2. Then, this gas mixture is introduced into the counter. The counter operates in the static mode (no gas circulation) under a pressure of 1020-1060 mbars. The total volume of the chamber is 245 cm^3. The usual Mylar window supporting the solid state source is replaced by a Copper plate with O-ring seals to ensure airtightness. We added Copper blocks in which airtightness connections with O-ring seals can be properly fitted. Extremities of the anode wire are embedded into cavities filled with resin. Guard counters are located above and below the proportional counter containing the sample. The guard counters and the main counter are located inside a lead shield. The whole system is equipped with anticoincidence electronics. Approximate proportions of the different gases that constitute the final mixture are 87.3% Ar, 9.7% CO_2, 3% N_2, 0.3% Kr, and small amounts of O_2. We optimized these proportions with respect to efficiency and background. Background is about 5 counts minute^{-1}. Counting efficiency of this device with the aforementioned gas mixture is about 70%. Before starting, the counter is vented with a 90% Ar -10% CO_2 mixture, to eliminate any memory effect. At present, the altered "Mini20" counter suffers from a problem of leakage, which prevents from obtaining a stable efficiency for 6 hour - counting times, and is not yet available for routine measurements.

Krypton volume measurement is performed by a gas chromatograph ("Peri-1200", Perichrom, Saulx-le-Chartreux, France) integrated into the system. These measurements are not hindered by the presence of CH_4 that was eliminated before passing through the permeation membrane. Calibration of the gas chromatograph is performed thanks to three successive measurements of a gas standard with certified Kr concentration.

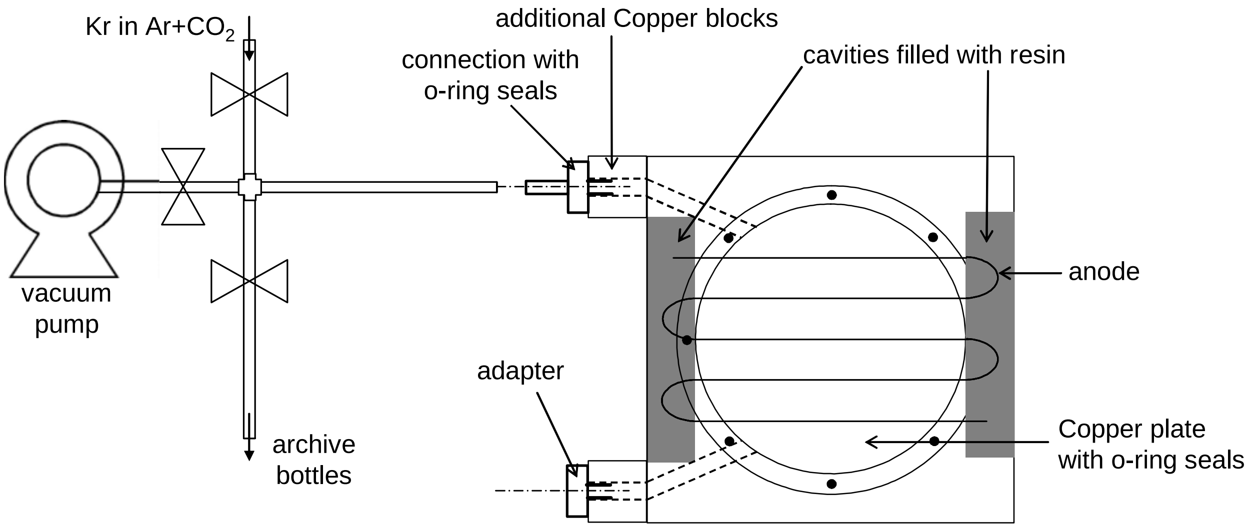

FIG. 2. Diagram of the modified proportional counter. The Mylar film supporting the solid state source is replaced by a Copper plate with O-ring seals to ensure airtightness. The proportional counter is filled with the Kr and Ar+CO_2 mixture and operated in the static mode (no gas circulation).

3. System performance features

We tested the whole system, except the proportional counter, for 24 6-hour cycles. Some of these cycles (number 8 to 13) were done continuously to test connection of cycles. Four samples were sent to the Bundesamt für Strahlenschutz (BfS, Freiburg-im-B., Germany) for Kr concentration and ^{85}Kr activity measurements. We gathered all the results in the Figure 3 below. Concerning Krypton volume measurements, our results are in relatively good accordance with BfS measurements, considering that the BfS results are lowest values because only the minimum volume of gas lost during the transfer from the CEA archive bottle to the BfS measurement device is known. Thus, BfS volume measurements may be slightly underestimated. Mean Kr volume obtained for 6-hour cycles is 0.72 cm^3.

The four activity measurements performed by BfS show moderate variations: from 1.57 to 1.8 $Bq{\cdot}m^{-3}$ air STP. We have no idea of short-term (6 hours) variations of the ^{85}Kr background in Western Europe as measurements carried out by other laboratories and described in the literature are always weekly or monthly averaged measurements [6-7].

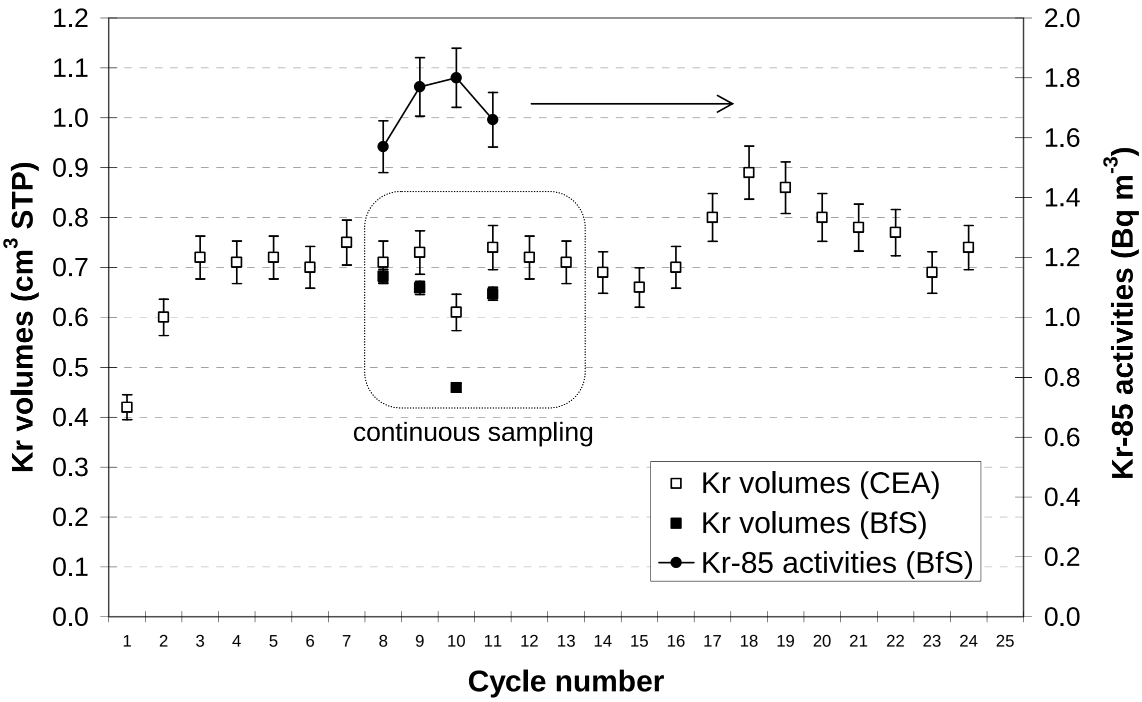

FIG. 3. Krypton volumes (left axis) and ^{85}Kr activities (right axis) measured for 6-hour cycles of the whole equipment, except proportional counter. Uncertainties are given with a coverage factor of two.

4. Further tests and developments

Work is under way to complete the development of the proportional counter and to integrate it into the prototype system. Above all, we must solve airtightness problems that lead to slow decrease with time of the counting efficiency. Next, we have to validate the proportional counter for low-level ^{85}Kr measurements and integrate the counter to the sampling, purification, and concentration process. In particular, we must adapt electronics and software to start and stop automatically the counting when needed.

Afterwards, the further step will be to operate the integrated automated prototype system, including the proportional counter, at a fixed place for 6-hour time resolution ^{85}Kr continuous monitoring, over a long time (at least several weeks). We will conduct this experiment in combination with atmospheric transfer calculations. On the one hand, these measurements will hopefully provide a real improvement in the knowledge of the 6-hour resolution time variations of the ^{85}Kr activities near Paris, France. On the other hand, it will also allow us to evaluate the potential of such measurements for locating reprocessing facilities in Western Europe, in connection with atmospheric transfer data and information about ^{85}Kr releases from reprocessing plants, especially La Hague, France. In the same way, it would also be possible to state more precisely the potential of a network of Krypton automated samplers and analyzers, similar to what exist for radioxenons in the frame of CTBT implementation, for detecting and localizing illicit reprocessing activities, along with source modeling and atmospheric transfer calculations.

5. Conclusion

We designed and developed an automated sampling, purification, concentration, and measurement system for ^{85}Kr. The purification from other gases, notably Radon, and the concentration are carried out thanks to a combination of gas permeation through a Nitrogen generator membrane and room temperature adsorption – high temperature desorption cycles on ”molecular sieve” carbon. The Krypton concentration factor with respect to normal air is about 2,500. The system allows to sample about 0.7 cm^3 of Krypton for a 6-hour operating cycle. We modified a commercial proportional counter to adapt it to low-level ^{85}Kr measurement. This whole system has the potential for detecting automatically moderate increase, typically a few tenth of Bq·m^{-3}, of the ^{85}Kr atmospheric activity above background.

We plan to operate the whole system in the next months for a continuous monitoring of ^{85}Kr activity in the South of Paris. In combination with atmospheric modeling, we will probably learn a lot about ^{85}Kr activity variations with a 6 hour - time resolution and capability to relocate ^{85}Kr emitting sources in Western Europe. This will provide an evaluation of such automated equipment to detect and locate illicit reprocessing activities.

Moreover, an interesting feature of this system is that it could conceivably be divided in two parts, provided necessary alterations are performed. For instance, on the one hand a sampling unit that would be operated on the field (mobile sampler) and on the other hand, a detection unit that would remain in the laboratory. In such a case, some modifications, like use of cryogenic trapping with liquid Nitrogen, may be necessary to decrease the volume of adsorbent and the volume of gas sampled and treated by the sampling unit.

Along with the development of this technology, a reflexion has to be performed on possible schemes regarding the use of this technology and the needs of the Agency, as the outcome of this process may have a feedback on the development of the system.

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Shallow geophysics methods for DIV safeguards inspections*

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Abstract. The new framework of strengthened safeguards as well as the recent developments in shallow geophysics, led the IAEA to develop their own verification capability in this area, with the active participation of the French Support Programme.

In that goal, a dedicated programme has been build to fulfil the needs of Agency and reach basic geophysics survey capacities for inspection teams. It focused mainly on Ground Penetrating Radar (GPR) techniques. GPR has been selected according to its "*apparent"* operational easiness taking into account its wide range of application and the recent technical developments. The program comes into five main steps: 1) Identification of Agency goals and priorities, 2) Analysis of capabilities and limitations of GPR versus other techniques, 3) Assessment of available GPR equipments 4) Development of manual of references and training programme 5) Basic analysis of potentialities of other geophysical methods.

This program draws on Safeguard experience. It relies on LCPC skills and their calibrated test site and on CGG operational experience of shallow geophysics, all over the world. The programme is coordinated by CEA/DAM dealing with treaties verification R & D and environmental monitoring studies. Approach has been fed by LCPC skills in environmental survey and non-destructive testing and by AGAP, a French Association that has issued methodological guidelines in geophysics.

The context of safeguards operation is particular. Usually, data are not interpreted and processed on the field. Few inspection teams would be trained and most inspectors may have no skills in geophysics. Then, the main programme objective was to issue guidelines and training programmes that meet operation's needs. Two main goals were to identify situation where GPR would be ineffective and to ensure valid data acquisition. An in-depth analysis of priority goals of the Agency in the framework of Design Information Verification (DIV) tasks led to take into account several general configuration (Global Configuration) for various types of survey: inside a wall or a slab, behind a wall or a slab, buried facilities or structures close to a building.

GPR uses high-frequency electromagnetic (EM) waves ranging from 50 to 3000 MHz that propagate trough the surface. It allows acquiring information on subsurface structures. GPR detects changes in EM properties at boundaries (dielectric permittivity, conductivity and magnetic permeability), as function of ground material,

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* Only an abstract is presented here, as the full paper was not available.

water content and imbedded objects. GPR is used for non-destructive surveys, facilities survey or environmental survey. Recently, user-friendly acquisition interfaces and very high frequency antennas have been developed.

The analysis of performances and limitation of GPR was performed using Agency historical case. Global Questions was divided into Basic Questions as a “void in a wall”, each BQ being divided into Basic Requirements (BR) as “wall material”, common to several Basic Questions. Analysis trees were developed to assess GPR performances and limitations for different situations. Potential capacities of other geophysical and thermal techniques, which could validate or replace GPR, have been explored. Such analysis brought strong basis for GPR capacities assessment and for GPR data analysis for Operation Guidelines. The analysis of buried facilities or structures in surround of a building was enriched by a chart providing to the inspector an idea of the capacities of techniques for different objectives assumptions and depending on field context observation.

To enable objective selection of equipments accordingly to Agency targets, an assessment of GPR was conducted at LCPC test site with four manufacturers among a selection of five ones. The test included different calibrated objects buried in a controlled geological medium and stuffed with LCPC concrete structures with hidden objects. A protocol was laid down, for antennas, acquisition parameter, real or test survey design as for pre-processing parameters. Free survey was possible. Data were processed with the same parameters, manufacturer and software, REFLEXW. Equipment was assessed along geophysical factors such as penetration, logistical parameters and utilization features. This test confirmed that the GPR could detect features buried or hidden behind a wall, such as metallic or plastic pipe. It provided a basis to build the training programme, the user guideline and the reference manual including on-site data assessment and quality control at each step. The analysis allowed identifying the performances of some manufacturers in a given domain. It provided the agency an exhaustive and objective view of equipment capacities accordingly to their needs.

The field test confirmed that GPR could be an efficient tool for safeguards. Performances and limitations analysis is used to develop guidelines and reference manuals for operations. The approach aims to perform sound data acquisition on the field. In the perspective of in-depth analysis of other geophysical techniques, this programme should enable a first stage of implementation and provide new data. Even if data fusion and methodology have to be further developed and in spite of known limitations, GPR by providing new information, will strengthen IAEA safeguards capabilities

Ground penetrating radar method for safeguards – Examples at Olkiluoto spent fuel disposal site in Finland

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Abstract. Assessment of geophysical methods and experiences from site investigations contribute to the development of safeguards approach for spent fuel repository at Olkiluoto, Finland. Electromagnetic ground penetrating radar (GPR) and borehole radar were considered as being applicable, commercially available non-destructive methods. Radar has been used for detection of cavities, man-made structures and large diameter holes. Applicability depends on specific bedrock properties (electrical conductivity, dielectric permittivity, and rock type itself). In Olkiluoto the radar would image the bedrock volume from tunnels up to 30 – 40 m distance when using low 30 – 60 MHz frequency. According to modeling radar can detect a 2 metre cubic void through 5 metres of typical crystalline rock. Most useful survey layouts would be long lines along tunnel walls, roof and in nearby boreholes. Low frequencies shall be used for maximum penetration and to enhance visibility of large engineered object from natural background. Surveys can be baseline and inspection (monitoring) measurements. Qualified expert(s) should design the survey, operate the instruments and carry out processing. Data management needs a system for processing, knowledge, site geometries and long-term data archiving. Radar investigations in safeguards seems to be limited to confidence building on reported underground as-built information and this is possible from limited accessible volumes. It is possible to partially disclose existence of a non-reported feature. All anomaly origins cannot be explained or conclusively proofed without supporting data. Naturally originating (geological) reflectors can act as indicators of geologic media behind covered surfaces.

1. Introduction

Radiation and Nuclear Safety Authority (STUK) is the national authority responsible for the safeguards implementation in Finland. Assessment of methods and experiences will contribute to the development of an effective safeguards approach for the final repository of spent fuel at Olkiluoto. The high level spent nuclear fuel disposal into the bedrock has been studied in detail in Finnish, Canadian, German, Swiss and Swedish projects over several decades. The investigations 1985 - 2000 demonstrated the Olkiluoto investigation site and its geological conditions suitable and feasible with respect to long-term nuclear safety (Safety Case) and constructability.

Electromagnetic Ground penetrating radar (GPR) and borehole radar has been considered as one possible non-destructive geophysical electromagnetic method to be applied. Radar instrumentation is commercially easily available and has been also applied for a variety of engineering purposes like structural studies, detection of cavities, man-made structures and large diameter holes. Study was

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conducted for STUK to collect methodological experiences, Olkiluoto site specific conditions, previous results and present an assessment of applicability of geophysical radar method in safeguarding underground repositories [1]. Both ground radar and borehole radar types of measurements are considered. The study localized the concept to Olkiluoto bedrock properties and to the current Finnish spent fuel disposal concept. The particular questions set were:

1) can the radar method be applied in safeguarding Olkiluoto repository site to routinely and systematically disclose any suspect of deviation from reported activities,
2) can the radar method be applied for cases, where specific properties are re-evaluated on demand,
3) assess and report the site properties having effect on radar method,
4) analyse the method applicability against Olkiluoto bedrock properties and construction lay-outs,
5) estimate available best practices for radar surveys.

2. Radar method

The principles and theoretical aspects of radar method are beyond the scope of this paper and the reader is referred to scientific text books available. Radar method uses mainly continuously transmitted pulse type of signal in frequency range from MHz to GHz and resulting response signal is recorded within the specified time window. Pulse has its nominal center frequency which determines penetration range in media and detectability and resolution of objects. Pulse is repeated continuously for stacking and to achieve desired coverage along the measurement profile. Stacking suppresses non-coherent noise and improves the signal-to-noise ratio. Radiated power has also considerable influence to investigation range but increase of it has several limitations. In the range of radio wave frequencies the dominating processes and events are transmission, attenuation, refraction and reflection.

Two main measuring modes exist: reflection measurement and transmission measurement. Reflection measurement is the main type of utilization of the radar method. Reflection measurement is useful because it provides an easy to understand and visual result directly from subsurface object boundaries. Example is given in Figure 1 where measurement line is horisontal and radar pulses are plotted against time in vertical section and amplitudes are displayed on grey scale. Some wave reflections are emphasised with a red marker line. It is also valid in a majority of situations that subsurface objects do encompass differences in their physical properties and thus give raise to wave reflections.

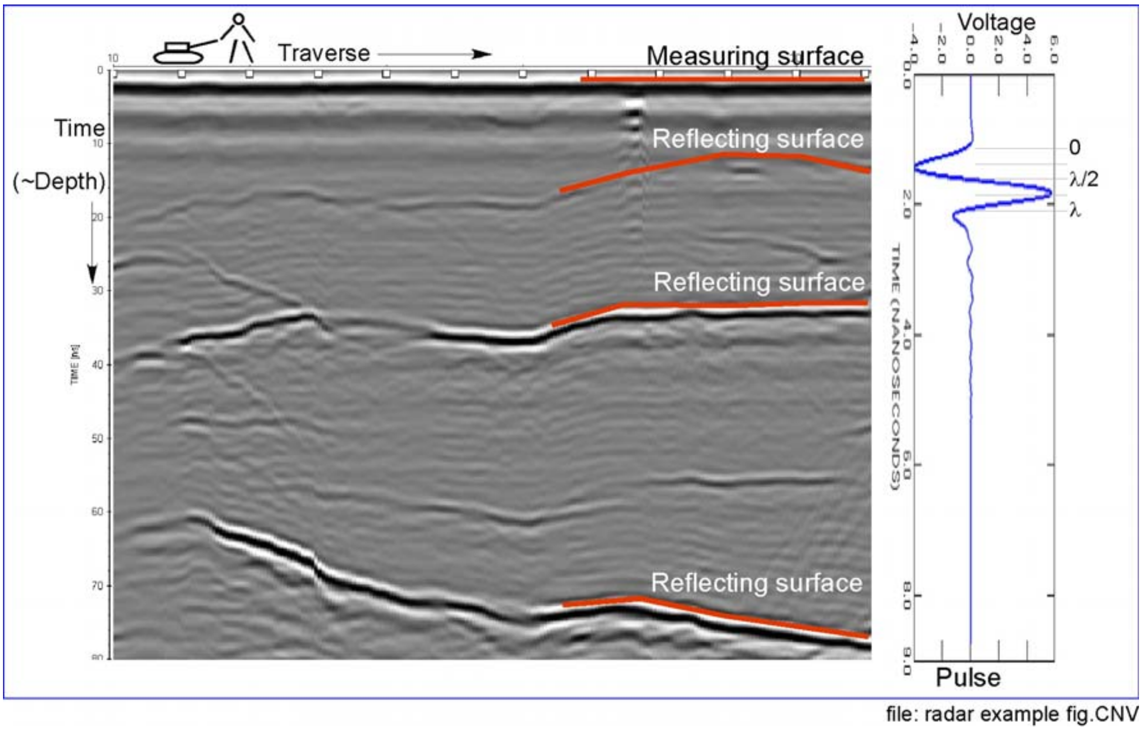

Figure 1. *Basic radar surveying, reflection image and example of radar pulse form [1].*

3. Geological conditions at Olkiluoto site and measurement results

The Olkiluoto lithology mainly consists of migmatitic gneisses, varying from banded migmatite (veined gneiss) to fine grained mica gneiss, and more pervasively migmatized diatexitic gneiss. Some amphibole containing mafic gneiss variants are found. There are also few diabase dykes. In the gneiss there occurs both intruded and partially remelt pegmatite veins.

Ground level radar surveys were not applicable for bedrock investigations and mostly soil layers can be mapped. On surface moderate penetration can be reached at the best. Site investigations have included borehole radar soundings in nine of the deep boreholes with lower frequencies of 22 MHz and 60 MHz, using at 60 MHz the directional tool. Borehole KR10 contains section over 40 – 140 m measured with several radar frequencies, 22 MHz, 60 MHz directional, 100 MHz and 250 MHz. Four tunnel pilot boreholes surveys have utilised borehole radar measurements at 250 MHz.

Electrical conductivity and dielectric permittivity of the rock mass are the affecting physical parameters. Averagely fractured Olkiluoto bedrock has the relative electrical permittivity ε_r = 6.6 – 6.8 (velocity 115 – 122 m/μs). Porosity and water content affect the radar range. Porosity ranges from less than 0.1 % of non-broken host rock to 3 – 7% in broken rock mass. The galvanic (DC) resistivity varies from very high 10.000 – 60.000 ohm-m in weakly fractured (< 3 fractures per meter), low-porosity gneissic and granitic rocks, to moderate 1.000 – 10.000 ohm-m in moderately fractured 3 - 10 fractures /m or foliated rock mass. Values in the range <1 – 1.000 Ohm-m are met in intensely fractured or altered rock mass, often containing of pyrite and graphite. Saline groundwater reduces achievable radar range. The resistivity is frequency-dependent and useful parameter is the resistivity (lower) at the radar frequencies. Resistivities and ranges from various rock types are listed in Table 1.

Table 1. *Olkiluoto rock mass conditions, resistivities (in ohm-m) and radar ranges [1].*

Bedrock and groundwater conditions	22 MHz		60 MHz		100 MHz		250 MHz	
	Range, m	Resistivity	Range, m	Resistivity	Range, m	Resistivity	Range, m	Resistivity
Grey gneiss, granite, veined migmatite, sparsely fractured, fresh water (DC resistivity >10.000 ohm-m)	20-50	300-1000	15-30	280-700	15-20	300 – 420	9-14	170-300
Migmatite, mica gneiss, moderately 3-10 1/m fractured, fresh water (DC resistivity 5.000-10.000 ohm-m)	10-20	200-300	10-15	160-280	8-15	170 – 300	5-9	82-170
Grey gneiss, granite, migmatite, sparsely fractured, saline water (DC resistivity 5.000-10.000 ohm-m)	10-20	200-300	10-15	160-280	8-15	170–300	5-9	82-170
Migmatite, mica gneiss, moderately 3-10 1/m fractured, saline water (DC resistivity 1.000-5.000 ohm-m)	5-15	65-240	5-10	75-160	5-8	75 – 170	3-6	44-103
Densely >10 1/m fractured zones, intensely foliated migmatite, altered or porous zones, saline water	<10	<200	<10	<160	3-8	40 – 170	2-5	27–82
Sulphide or graphite bearing zones, fracture zones with saline water (DC resistivity < 200 ohm-m)	None		None		None		None	

Comparison of images and ranges and observability of natural fractures is composited for borehole KR10 depth section 40 – 140 m, where several frequencies 22 MHz, 60 MHz, 100 MHz and 250 MHz have been used. Radargrams are displayed in Figure 2.

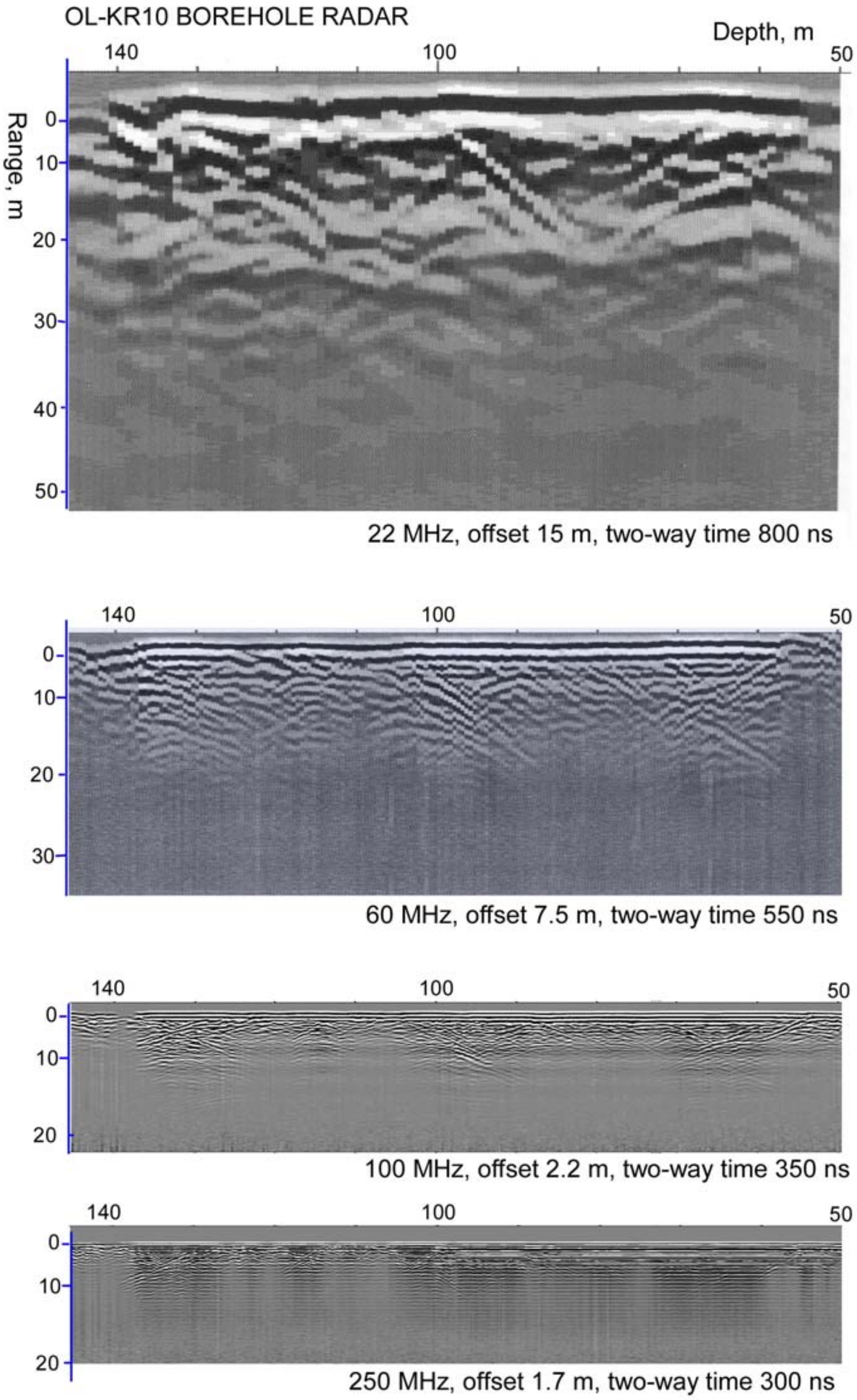

Figure 2. *Radar images of frequencies 22 MHz, 60 MHz, 100 MHz and 250 MHz [1].*

The natural conditions in Olkiluoto bedrock display a large amount of reflectors of different amplitude intensities and continuities. Within section 40 – 140 m, seven reflectors were recorded at 22 MHz, 13 at 60 MHz, 48 at 100 MHz and 78 at 250 MHz. This is normal because higher frequencies and shorter offsets can map geometrically smaller and thinner physical inhomogeneities. Considerable ringing caused of antenna is also visible as horizontal striping, see 250 MHz result. The natural reflections are at low (22 – 50 MHz) frequencies and large transmitter-receiver offsets (2 – 15 m) most often local or regional fracture zones and continuous layers containing conductive minerals. At higher frequencies 100 – 500 MHz, shorter tool offsets (0.2 – 1 m) and dense sampling rates, the amount of reflectors is dramatically increasing, representing occurrence of individual fractures (coated, clay or water filled), surfaces of intense foliation, and rock type or vein contacts. Point like natural reflectors do also exist in Olkiluoto bedrock mass. The natural reflectors are sometimes very continuous and strong although their geometric character may be irregular (example is a metadiabase dyke).

4. Numerical modelling

Extensive numerical modelling exercise has been done previously by Seidel et al. [2] in relationship with radar method and safeguards use. Supplementary 2-D modelling were made with ReflexW software [3], version 3.0.7, in this study [1]. Two additional models were derived to for Finnish repository construction conditions. Frequency selected was 50 MHz. First calculation (Model 1) dealt with a 2 x 2 meter opening at the distance of 5 meters from tunnel surface and the excavation damage zone (EDZ) including the grout layer added (0.5 m thick layer). Model used electrical parameters from Olkiluoto and various estimates for EDZ. Opening was either air or water filled one.

Results showed that both host rock as well as EDZ and grouting layer with decrease in resistivity diminish strongly surveying range and reflections. In all cases reflections from EDZ and grouting layer are very strong. It is likely that in practise these will cause secondary reflections and ringing even more than the numerical results indicated. Response from the opening is strong in both air and water filled cases because the physical contrast is very high.

Second example (Model 2) was a theoretical situation where detection of a canister and deposition hole is studied from a near-by tunnel. The geometry is naturally a full 3-D situation but some insight could be gained with the help of 2-D model. GPR line is run along tunnel wall characterised and excavated. Disposal activities run parallel to this in an another tunnel at distance of about 25 meters. Planned disposal operations consist of installing the canister, backfilling and construction of sealing structures. Three different cylinders of diameter 1.5 meter were modelled. The first cylinder represents the copper canister within a layer of bentonite, the second cylinder is filled with air and the third one is filled completely with bentonite. Reflections are small in amplitude from such a 20 – 25 m distance. Reflection from bentonite only filled canister hole is weaker than from the copper cylinder filled ones.

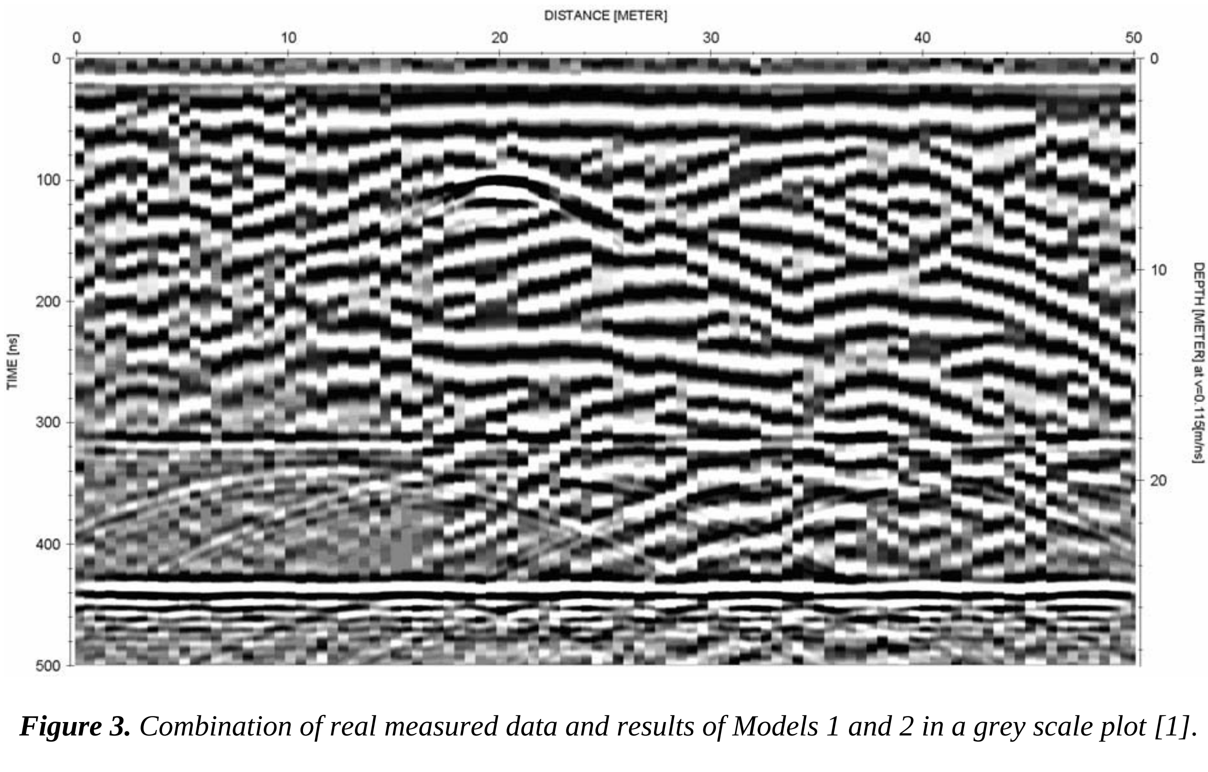

Figure 3. *Combination of real measured data and results of Models 1 and 2 in a grey scale plot [1].*

The modelled data was stacked with a real measured data set from borehole KR10. Results from Model 1 and 2 were included to real measured data set comparison, presented in Figure 3 as grey scale image. All data sets were balanced as well as possible and stacked. Reflection of 2 x 2 m wide tunnel from 5 m of the tunnel wall is clearly detectable. Natural reflections are masking severely the signal from the tunnel and canister objects. Thus it would be essential to plan well real survey parameters to distinguish the man-made objects from natural. In a monitoring type of survey it is favourable that accurate geometries are known as a basis. Man-made objects form straight and coherent reflections.

5. Future applicability

In the repository application of radar, most useful are longer survey lines along tunnel walls, roof and in boreholes in close proximity (Figure 4). Filling material on the floor may hinder radar performance on it. Low 30 – 60 MHz frequencies shall be used for maximum penetration and enhancement of larger engineered object visibility from natural background reflections. Surveys divide themselves to primary (baseline) and inspection (monitoring) types, where the inspection measurements would be compared to preceeding baseline data and any changes monitored. Changes in the bedrock properties would also be seen in the data. All the settings and tool and geometrical properties should be similar in the repeat sections.

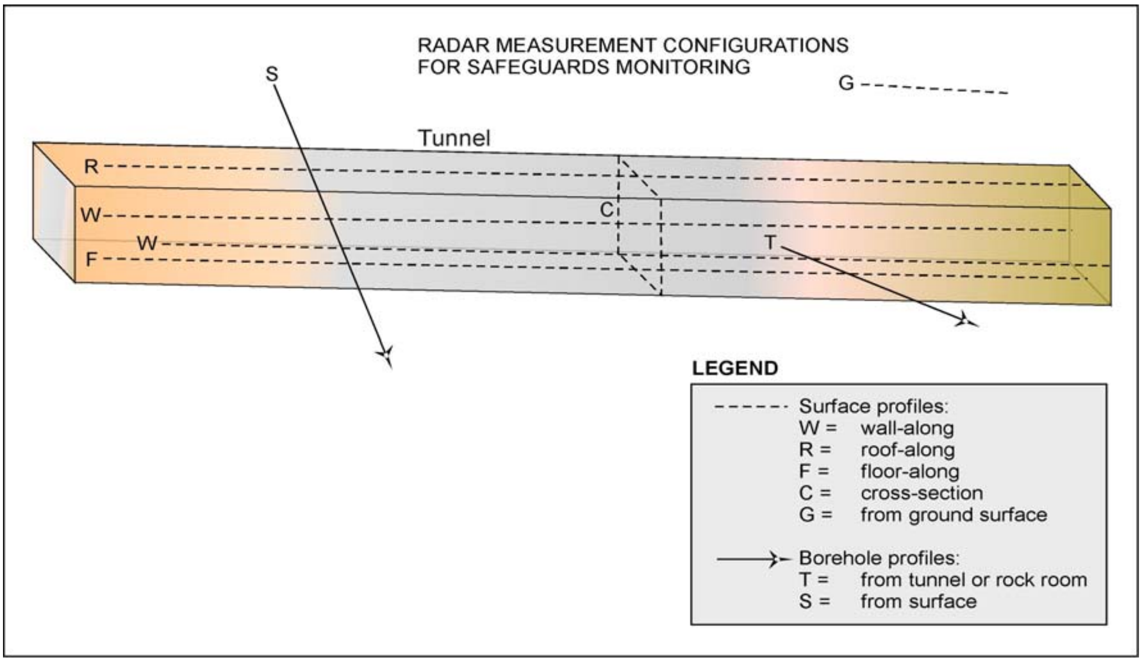

Figure 4. *Radar measurement configurations for safeguards surveying [1].*

The application of radar requires qualified expert for designing the survey, operating the instruments and carrying out processing. Data management requires a system where processing, knowledge, geometries and long-term data archiving are done. The procedures should be thoroughly documented and quality assured, tested and validated. The tools can be tested at a suitable site to study instrument effects, repeatability, sensitivity and lay-out issues. Availability of boreholes and underground rock rooms for radar measurements may vary considerably through time. Positioning and reliable coordinate data are important in the survey.

Natural reflectors are mixed with man-made objects and materials in the results. Important is to distinguish between geological objects, known man-made objects and non-identifiable ones (either natural or man-made). Engineered structures, communication links and electricity installations, as well as irregularities in tunnel shape would be visible in the results. On the other hand, the possible unreported structures may imply a distinguishable signature in the data. Preferrably non-covered tunnel surface is optimal for the survey compared to shotcrete reinforced one.

Results should be labeled in processed data results, and classified to screen features that cannot be explained with available reference data, conform with characteristic target shapes, and not explained with identified engineered objects. These need further evaluation. Careful evaluation is required as it has been shown that in Olkiluoto crystalline rock mass responses from various types of geological objects are met. Characteristics of the anomaly can discriminate its source. The shape of anomaly – especially in form of hyperbola originating from tunnel and small rock room – is characteristic for man-made objects. Magnitude of anomaly from a man-made object is typically large because of the high contrast in physical properties. Bearing this in mind a large number of near-lying objects associated with weak anomalies can be classified as of geological origin. In distant and margin areas reached by radar also response from open rock space is a small one but normally larger than from a geological object residing at a similar distance.

Different timing options and schedules of the surveys are presented in Figure 5. First radar survey forms a baseline which is comparable against the later conducted ones. Second type of survey is of monitoring or inspection type. Its purpose is to map differences between two or several surveys.

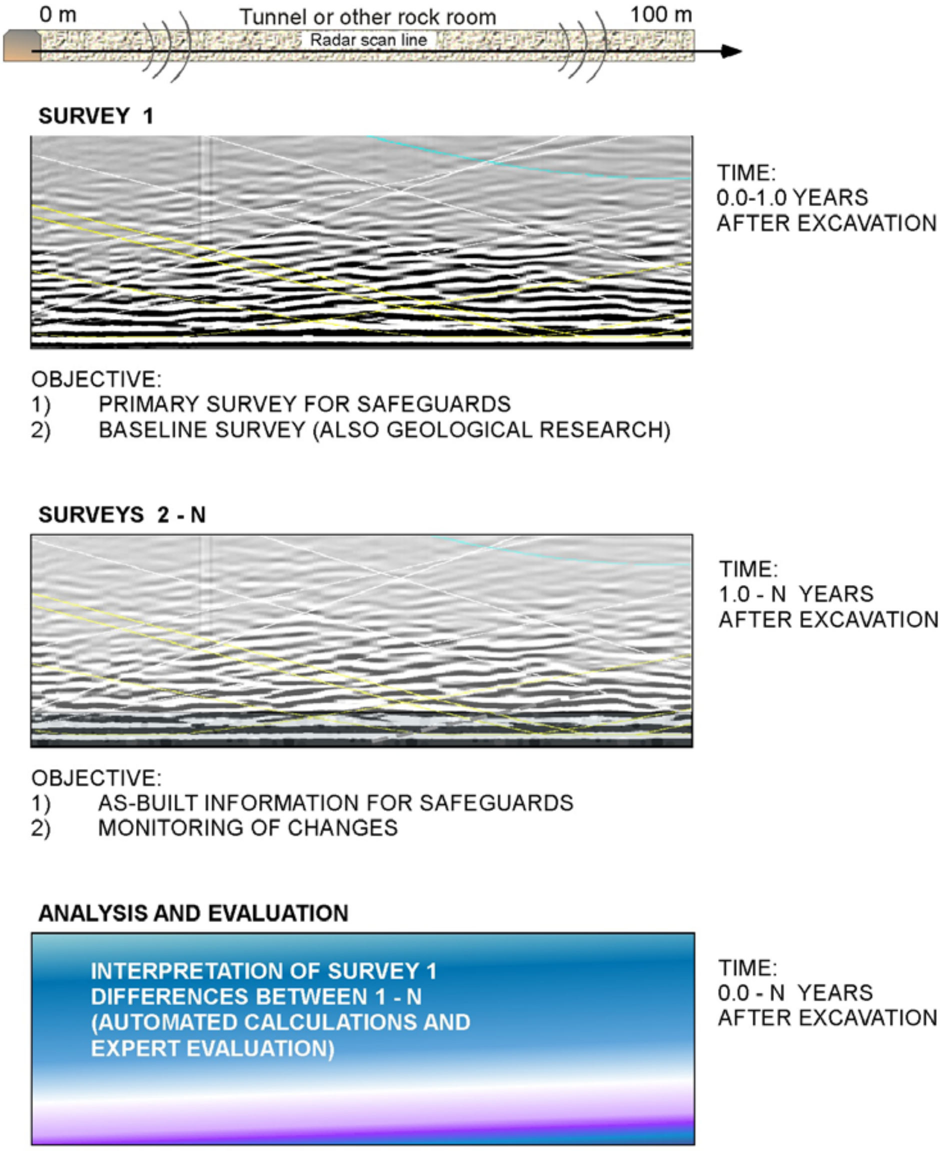

Figure 5. *Schematic of the radar measurement timing and purpose during repository construction [1].*

Results from a monitoring survey contain in most cases add-ons. Noise signals from constructions and changes in natural conditions may be integrated to results at any phase. Changes in rock mass can take place through drying of rock, stress state, new microfractures, mineralogical-chemical changes and changes in groundwater composition. Amount of reinforcement and grouting will depend on local conditions, and may thus vary even within short distances along the tunnel wall. Tunnels and rock rooms contain also cables, permanent instruments, electromagnetic noise sources like communication systems, lighting devices, dielectric and other conductive installations. Also new boreholes for characterisation or grouting may have been established as well as near-by rock rooms. During a repeated survey it is important to follow the positions, geometry, instrumentation, settings and conditions from previous survey as accurately as possible. All deviations need to be recorded.

Natural geological conditions and its inherent presence in radar images is important part in the comparison of the results from the surveys. Two main uses can be:

- comparison between two radar images and finding differences in them which are not explained by geological features, known man-made structures or changes in the environment.

- using radar response from hosting rock mass as evidence of natural, geological conditions. Parts of radar profiles indicating disturbances may point out areas where unknown rock rooms or open spaces situate.

6. Conclusions

Radar method for safeguards seems to be restricted onto building confidence on reported as-built information and on characterizing geological features around underground rock rooms . This would be possible from limited volumes owing to access and rock mass conditions (range). Partial verification of the absence or disclosure of non-reported features is possible. Radar method was estimated to be able to image a 2 metre cubic void at 5 metres behind covered rock face in crystalline rock conditions.

A number of radar reflections originate from bedrock itself, especially at Olkiluoto. The origins of all of anomalies can not be verified in a conclusive manner without supplementary investigations (direct observations). The nature of observed reflectors can act as indicator of geologic media behind the covered surfaces. The use of radar survey results would essentially be based also on possibility to combine data from different disciplines, mainly geology and underground room design information. These should be used parallel with radar interpretation.

Methods can be applied either directly in concurrent task, or in several phases designed so, that first baseline measurement would be conducted and documented immediately after completion of excavation, and possible changes monitored later on. Good data management and visualization resources are required for this purpose.

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Deterrence, Technology and the Sensible Distribution of Verification Resources

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Abstract

In the context of the integrated safeguards regime as defined under the Additional Protocol, we address the determination of a rational basis for the distribution of available resources for routine inspections, both among locations within a controlled State as well as among States themselves. In the spirit of the original conception of the NPT safeguards system, which emphasized objectivity, we adopt a formal, quantified point of view.

1 Introduction

Under the NPT Additional Safeguards Protocol [1], both the obligations of States to provide safeguards-relevant information as well as the scope of inspections have been, in principle at least, considerably expanded. It is apparent that, if this extension is to bring with it an improvement in safeguards effectiveness and efficiency, it should not lead to an amplification of the situation in which States with large nuclear fuel cycles and minimal motivation to violate are most heavily controlled, while other States with obvious motivations are able to deceive the safeguards system successfully.

In the context of this emerging, integrated safeguards regime, we shall address the determination of a rational basis for the distribution of available resources for routine inspections, both among locations within a controlled State as well as among States themselves. In the spirit of the original conception of the NPT safeguards system, which emphasized objectivity and impartiality, we adopt a formal, quantified point of view. The mathematical framework for quantitative analysis is provided by the theory of rational behavior in conflict situations, i.e. game theory. In previous investigations of this kind [2, 3, 4, 5], two general criteria have been considered and related quantitatively to one another:

- the technical capabilities of the inspection authority, expressed as probabilities of detection achievable for hypothetical illegal activities, and
- the political motivation and priorities of the parties involved, described in terms of their respective utility functions.

In the present work we include, additionally,

- the technical resources of the controlled states, parameterized by critical conversion times.

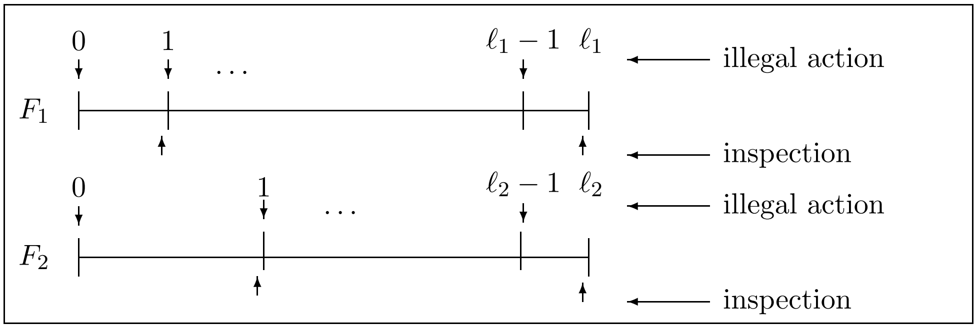

Figure 1: Two facilities in one State. The number of critical time periods within the reference interval is ℓ_i, $i = 1, 2$. An inspection can occur at the end of any critical time period. An illegal action will occur, if at all, at the beginning of a critical time period. Coincidence of inspection with illegal action is interpreted as non-detection.

The notion of critical time and its implications for inspections was introduced for IAEA safeguards a long time ago [6]. The idea is that the illegal construction of a nuclear explosive device, starting, e.g. with plutonium diverted from a declared fuel fabrication plant, takes some critical time, and that this illegal action should be detected within that time. This has been studied in detail recently in the context of a single facility in a single State [7], whereby sequential and non-sequential game-theoretical models were differentiated. In the next section we extend this investigation by using a non-sequential model with two facilities, which nevertheless captures the basic interaction between the criteria. We place emphasis on the conditions necessary for fulfillment of the fundamental safeguards objective of deterrence and we derive reasonable and intuitively understandable conclusions.

Next, in Section 3, we consider two States with one facility each. Here the meaning of the critical time is slightly changed: Due to their technical resources and their scientific competence, the two States may need different critical times for a specified illegal action - if they wish to commit one - even if they start with the same material. Therefore we measure their resources and capability by their respective critical times. We maintain, however, the requirement that the Inspectorate detect an illegal action within the critical time in order to fulfill its inspection goals.

2 One State

We consider first a reference time interval, such as a calendar year, and a State with two facilities in which an illegal action, in the sense of some control agreement like the NPT or some other arms control treaty, e.g. CWC or CFE, can occur during that reference time. The illegal action takes place over a certain *critical time* during which it may be detected by routine inspection in a timely way, again in the sense of the control agreement.

For simplicity, assume that the there is an integer – but different – number of critical time intervals for each facility within the reference interval and that the Inspectorate has precisely one inspection at his disposal. This strategic situation is illustrated in Figure 1.

The utilities of the two players – Inspectorate and State – for the possible outcomes can be expressed as follows:

$(0, 0)$ for legal behavior on the part of the State

$(-a_i, -b_i)$ for timely detection of illegal activity in facility i

$(-c_i, d_i)$ for no timely detection of illegal activity in facility i

whereby $i = 1, 2$ and

$$0 < a_i < c_i, \; 0 < b_i, 0 < d_i \quad \text{for } i = 1, 2. \tag{1}$$

Denote by p_i, $i = 1 \dots \ell_1$ the probability that the inspection takes place at time i in facility 1, and by p_i, $i = \ell_1 + 1 \dots \ell_1 + \ell_2$ the probability that the inspection takes place at time i in facility 2. Similarly let q_i, $i = 1 \dots \ell_1$ denote an illegal action at time $i - 1$ in facility

1, q_i, $i = \ell_1 + 1 \ldots \ell_1 + \ell_2$ denote an illegal action at time $i-1$ in facility 2 and q_0 denote the probability that the State behaves legally.

The strategic situation for the protagonists Inspectorate and State can then be formulated as a two-person non-cooperative game with finitely many strategies (so-called bimatrix game) with Inspectorate as "Player 1" and State as "Player 2". We then can prove the following result, in which the behaviors of the two players are understood to be their Nash equilibrium strategies [8].

Theorem 1 *Assume without loss of generality that $d_1 > d_2$.*

Under the condition

$$1 + \frac{\ell_1 d_2}{b_1 + d_1} < \frac{\ell_1 d_1}{b_1 + d_1} \tag{2}$$

the State will behave illegally in only in Facility 1 and the Inspectorate will inspect only that facility. From the symmetry of the game, the probabilities for choice of one of the possible time points for inspection or illegal activity are all equal and given by $1/\ell_1$. The payoffs to Inspectorate and State are, respectively,

$$I_1^* = -\frac{a_1}{\ell_1} - \frac{\ell_1 - 1}{\ell_1} \cdot c_1, \quad I_2^* = -\frac{b_1}{\ell_1} + \frac{\ell_1 - 1}{\ell_1} \cdot d_1. \tag{3}$$

Under the condition

$$1 - \frac{\ell_2 d_2}{b_2 + d_2} < \frac{\ell_1 d_1}{b_1 + d_1} < 1 + \frac{\ell_1 d_2}{b_1 + d_1} \tag{4}$$

inspection and illegal action can occur at both facilities. Strategies and payoffs are as follows:

$$\begin{aligned} p_i^* &= \frac{\ell_2(d_1 - d_2) + b_2 + d_2}{\ell_1(b_2 + d_2) + \ell_2(b_1 - d_1)}, \quad i = 1 \ldots \ell_1 \\ p_i^* &= \frac{\ell_1(d_2 - d_1) + b_1 + d_1}{\ell_1(b_2 + d_2) + \ell_2(b_1 - d_1)}, \quad i = \ell_1 + 1 \ldots \ell_1 + \ell_2 \end{aligned} \tag{5}$$

$$I_2^* = \frac{\frac{\ell_1 d_1}{b_1 + d_1} + \frac{\ell_2 d_2}{b_2 + d_2} - 1}{\frac{\ell_1}{b_1 + d_1} + \frac{\ell_2}{b_2 + d_2}} > 0 \tag{6}$$

$$\begin{aligned} q_i^* &= \frac{c_2 - a_2}{\ell_1(c_2 - a_2) + \ell_2(c_1 - a_1)}, \quad i = 1 \ldots \ell_1 \\ q_i^* &= \frac{c_1 - a_1}{\ell_1(c_2 - a_2) + \ell_2(c_1 - a_1)}, \quad i = \ell_1 + 1 \ldots \ell_1 + \ell_2 \end{aligned} \tag{7}$$

$$I_1^* = \frac{1 - \frac{\ell_1 c_1}{c_1 - a_1} - \frac{\ell_2 c_2}{c_2 - a_2}}{\frac{\ell_1}{c_1 - a_1} + \frac{\ell_2}{c_2 - a_2}}. \tag{8}$$

Under the condition

$$\frac{\ell_1 d_1}{b_1 + d_1} < 1 - \frac{\ell_2 d_2}{b_2 + d_2} \tag{9}$$

the State behaves legally, $q_0^* = 1$. The (non-unique) strategy of the Inspectorate is

$$\begin{aligned} p_i^* &\geq \frac{d_1}{b_1 + d_1}, \quad i = 1 \ldots \ell_1 \\ p_i^* &\geq \frac{d_2}{b_2 + d_2}, \quad i = \ell_1 + 1 \ldots \ell_1 + \ell_2 \end{aligned} \tag{10}$$

and the payoffs to both players are nil.

□

Note that the Inspectorate's strategies under both conditions (2) and (4) against the State's corresponding illegal equilibrium strategies also satisfy inequalities (10). Thus the Inspectorate can assume illegal behavior, even when the State actually behaves legally, and still be playing a Nash equilibrium strategy.

Condition (9) for legal behavior on the part of the State has a threefold generalization:

1. Probabilities β_i, $i = 1, 2$, for second kind errors in the inspection procedure can be included. These are the probabilities with which an illegal action will not be detected in facility i in spite of a timely inspection having taken place there (non-detection probabilities).
2. The number of facilities can be increased from 2 to any $k > 2$.
3. The number of inspections can be increased form 1 to $n > 1$.

Then condition (9) for legal behavior becomes, with $\tau_i = 1/\ell_i$, $i = 1 \ldots k$,

$$\sum_{i=1}^{k} \frac{1}{\tau_i} \cdot \frac{1}{1-\beta_i} \cdot \frac{1}{1+b_i/d_i} < n. \tag{11}$$

The complete solution of this generalized model is very complicated, as one can guess from the special solution presented in Theorem 1. However the condition for legal behavior is from the point of view of the Inspectorate the crucial aspect. Indeed it is then possible, when this condition is fulfilled, to write down the equilibrium strategy for the Inspectorate:

$$\begin{aligned}
p_i^* &\geq \frac{1}{1-\beta_1} \cdot \frac{1}{1+b_1/d_1}, \quad i = 1 \ldots \ell_1 \\
p_i^* &\geq \frac{1}{1-\beta_2} \cdot \frac{1}{1+b_2/d_2}, \quad i = \ell_1 + 1 \ldots \ell_1 + \ell_2 \\
&\vdots \\
p_i^* &\geq \frac{1}{1-\beta_k} \cdot \frac{1}{1+b_k/d_k}, \quad i = \ell_1 + \ldots + \ell_{k-1} + 1 \ldots \ell_1 + \ldots + \ell_k.
\end{aligned} \tag{12}$$

The normalization of the p_i^*, $i = 1 \ldots \ell_1 + \ldots + \ell_k$, in fact leads to (11). On the basis of condition (11) we can conclude that the State will behave legally when:

- the critical times τ_i are not too short, that is, when the technological abilities of the State are sufficiently limited,
- the detection probabilities $1 - \beta_i$ are sufficiently high, that is, when the safeguards system is sufficiently effective, and
- the ratio of sanctions b_i for detected illegal behavior to incentive d_i to act illegally is sufficiently large, that is, when effective deterrent measures accompany the control system.

It is of course reasonable to assume that, for a single State, the deterrence parameters would be essentially the same for all of its facilities,

$$\frac{b_i}{d_i} = \frac{b}{d}, \quad i = 1 \ldots k. \tag{13}$$

In that case one can obviously unravel condition (11) such that the technical parameters τ_i, β_i, k and n are on one side of the inequality and the less tangible deterrence parameters b and d are on the other. This would eventually allow a parameterized implementation of control measures in which technical aspects alone determined inspection effort. That is, by specifying required inspection frequencies, detection probabilities and critical times, the politically sensitive deterrence parameters are *implicitly* defined. In the situation described in the next section, where the case of more than one controlled State is considered and a formally identical condition to inequality (11) arises, this will not be possible.

3 Two States

Let us now assume that the Inspectorate is dealing with two sovereign States, not cooperating with one another and each in possession of one safeguarded facility having, in general, differing critical times $\ell_1 > 1$, $\ell_2 > 1$, per reference period. Each of the three players is restricted to one action (inspection or illegal activity) during the reference period. We are now dealing with a three-person non-cooperative game, the payoff parameters of which are as given in Section 2 (where now the index refers to the State) and satisfy (1). The payoff to the inspector is the sum the payoffs he receives through controlling the two States. We can prove the following theorem in which, as before, the behaviors of the three players are understood to be their Nash equilibrium strategies.

Theorem 2 *Assume without loss of generality that*

$$\frac{\ell_2}{c_2 - a_2} < \frac{\ell_1}{c_1 - a_1}. \tag{14}$$

Under the condition

$$\frac{\ell_2 d_2}{b_2 + d_2} > 1 \tag{15}$$

both States behave illegally, $q^*_{1,0} = q^*_{2,0} = 0$. *The probabilities for performing the illegal actions are uniformly distributed over the possible times and given by* $1/\ell_1$ *resp.* $1/\ell_2$. *The Inspectorate only controls State 2, that is,*

$$\begin{aligned} p^*_i &= 0, \quad i = 1 \ldots \ell_1 \\ p^*_i &= 1/\ell_2, \quad i = \ell_1 + 1 \ldots \ell_1 + \ell_2. \end{aligned} \tag{16}$$

The payoffs to the three players are

$$\begin{aligned} I^*_0 &= -c_1 - c_2 + \frac{c_2 - a_2}{\ell_2} \\ I^*_1 &= d_1 \\ I^*_2 &= d_2 - \frac{b_2 + d_2}{\ell_2}. \end{aligned} \tag{17}$$

Under the conditions

$$\frac{\ell_2 d_2}{b_2 + d_2} < 1 \quad \text{and} \quad \frac{\ell_1 d_1}{b_1 + d_1} + \frac{\ell_2 d_2}{b_2 + d_2} > 1, \tag{18}$$

State 1 behaves illegally:

$$q_{1,i} = 1/\ell_1, \quad i = 1 \ldots \ell_1,\ q_{1,0} = 0, \tag{19}$$

and State 2 chooses with positive probability both legal and illegal strategies,

$$q^*_{2,i} = \frac{1}{\ell_i} \cdot \frac{c_1 - a_1}{c_2 - a_2},\ i = 1 \ldots \ell_2, \quad q^*_{2,0} = 1 - \frac{\ell_2}{\ell_1} \cdot \frac{c_1 - a_1}{c_2 - a_2}, \tag{20}$$

the Inspectorate's strategy is

$$\begin{aligned} p^*_i &= \frac{1}{\ell_1}\left(1 - \frac{d_2}{b_2 + d_2}\right), \quad i = 1 \ldots \ell_1 \\ p^*_i &= \frac{1}{\ell_1} \cdot \frac{d_2}{b_2 + d_2}, \quad i = \ell 1 + 1 \ldots \ell_1 + \ell_2, \end{aligned} \tag{21}$$

and the payoffs to the three players are

$$\begin{aligned} I_0^* &= -c_1 + (c_1 - a_1)\frac{1}{\ell_2}\cdot\frac{b_2}{b_2+d_2} \\ &\quad + \frac{\ell_1}{\ell_2}\left(-(c_1-a_1)\frac{c_2}{c_2-a_2} + (c_1-a_1)\frac{d_2}{b_2+d_2}\right) \\ I_1^* &= d_1 - (b_1-d_1)\frac{1}{\ell_2}\cdot\frac{b_2}{b_2+d_2} \\ I_2^* &= 0. \end{aligned} \tag{22}$$

Under the condition

$$\frac{\ell_1 d_1}{b_1+d_1} + \frac{\ell_2 d_2}{b_2+d_2} < 1 \tag{23}$$

both States behave legally,

$$q_{1,0}^* = q_{2,0}^* = 1, \tag{24}$$

the Inspectorate's non-unique strategy is

$$\begin{aligned} p_i^* &\geq \frac{d_1}{b_1+d_1}, \quad i = 1\ldots\ell_1 \\ p_i^* &\geq \frac{d_2}{b_2+d_2}, \quad i = \ell_1+1\ldots\ell_1+\ell_2, \end{aligned} \tag{25}$$

and the payoffs to all players are nil.

□

Whereas the condition (23) for legal behavior on the part of both States is formally identical to the condition (9) for legal behavior of one State with two facilities, the assumptions and solutions for illegal behavior are very different. Moreover these solutions do not satisfy (25), i.e., they are not Nash equilibria if the States behave legally. The strategy (20) for instance is of particular interest because the second State mixes legal and illegal behavior although it obtains the same payoff as for purely legal behavior. This property of Nash equilibria can be observed in other inspection games. We note also that in this case, unlike the solution for the two-person game of Section 2, each player's payoff does not depend only upon its own utilities, but on those of the other protagonists, see Equations (22).

A generalization of the condition (23) for legal behavior completely analogous to that of Section 1 can be made:

$$\sum_{i=1}^{k} \frac{1}{\tau_i}\cdot\frac{1}{1-\beta_i}\cdot\frac{1}{1+b_i/d_i} < n. \tag{26}$$

Here the sum is over the number k of States (each with one facility) involved in the inspection regime.

4 Conclusion

Since in condition (26) we are dealing with different sovereign States it is patently impossible to make an assumption like (13), that is, to equate the utility ratios b_i/d_i with the aim of "factoring out" the technical aspects from the political ones. All of the parameters are inextricably woven together and, in the rational planning of routine inspections (deciding on required detection probabilities, inspection frequency and detection times), the assessment of States' incentives to illegal behavior and perceptions of the consequences of detection cannot be avoided. Precisely this sort of assessment is implied in the Additional Protocol, where States' openness and degree of cooperation in making their peaceful nuclear activities as transparent as possible to the safeguards authority should influence the intensity of routine verification effort.

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Systems analysis for evaluation of safeguards effectiveness

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Abstract. The U.S. Department of Energy (DOE) is interested in developing tools and methods for use in designing and evaluating safeguards systems for current and future plants in the nuclear power fuel cycle. The DOE is engaging several DOE National Laboratories in efforts applied to safeguards for chemical conversion plants and gaseous centrifuge enrichment plants. As part of the development, Lawrence Livermore National Laboratory has developed an integrated safeguards system analysis tool (LISSAT). This tool provides modeling and analysis of facility and safeguards operations, diversion path generation, and safeguards system effectiveness. The constituent elements of diversion scenarios, including material extraction and concealment measures, are structured using directed graphs (digraphs) and fault trees. Statistical analysis evaluates the effectiveness of measurement verification plans and randomly timed inspection plans. Time domain simulations analyze significant scenarios, especially those scenarios involving alternate time ordering of events or issues of timeliness. The simulation can provide additional information to the fault tree analysis and can help identify the range of normal operations and additional plant operational signatures of diversions. LISSAT analyses can be used to compare the probabilities of detection of diversion for individual safeguards technologies as well as overall strategy implementations for present and future plants. Additionally, LISSAT can be the basis for a rigorous cost-effectiveness analysis of safeguards and design options. LISSAT has been applied to evaluation of safeguards for uranium chemical conversion plants and is being applied to safeguards for gaseous centrifuge enrichment plants and nuclear power reactors.

1. INTRODUCTION

Lawrence Livermore National Laboratory has developed an integrated safeguards system analysis tool (LISSAT), which is a framework for performing systems analysis of safeguards effectiveness for all stages of the nuclear fuel cycle. The method has been applied to assess safeguards effectiveness for a conversion facility [1], an enrichment facility [2] and for various types of nuclear reactors [3]. LISSAT has the potential for evaluating safeguards for future proliferation resistant designs, and evaluating current safeguards tools/methods/on-the-shelf tools to assess safeguards strategies beyond current methods. The IAEA hosted a technical meeting in Vienna on April 18-22, 2005 with the aim of further strengthening its inspection and verification approaches applied to uranium enrichment activities. The U.S. Department of Energy (DOE) is interested in developing tools and methods for potential U.S. use in designing and evaluating safeguards systems and for support of IAEA goals [4, 5].

The key components of LISSAT include digraph/fault tree analysis, statistical analysis and time-domain simulation as outlined in Figure 1. The digraph fault tree methodology presents a well-structured systematic approach for generation and analysis of the diversion paths and is more comprehensive and systematic than traditional safeguards analysis methods. The digraph analysis is an effective method to organize and structure the possible diversion activities in a diversion scenario together with the safeguards measures and activities relevant to the diversion scenario.

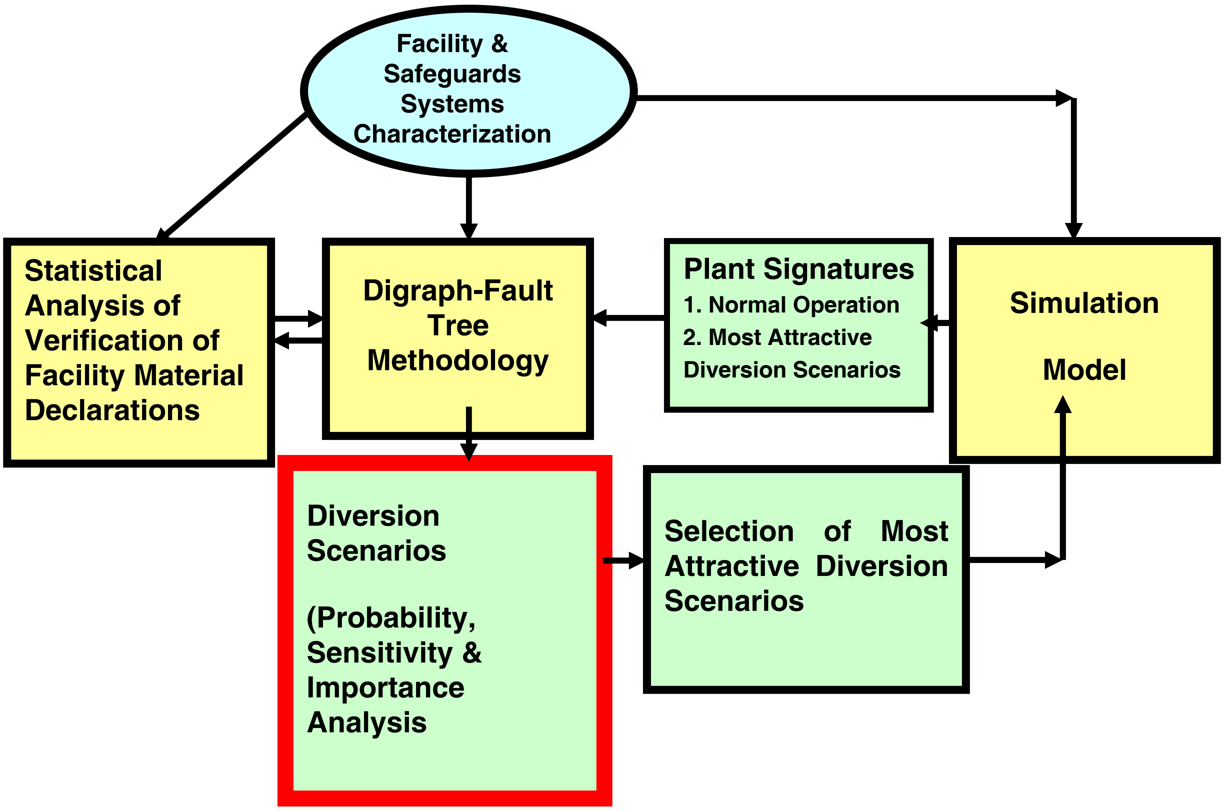

Figure 1. LLNL Integrated Safeguards System Analysis Tool LISSAT for evaluating the the effectiveness of a safeguard system for a nuclear fuel cycle facility.

The fault tree analysis incorporates possible failure modes of the safeguards measures and develops a fault tree for the safeguard system in this situation. Output of the digraph-fault tree analysis provides an identification of the safeguards elements of most importance that has the potential for reducing the probability of diversion of nuclear material. Additional outputs include the ranking of the various diversion scenarios.

Statistical analysis provides the basic event probabilities for the fault tree and addresses the probabilities that a short notice random inspections plan will encounter time-clustered physical diversion activities, the probabilities that inspection sampling and measurement plans will detect concealment schemes that use large or medium misdeclarations on some material items, deliberate bias in measurements, and leaving some loss in the material-unaccounted-for balance.

The most attractive diversion scenarios are selected for time-domain simulation. The continuous event simulations of conversion and enrichment track the uranium flow through the facility. The simulations include normal operation, intermediate storage, normal variations of input flow, and diversion scenarios. Simulation outputs are the time series of material outputs, which illustrate the data signatures of normal operation and diversion schemes. The simulation results for diversions show changes both in accumulated totals of intermediate and final material production, and in the time dependence of production. The amplitude of the results shows by how much and how soon the monitored signals are above normal fluctuations.

As a result of the analysis, if diversion paths have an unacceptably high nondetection probability, the results of both the digraph-fault tree analysis and time-domain simulation can suggest further safeguards measures. The fault tree importance analysis can suggest where further redundancy or more reliable instrumentation is required. The results of the simulation help identify the material that need to be monitored and the ideal placement of monitors for detecting diversion scenarios. The simulation may also suggest what further observations the inspector could observe. Digraphs and fault trees would be modified and reanalyzed to determine the reduction in the nondetection probability. In addition the cost of the modifications and their intrusiveness on operations would also be considered.

LISSAT analyses can be used to compare the probabilities of detection of diversion for individual safeguards technologies as well as overall strategy implementations. Additionally, LISSAT could be the basis for a rigorous cost-effectiveness analysis. Finally, the simulations could be used on a facility or process level to aid inspectors in detecting possible material diversions or difficulties with specific instruments in the field.

The following sections illustrate the application of LISSAT to reference generic plants for chemical conversion and centrifuge enrichment.

2. GENERIC CHEMICAL CONVERSION PLANT SAFEGUARDS ANALYSIS

The generic conversion plant takes uranium ore concentrates (UOC) and produces purified uranium hexafluoride, UF_6, with an input capacity of 100 MT uranium per year. The generic plant design and the reference safeguards system design were developed by a DOE multi-laboratory team [4,6,7]. The production process starts with yellowcake dissolution with nitric acid, followed by purification using solvent extraction techniques, and then evaporation to produce a concentrated, purified uranyl nitrate solution. Ammonia or ammonium hydroxide and carbon dioxide are used to produce a precipitate, ammonium uranyl carbonate (AUC). The precipitation process is operated in the batch mode. After precipitation, calcination in the presence of hydrogen produces UO_2 powder. The UO_2 is hydrofluorinated to UF_4 using HF. The UF_4 is then fluorinated into UF_6 using F_2. Each of the unit processes can operate independently. Buffer inventories of feed materials are maintained for each unit process. The reference safeguards system includes traditional safeguards measures, unattended monitoring systems, and a continuous load cell on the product filling station.

2.1 Digraph and fault tree analysis

We considered several diversion scenarios. As an example we consider diversion of 10 MTU (as UF_6) from a cold trap, with concealment of the material imbalance by trying to pass a gross defect container of the product. The actions of the plant operator and inspectors relevant to drawing the balance of material unaccounted for (MUF) are shown in digraph form in Figure 2. The corresponding fault tree is shown in Figure 3. The sampling plan is adequate to detect a MUF of one significant quantity (SQ) with the specified probability of better than 0.5, based on material accounting. With the addition of surveillance at the cold trap and the product filling station, the probability of an undetected diversion is greatly decreased, as shown in Table 1.

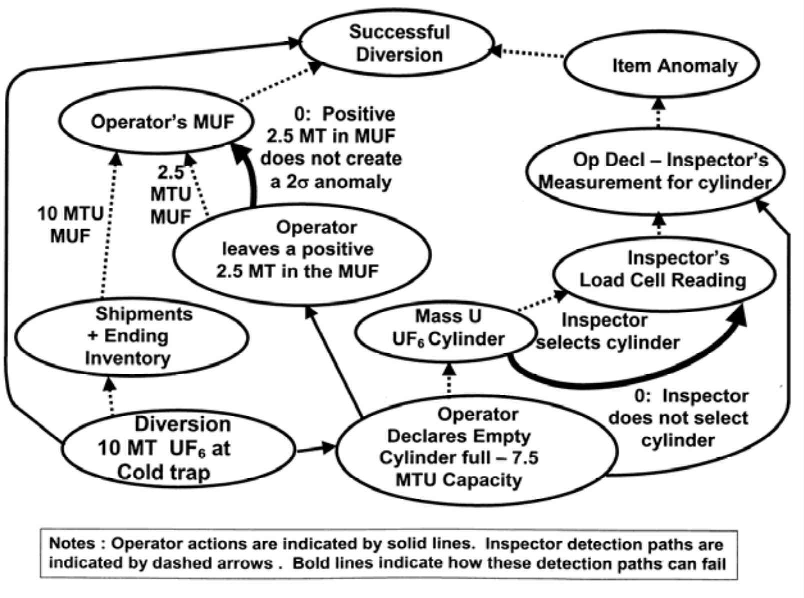

Figure 2. Digraph for Diversion of 10 MTU as UF_6 at a cold trap.

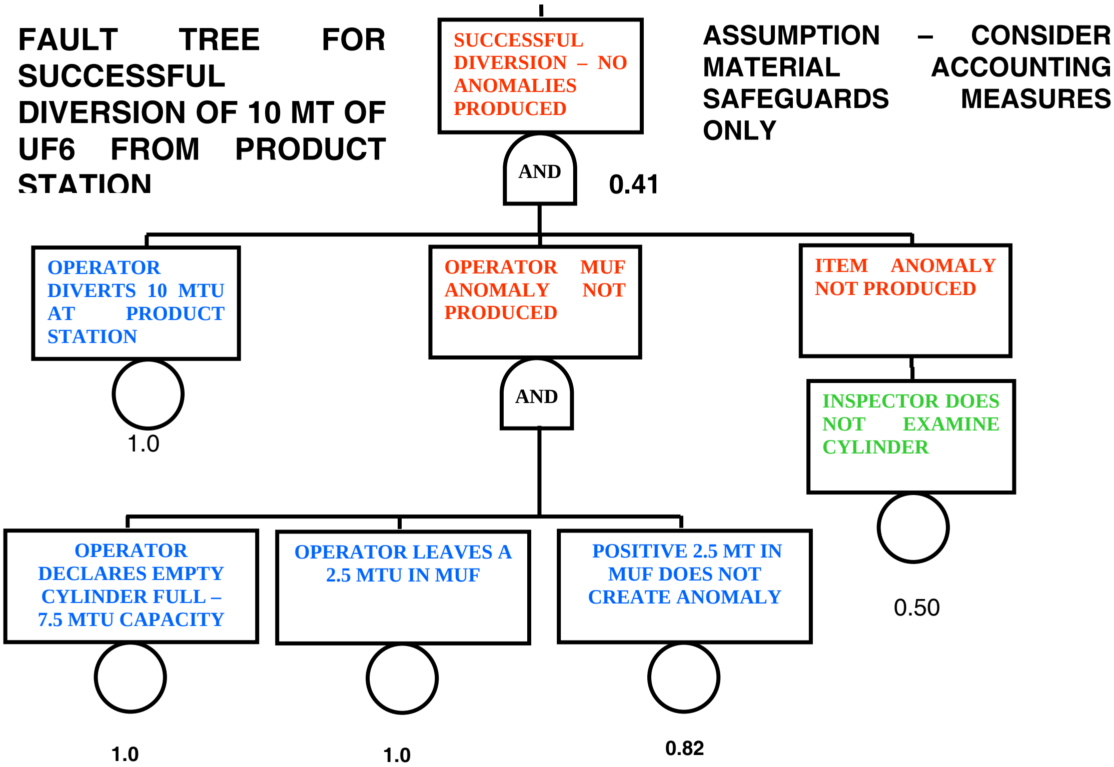

Figure 3. Fault Tree for Diversion of 10 MTU as UF_6 at the Cold Trap.

Table 1. Probability of non-detection of diversion at the cold trap for various combinations of safeguards measures.

		Material Accounting Current Safeguards practice		**Surveillance Camera At Cold Trap** Added Safeguards measure		Use both Safeguards Measures
	No Safeguards Measures	With Material Accounting	Decrease in Probability of Diversion	With Surveillance	Decrease in Probability of Diversion	
Diversion at the Cold Trap	1.0	0.41	factor of 2.5	0.011	factor of 90	0.0045

2.2 Simulation Modeling

An Extend® simulation model was developed for the generic conversion facility. Extend is a graphical, interactive, general-purpose simulation program for both discrete event and continuous modeling [8]. The simulation model consists of eleven modules, each representing part of the chemical conversion process, from receipt of material to UF_6. Figure 4 shows the plant signature when the operator diverted 10 MTU contained in UF_6 at the cold trap during a one-year period, as compared to a signature of normal operation. The total amount of uranium contained in UF_6 produced during the year dropped to 72 MTU compared to approximately 84.5 MTU, which is less than 100 MTU allowing for startup of operations. During diversion, it took longer time to produce a batch of UF_6 compared to normal operation. This time delay is another available signature of diversion.

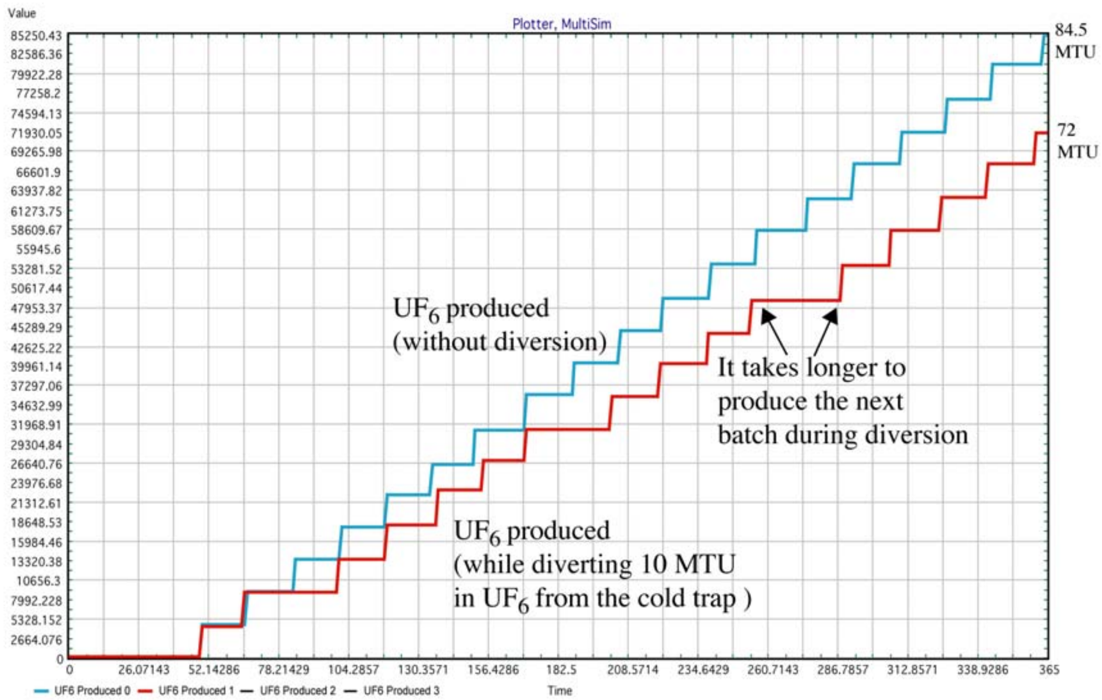

Figure 4. Simulated Total Production of UF_6 Measured in MTU under Normal Plant Operation versus under Diversion of 10 MTU in UF_6 from the Cold Trap.

3. GENERIC CENTRIFUGE ENRICHMENT PLANT SAFEGUARDS ANALYSIS

We evaluated current and potential safeguards systems a generic gaseous centrifuge enrichment plant described by Oak Ridge National Laboratory for use in a training course [9]. This is a medium sized plant with a capacity of 500 MTSWU per year, with 50 cascades of 250 centrrifuges each, producing product at 3.5% enrichment. There are several autoclaves for feeding UF_6 at natural enrichment into the plant, with one in operation at any time. Similarly there are several product and tails stations. The simulation models the operations of the storage area, feed, product, and tails stations, and weighing and sampling area, and transfers between areas. The cascade hall operation is modeled down to the cascade unit.

There are several safeguards concerns regarding GCEPs, including diversion of low-enriched uranium (LEU), excess production of LEU, and reconfiguration of part of the plant to produce HEU. As an example of the scale of a diversion scenario, skimming of 2% of the product (the product of one cascade) over a year will divert 2300 kg of LEU, containing 80 kg of ^{235}U. This is slightly above the IAEA significant quantity of LEU, which is LEU containing 75 kg of ^{235}U.

Two withdrawal points were considered for skimming of LEU:

1. Normal (outside cascade hall) – product station
2. Inside cascade hall – assume collection carts are used to remove product

The safeguards measures considered in the first analysis are combinations of:

FMI Fixed Monthly Inspections
SNRI Short Notice Random Inspections
LFUA Limited Frequency Unannounced Access inside Cascade Hall
Load Cells for weighing feed, product and tails during emptying or filling
Video S.: Video Surveillance Feed Product and Tail Station
CEMO Continuous Enrichment Monitor on cascade product headers

IAEA sampling and measurement plans are used for the input and output verification measurements for MUF under the FMI or SNRI. Observation measures include inspector observations during FMI or SNRIs outside the cascade hall and during LFUA inside the cascade hall. Options examined for continuous monitoring are load cells and video cameras at the feed, product, and withdrawal stations outside the cascade hall, and continuous enrichment monitors (CEMO) on cascade product headers.

The material acccounting verification plan is based on IAEA sampling and measurement plans, and on measurement capabilities taken from Ref. [10]. The observation activities includes inspector activities during inspections and continuous unattended video cameras, load cells, and CEMOs. The video cameras and load cells preclude attaching undeclared cylinders but do not by themselves preclude tampering with declared cylinders after they have been removed from sight of the cameras. The probability values for a table similar to Table 1 are being developed. Probability values for observation and monitoring measures should be considered indicative guidelines at this time rather than definitive values.

CONCLUSION

LISSAT provides an integrated analysis capability for evaluating proposed and potential future safeguards systems for facilities in the nuclear fuel cycle. The directed graph (digraph)/fault tree analysis provides a systematic approach to structure the moves and countermoves in a diversion/safeguards interaction. This analysis helps quantify the change in probability of detection of a diversion due to the introduction of new safeguards procedures or technology. Significant scenarios can be transferred to time domain simulation, especially those scenarios involving alternate time ordering of events or issues of timeliness. The simulation can provide additional information to the fault tree analysis and can help identify additional plant operational signatures that might assist inspectors as indicators of diversion. The complete analysis system can provide information on the relative operational and cost effectiveness of proposed safeguards procedures and technology and plant design features. The LISSAT system provided a structured framework for a multi-laboratory project for extending safeguards to uranium chemical purification and conversin plants. Applications for safeguards at GCEPs and nuclear power reactors are ongoing.

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Implementation of IAEA Policy Paper 18 in Canada

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Abstract. In 2003 the IAEA issued Policy Paper 18, re-interpreting the definition of the starting point of safeguards laid out in Article 34(c) of the Safeguards Agreement. The new policy required full IAEA safeguards measures at an earlier stage in uranium processing, thereby affecting safeguards implementation at conversion plants processing natural uranium. In Canada, this impacted the natural uranium refinery in Blind River and the natural uranium conversion facility in Port Hope (both owned by Cameco Corporation). After extensive preparatory work by the Canadian Nuclear Safety Commission (CNSC), the IAEA, and the operator, the initial physical inventory verifications were carried out at the Cameco facilities in 2005. Despite minor problems – which were to be expected for an activity of this magnitude – the overall verification was successful. This paper provides a brief overview of the implications of Policy Paper 18, as well as general descriptions of the Blind River and Port Hope facilities. Additional detail is provided on the strategies applied to the implementation of safeguards at these facilities and the main parameters of the approach employed. Future challenges related to the application of this new policy within Canada are also discussed.

1. Background

Canada's nuclear fuel cycle encompasses a broad spectrum of nuclear activities from uranium mining, through refining, conversion, fuel fabrication, and the production of nuclear energy in CANDU reactors, to the storage of spent fuel. Although technically a single process, the refining and conversion of natural uranium take place at two separate facilities in Canada, located respectively in Blind River and Port Hope, both in the province of Ontario, and both owned by Cameco Corporation. These facilities are among the largest of their kind in the world.

1.1. Cameco Blind River

Cameco Blind River (CBR), built in 1983, receives natural uranium ore concentrates (UOC) from mines and mills within Canada and around the world. The UOC is refined to uranium trioxide (UO3), most of which is sent directly to Cameco Port Hope, although some is exported. Natural uranium scrap from the conversion and fuel fabrication plants is also recycled through CBR and is added to the process at the same point as the UOC. Two by-products are produced in the refining process: regeneration product, an organic compound generated in the solvent treatment circuit representing the unusable portion of the plant treatment solvent; and calcined product from the raffinate area, which contains most of the impurities from the UOC and scrap feed. These materials are shipped to a uranium mill and essentially end up as tailings. The throughput of the Blind River plant is approximately 18,000 tonnes of uranium per year. See Figure 1 for a schematic of inputs and outputs for Cameco Blind River.

1.2. Cameco Port Hope

The conversion plant now known as Cameco Port Hope (CPH) has a long history of handling uranium, beginning in 1935 as a radium extraction facility. The site activities have changed over the years, including a period of time involving the production and casting of uranium metal. CPH now converts UO3 (initially produced on-site and now received from CBR) into either uranium hexafluoride (UF6), which is exported for subsequent enrichment or uranium dioxide (UO2), which is used primarily for the domestic production of CANDU fuel. A potassium fluoride (KF) by-product is produced from the UF6 line and is handled similarly to the by-products from CBR. As a legacy of its long and varied operating history, CPH also has large stores of depleted uranium metal as well as various forms of natural and depleted uranium scrap and waste. In addition, there are lab quantities of LEU on site, remnants of a project to blend slightly enriched powder for fuel fabrication, which has since been discontinued. The throughput of the UF6 plant is approximately 12,500 tU per year, while the UO2 plant processes approximately 2,000 tU per year. See Figure 2 for a schematic of inputs, outputs, and main process lines at Cameco Port Hope.

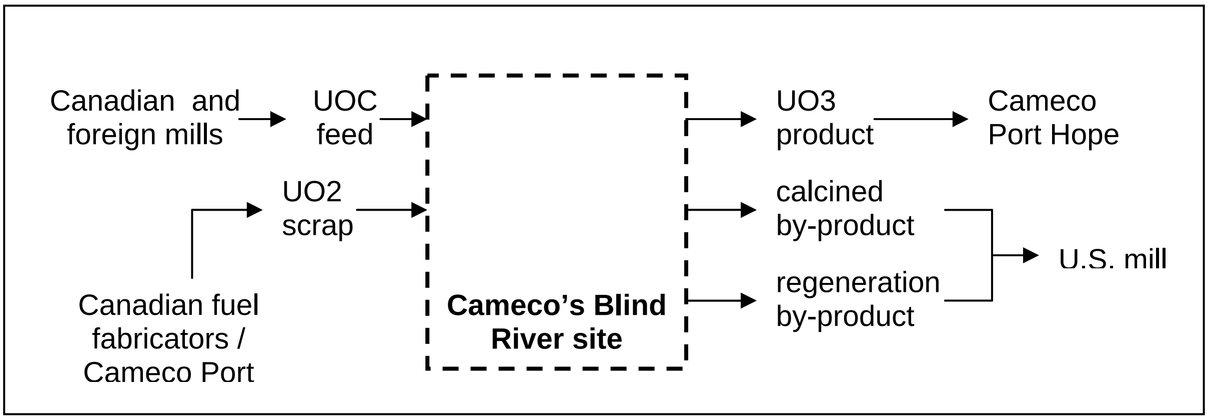

Figure 1. Main Process Line and Flows at Cameco Blind River.

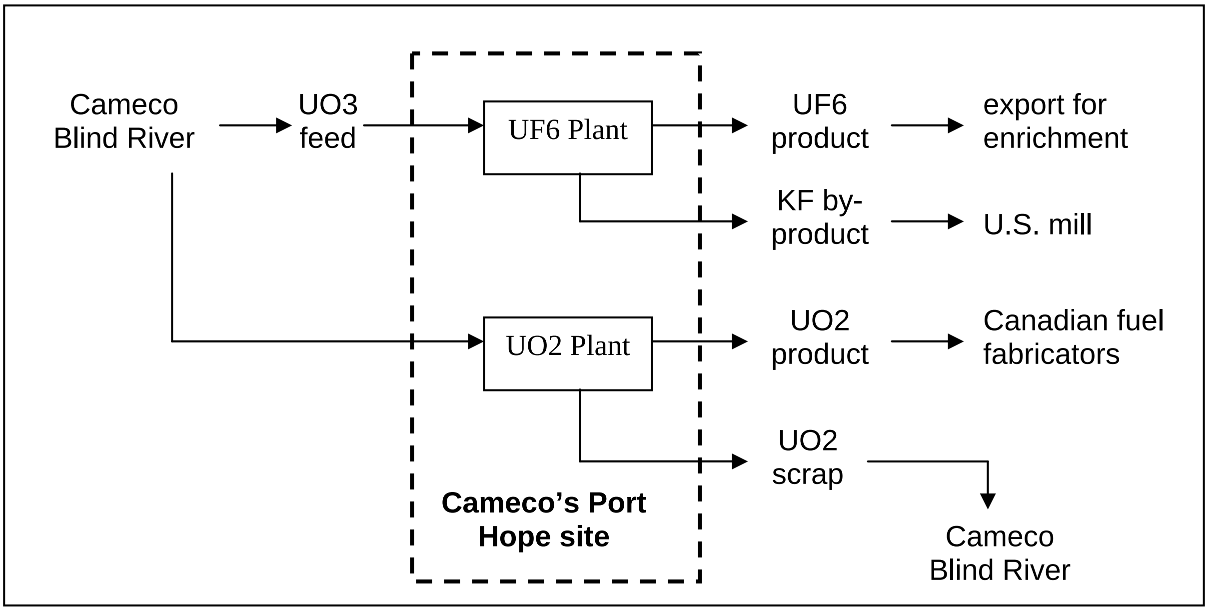

Figure 2. Main Process Lines and Flows at Cameco Port Hope.

2. Policy Paper 18

Paragraph 34(c) of the Safeguards Agreement states that safeguards begins: "When any *nuclear material* of a composition and purity suitable for fuel fabrication or for being isotopically enriched leaves the plant or the process stage in which it has been produced" [1]. This had traditionally been interpreted as the final products of the natural uranium conversion process, which in Canada's case had been the production of UF6 or UO2 at CPH.

In 2003, the IAEA produced Policy Paper 18, entitled "Safeguards Measures Applicable in Conversion Plants Processing Natural Uranium", which re-interpreted the composition and purity conditions required for the starting point of safeguards, as laid out in Para. 34(c). According to the policy paper, purified uranyl nitrate, a material produced near the beginning of the natural uranium conversion process, is the first material in this process that is suitable for chemical enrichment. If the production of uranyl nitrate in a given plant is not associated with a practical or economically viable accountability point, Policy Paper 18 states that safeguards should be applied "at the first practicable point earlier (i.e. 'upstream') in the plant" [2].

As a consequence of Policy Paper 18, States were requested to submit Design Information (DI) for the whole of each conversion plant located in the State and to subsequently place these facilities under full safeguards. For Canada, this meant the application of safeguards to the entire Port Hope conversion plant as well as, for the first time, to the Blind River refinery.

3. Process for implementing Policy Paper 18 in Canada

The implementation of Policy Paper 18 in Canada involved close collaboration by the Canadian SSAC (the Canadian Nuclear Safety Commission, or CNSC) with both the IAEA and the facility operator. The process was launched with a letter sent to Cameco by the CNSC in September 2003 informing management of the change to the starting point of safeguards and indicating that both Cameco facilities would be affected. Numerous subsequent CNSC-Cameco meetings took place, including a day-long general outreach meeting in 2004 involving Cameco staff from Blind River and Port Hope, which described the implications of Policy Paper 18 and solicited operator input. A number of meetings were also held between CNSC and Cameco upper management, to ensure support for the implementation of Policy Paper 18 at the highest level. Obtaining operator buy-in to the process at *all* levels was an important step, especially in the case of Cameco Blind River, which had no previous experience with safeguards outside of the occasional Complementary Access requested under the Additional Protocol. By ensuring that the operator was included in every step of the implementation project and was afforded ample opportunity for input, the end product was a practical approach which took into account the realistic operational parameters of the plants.

Meanwhile, periodic meetings were held between the CNSC and the IAEA to discuss progress, set implementation milestones, and establish the parameters of the safeguards approaches for the Cameco facilities. For the most part, these discussions took place within the context of general IAEA-CNSC safeguards consultations, although a separate meeting on the implementation of Policy Paper 18 was held in Vienna in March 2005. IAEA participation in this meeting involved not only representatives from Operations B (SGOB), but also Concepts and Planning (SGCP), and Information Technology (SGIT). Inclusion of these other divisions ensured that the implementation project was examined from all angles. In addition, two senior management representatives from Cameco were also invited to attend this meeting. Operator participation was seen as advantageous by all parties, both in terms of demonstrating transparency by providing the IAEA with an immediate link to operational and technical information regarding the Cameco facilities, and in engaging the operator directly in the development of the safeguards approaches.

In addition to the general need to comply with a new IAEA policy, the implementation of Policy Paper 18 in Canada was also seen as a prerequisite to attaining the broader safeguards conclusion, a goal towards which Canada had been working since signing the Additional Protocol in 2000. As a result, it was assigned a high priority and was carried out according to an accelerated timetable: from the official receipt of Policy Paper 18 to the initial physical inventory verifications at the two Cameco plants (signalling full implementation) the entire project took almost exactly two years. Two months later, the IAEA reached the broader positive conclusion for Canada (in September 2005).

3.1. Implementation parameters

In the course of the implementation of Policy Paper 18 there were two fundamental parameters that had to be established before further progress was possible: the precise location of the starting point of safeguards, and the safeguards accountancy structures for the Cameco facilities. These issues are discussed in further detail below.

3.1.1. Starting point of safeguards

In accordance with Policy Paper 18, the production of purified uranyl nitrate at Cameco Blind River was initially investigated as the appropriate first point at which to apply safeguards. This was considered suitable since the uranyl nitrate stream is well-measured by the operator, both in terms of flow and uranium content, on a twice-daily basis. It is moreover a well-established sampling point. It was also considered advantageous in that it was downstream from the production of the calcined and regeneration by-products, which would therefore remove the need to safeguard these two outflows. However, the production of uranyl nitrate is also downstream from the addition to the process of uranium scrap being recycled from the conversion and fuel fabrication plants, which is a safeguarded stream. Since it is not acceptable to have safeguarded material mixing with unsafeguarded material, the production of uranyl nitrate was discounted as an acceptable starting point for safeguards.

The next process point examined, and the one that was ultimately accepted as the starting point, was the addition of UOC to the front end of the production line. Drums of UOC are added to process according to a blend sheet, or formula, which prescribes a given blend based on the impurities of the associated lots of concentrate. These blend sheets are considered source documentation available to the IAEA. This choice of starting point captures the addition of safeguarded uranium scrap to the process and yet avoids the need to safeguard the tens of thousands of UOC drums that are stored in CBR’s yard.

3.1.2. Accountancy structure

While Cameco Blind River is a fairly straightforward plant in terms of safeguards accountancy and was set up from the beginning as a single facility with a single Material Balance Area (MBA), Cameco Port Hope posed more of a challenge. An initial decision was taken to split the site into four facilities (each with a single MBA): the UO2 plant, the UF6 plant, the LEU process line (which was in the proposal phase at the time); and a storage facility for metal and other scrap and waste. This was seen as a logical way to divide the many types of material handled at CPH, which would, in turn, be helpful when reconciling the large inventories involved. However, it also introduced the need to track inventory movements within the site, which was a daunting task given the frequent flow of material among the various plants and storage areas.

After much discussion, this plan was re-visited and CPH was proposed instead as a single facility, with a single MBA. This immediately reduced the number of internal inventory change reports, but introduced the enormous challenge of reconciling all the material on site to a single total for each enrichment category. To date, the operator has been successful in balancing the monthly safeguards ledgers, and it is hoped that this task will grow easier as the new system becomes more familiar.

3.2. Design information and verification

The design information questionnaires (DIQs) for the Cameco facilities were developed through an iterative process, with numerous meetings held with the operator throughout 2004/2005. IAEA feedback on specific aspects of the DIQs was solicited periodically, in parallel with the drafting process, in order to ensure the final product would meet with IAEA acceptance. Preliminary DIQs for

CBR and CNH were submitted to the IAEA in November 2004 for use in on-going discussions concerning the safeguards approaches for these facilities. This was followed by the submission of more complete draft DIQs in February 2005, in preparation for the initial inventory verifications planned for later that year.

Transparency was emphasized throughout this process, with IAEA inspectors given the opportunity during routine safeguards inspections to gather information and tour the plants in order to gain familiarity with the operational processes. Preliminary Design Information Verifications (DIVs) were also carried out at CBR and CPH in April 2005, in advance of the initial inventory verifications.

3.3. Initial inventory verification

The culmination of the project to implement Policy Paper 18 in Canada came with the initial physical inventory verifications (I-PIVs) at Cameco Blind River and Port Hope in the summer of 2005. The project was complicated by the need to coordinate the verification of the in-process inventories with their scheduled shut-downs, which are different for CBR, CPH's UO2 plant and CPH's UF6 plant. Finally, certain material at both sites had to be verified simultaneously with similar material at other facilities in Canada, since safeguards in Canada operates under a Zone Approach, using simultaneous verifications to cover borrowing. As a result of these factors, the I-PIV at each site was split into multiple parts, with book audits used to reconcile the respective inventories to a single date in July 2005.

Another significant challenge faced by the operator in preparing for the I-PIV was the sheer volume of material involved. The situation at Cameco Port Hope was especially difficult, as extensive inventories of historical scrap and waste have built up over many years of operation. An agreement was reached with the IAEA during the I-PIV to focus the verification activities on the more safeguards-significant inventories, although an initial declaration was made for all the material on site. In the meantime, the so-called historical inventories are being pursued on an on-going basis; ultimately, the characterization of this material will be carried out in such a way as to meet the Agency's safeguards requirements, while emphasizing an efficient and practical approach.

The I-PIVs were a cooperative effort, requiring extensive resources from the IAEA, the CNSC, and Cameco. The overall success of the undertaking was due in large part to the flexibility of the Agency, as well as to the cooperation and support of the operator.

4. Future challenges

With the physical inventory verifications that recently took place in July 2006, the two Cameco facilities completed one Material Balance Period under safeguards. While the conceptualization, development, and implementation of safeguards at these facilities as a result of Policy Paper 18 has, for the most part, proceeded smoothly, there are still some points to be addressed. CBR and CPH, as large throughput, long-running facilities that pre-date the application of safeguards, present unique challenges for the IAEA. These challenges include the determination of the appropriate level of effort required to characterize the huge and widely distributed inventory of historical waste and scrap stored on site, as well as to account for the large MUFs that are a natural effect of the scale of operations at the Cameco facilities.

Such situations would have required extensive effort and resources under a traditional safeguards approach. However, in Canada's case, these facility-specific challenges must be viewed in the context of the State-level approach for Canada being pursued as the result of the Agency's broader safeguards conclusion, which was attained, for the first time, in September 2005. In other words, our unique fuel cycle configuration, the inter-linkages between the various elements of the fuel cycle, the extensive information regularly provided to the Agency on the movement of nuclear material through the fuel cycle, and the verification activities undertaken by the Agency within Canada in any given year, will enable the IAEA to maintain the conclusion on the non-diversion of declared nuclear material and the absence of undeclared nuclear material and activities. In short, the Agency is able to use the tools available under the State-level approach for Canada in order to effectively apply safeguards to the Cameco facilities, despite the challenges they present.

5. Conclusion

The implementation of Policy Paper 18 in Canada was carried out with an emphasis on collaboration, cooperation, and transparency. All three parties – the CNSC, the IAEA, and Cameco – worked together in determining key aspects of bringing the Blind River and Port Hope facilities under safeguards. Multiple meetings were encouraged, with wide participation, to promote open communication and direct involvement in the decision-making process. The end result was a realistic, practical strategy for applying safeguards to Cameco's Blind River refinery and Port Hope conversion facility within a relatively short period of time, in the context of the IAEA's broader safeguards conclusion for Canada. Moving forward we will continue to work with the IAEA in realizing the efficiencies implicit in the State-level approach stemming from this conclusion, in particular its application to the Cameco facilities.

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Evaluating the decommissioned status of a LWR and RRCA facility to determine level of effort needed to safeguard facility

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Abstract. The International Atomic Energy Agency (IAEA) conducts comprehensive design information verifications (DIV) to affirm the declared status of a facility. When the IAEA verifies the conditions at a facility that is revising its status, it needs to have a technical basis for confirming that the facility has changed its declared status. There is little difficulty in verifying a change of status from "operating" to "closed-down" since that is mainly a matter of verifying that nuclear material has been removed from the facility. This is more of an inspection activity than a DIV. The IAEA has found that verifying a change in status from "closed-down" to "decommissioned" is more complicated. The distinction between a “closed-down” and “decommissioned” status is important because a decommissioned facility is subject to less stringent safeguards measures (actually no measures under traditional safeguards), so the Agency must be confident that the decommissioned facility cannot be restarted and misused (e.g., no inspections if decommissioned; complementary access (CA) possible in a state with the Additional Protocol in force).

INFCIRC/540 defines "Decommissioned" as meaning that residual structures and equipment essential for the facility's use have been removed or rendered inoperable [1]. The IAEA has compiled lists of "essential equipment" for major facility types to aid inspectors in DIV inspections [2]. However, the definition in INFCIRC/540 does not state how much of the essential equipment must be removed or rendered inoperable in order to consider the facility as a whole to be decommissioned. A facility may be “mothballed” with many pieces of nuclear steam supply system (NSSS) equipment still in place. The IAEA desires to develop guidelines to identify the essential equipment that must be removed or rendered inoperable for major facility types in order to classify a facility as decommissioned for safeguards purposes. The IAEA wants to compile factors such as the time, cost, and practicability of re-activating a facility as compared to building a new one elsewhere. Inspectors can use these guidelines when confirming that a facility is "rendered inoperable”.

The authors will be conducting a field study to identify criteria that the Agency can use to confirm the decommissioned status of a facility. The authors will investigate two facility categories: Light Water Reactors (LWRs) and Research Reactor and Critical Assemblies (RRCA). Three RRCAs located at Brookhaven National Laboratory will be included in the study. These facilities include a graphite-moderated natural and enriched fueled reactor, a small water-cooled LEU fueled reactor, and a D_2O moderated and cooled HEU fueled reactor which provide a diverse group of RRCAs. As regards commercial LWRs, both BWR and PWR types in the U.S. will be considered in the study.

This paper will address our preliminary recommendations for identifying criteria to verify the decommissioned status at both the RRCAs and LWRs. We will cover the safeguards effort needed to insure that a facility is not reconstituted and material processed before the IAEA can detect this undeclared activity in a timely fashion.

1. Introduction

The International Atomic Energy Agency (IAEA) has a special interest in conducting comprehensive design information verification (DIV), including verification of closed-down or decommissioned status of a nuclear facility. When the IAEA verifies a facility that is revising its status, it needs to have a technical basis for confirming that a facility changed its declared status. There is little difficulty in verifying a change of status from "operating" to "closed-down" since that is mainly a matter of verifying that nuclear material has been removed from the facility (except for residual contamination). The IAEA has found that in practice verifying a change in status from "closed-down" to "decommissioned" is more complicated.

2. Background

INFCIRC/540 defines "Decommissioned" as meaning that residual structures and equipment essential for the facility's use have been removed or rendered inoperable. However, the IAEA has found that the definition in INFCIRC/540 does not state how much of the essential equipment must be removed or rendered inoperable in order to consider the facility, as a whole, is decommissioned. A facility may be wholly demolished or only “mothballed” with many pieces of nuclear steam supply system (NSSS) equipment still in place. The IAEA needs assistance in developing guidelines on how much and which pieces of essential equipment must be removed or rendered inoperable for major facility types in order to accept that a facility is decommissioned for safeguards purposes. The IAEA wants to compile factors such as the time, cost, and practicability of re-activating a facility as compared to building a new one elsewhere. The IAEA also needs guidance in interpreting the term “rendered inoperable”.

The IAEA has requested that studies of different real cases of decommissioned facilities be conducted starting with the facility categories Light Water Reactors (LWRs) and Research Reactor and Critical Assemblies (RRCA). The IAEA desires operator interaction because of gaining their knowledge on the decommissioning process. The right to conduct DIVs is central to the IAEA ability to confirm that a facility has been decommissioned, as declared by the operator. Furthermore, the right to conduct complementary access (CA) is essential to the IAEA ability to confirm the decommissioned status of a facility. The IAEA’s right to conduct DIVs is stated clearly in INFCIRC/153. The right of the IAEA to conduct CA is stated under INCIRC/540.

3. Preliminary Research Reactor Case Studies – Three BNL RRCA Facilities

BNL has three shutdown research reactors in various stages of decommissioning and decontamination (Figure 1). We will describe the reactors, decommissioning and decontamination tasks, reconstitution and misuse scenarios, and possible safeguards approaches to detect misuse scenarios. We conducted preliminary interviews with BNL Department of Energy (DOE) site staff and BNL staff knowledgeable about the three reactors and the decommissioning projects on 6-7 June 2006. From this set of interviews we were able to understand from actual cases the ease or difficulty of reconstituting a shutdown research reactor. As will be seen, a mixture of technical, economic, and political factors determine a facility’s final fate and disposition. The IAEA will need to understand the range of these factors not just at the facility level but on a State Level Approach (SLA).

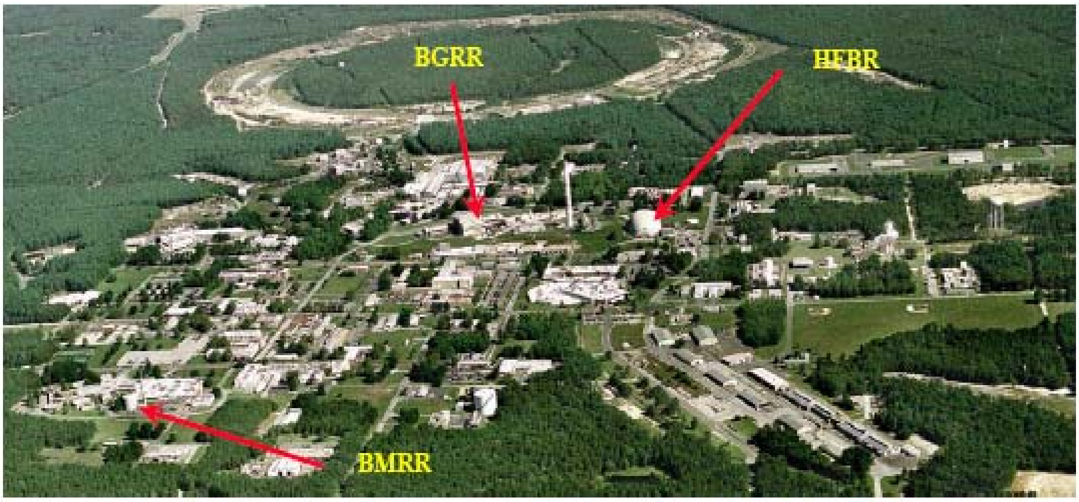

FIG. 1. BNL Research Reactors.

3.1. *Brookhaven Medical Research Reactor*

3.1.1. Description

The Brookhaven Medical Research Reactor (BMRR) is a 5 MW thermal power, light water moderated and cooled, tank-type MTR reactor with curved-plate fuel elements made of a uranium-aluminum alloy containing 12% (by weight) LEU (Figure 2). The BMRR was the first nuclear reactor built exclusively for medical research. It produced neutrons in an optimal energy range for a promising experimental treatment for a type of brain cancer. The BMRR operated from 1959 to 2000. It will be undergoing decommissioning at some time in the future as determined by aforementioned technical, economic, and political factors.

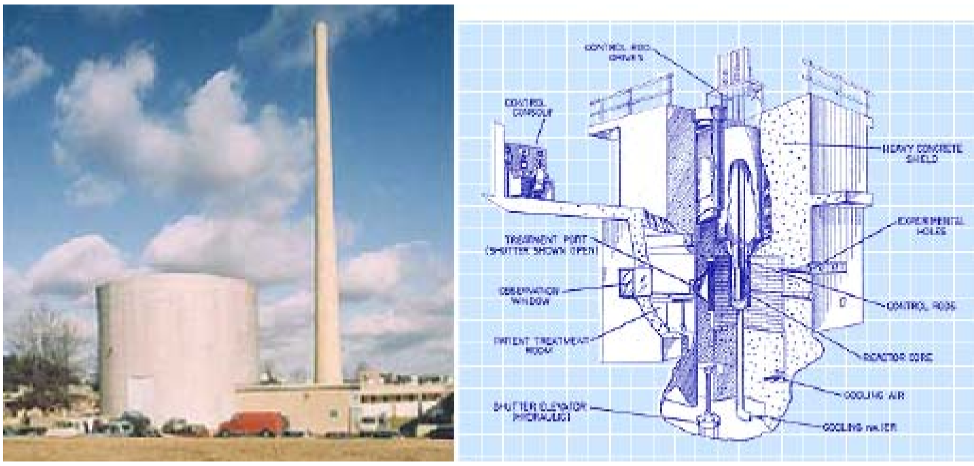

FIG. 2. BMRR - Exterior View and View of the Reactor Schematic.

3.1.2. Present Condition of the Facility

BNL removed the fuel, control rods, and control rod drives rendering the reactor inoperable (Figure 3). However, all the cooling systems are available and the facility is kept clean and secure. With sufficient funds the reactor could be reconstituted and misused since new fuel and control rods and control rod drives could be obtained. We will research in more detail the cost in money and time. However, the size and design of BMRR precludes any significant unreported plutonium production. The proliferation scenarios for this reactor would include: (1) reinstalling fuel and reactivating the reactor and (2) installing an alternate core to produce fissile material in significant quantities.

FIG. 3. BMRR Core – Operating State and in Decommissioned State.

3.2. High Flux Beam Reactor

The High Flux Beam Reactor (HFBR) is a heavy water moderated and cooled, 30-60 MWth reactor using HEU fuel (Figures 4 and 5). It operated from 1965 to 1999 providing neutron sources for various scientific and engineering research projects at BNL.

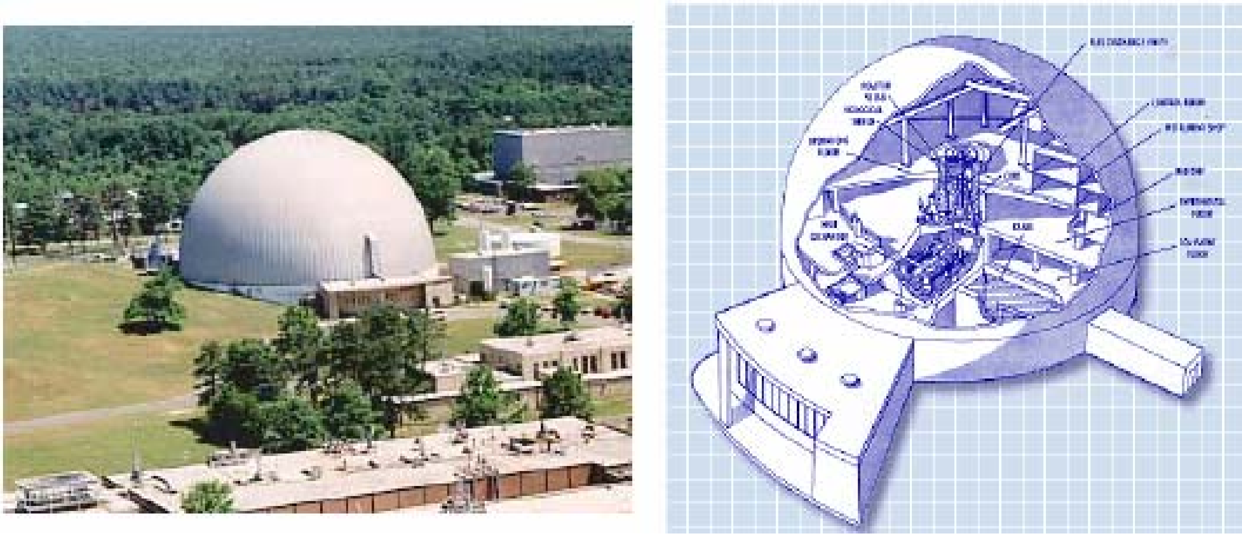

FIG. 4. HFBR - Exterior View and View of the Reactor Schematic.

FIG. 5. Top of HFBR Core in Present Decommissioned State.

3.2.1. Present Condition of the Facility

HFBR is shutdown with the fuel removed. The control rod blades have been dismantled and kept at HFBR since they are highly radioactive containing on the order of 400,000 Cu. Decommissioning is in planning phase for HFBR. Various decommissioning and decontamination (D&D) scenarios have been proposed and we plan to analyze them for cost benefit analysis so that the Agency can see the D&D strategies that a Member State could employ. With sufficient funds the reactor could be reconstituted and misused since new fuel and control rods and control rod drives could be obtained. Furthermore, there had been a serious proposal to replace the HFBR reactor vessel with a 2-3 GeV accelerator. This would provide a proliferator with the possibility to divert the beamline from the facility across the street to a clandestine spallation neutron source set-up with capabilities to breed ^{233}U from thorium or plutonium from ^{238}U. Depending on the power and current of such an accelerator, the "would-be" proliferator could produce significant quantities of weapon-usable nuclear material [3].

3.3. Brookhaven Graphite Research Reactor

The Brookhaven Graphite Research Reactor (BGRR) is an air-cooled graphite-moderated reactor that was capable of running on natural or low-enriched uranium (LEU) that operated at a nominal thermal power level of 28 MWth (Figure 6). BGRR operated from 1950-1968. The air-cooling system exhausted the BGRR cooling air up a 100m tall stack. This cooling system used ductwork lying below ground and above ground to provide a path for the heated air to the stack (Figure 7). In the later stages of operation it used LEU fuel.

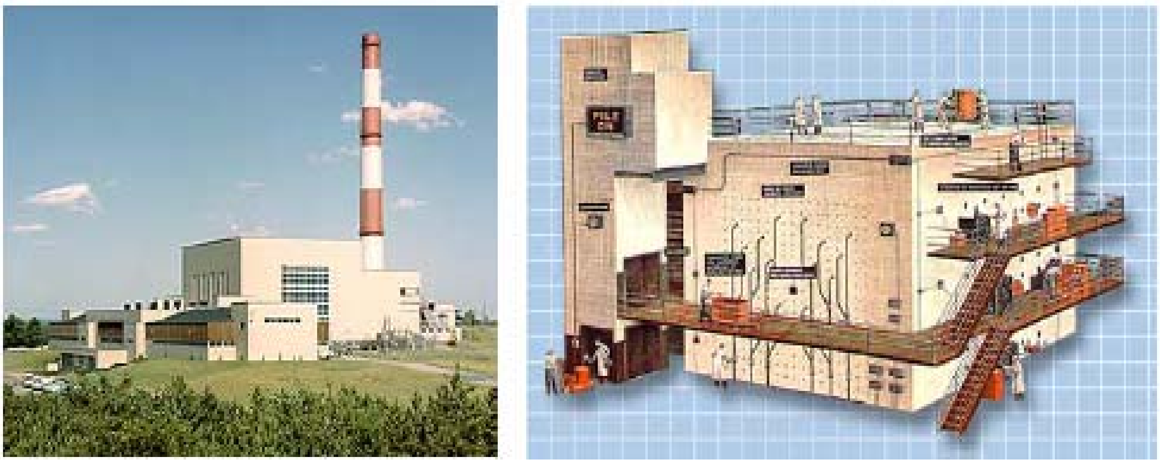

FIG. 6. BGRR - Exterior View and View of the Reactor Pile.

FIG. 7. BGRR- Path of Underground Ducts to Stack (Blue Arrow).

3.3.1. Present Condition of the Facility

BGRR ceased experimental use on June 10, 1968. BNL implemented a program to minimize the amount of radioactivity within the biological shield that surrounded the reactor pile, and to ensure that no residual radioactive contamination could escape from the shield. After the shutdown, BNL removed all of the fuel rods and shipped them to the Department of Energy's Savannah River facility. Most of the radioactive materials, including all of the spent fuel rods, were shipped off site by the end of 1972. Also in 1972, Brookhaven disconnected the borated steel control rods from their drives and permanently inserted them into the reactor pile. This action plus the fact that some fuel rods are probably stuck in the gap in the center of the graphite pile makes refurbishing the graphite pile an extremely difficult if not impossible task.

Brookhaven has already completed a large decommissioning and decontamination task at BGRR. Reconstituting and misusing BGRR would be possible only by rebuilding the facility from the ground up at this point since the fuel has long since been removed and the graphite pile is under decontamination and removal activities. The graphite pile will be removed as part of this environmental remediation. BNL has done the most decontamination and decommissioning work at BGRR compared to the other two RRCAs at BNL. The only reasonable proliferation scenario at BGRR is rebuilding a production reactor in the BGRR building after the removal of the present pile. A proliferator would also need to build a new air cooling system to connect to the present stack as the underground ducting has been decontaminated and removed in sections leading from the BGRR to the stack. All of these efforts would not be trivial.

4. Planned Power Reactor Case Studies

The authors will be traveling to the Shoreham Nuclear Power Plant (Shoreham, NY), the Indian Point Nuclear Power Plant (Buchanan, NY), and the Connecticut Yankee Nuclear Power Station (Haddam Neck, CT). These plants will represent both PWR and BWR types of LWR seen in the USA. Shoreham is one of only four US power reactor facilities totally decommissioned. Indian Point 1 Plant is in SAFSTOR (United States Nuclear Regulatory Commission (USNRC) definition: “a method of decommissioning in which the nuclear facility is placed and maintained in such condition that the nuclear facility can be safely stored and subsequently decontaminated to levels that permit release for unrestricted use”) and Connecticut Yankee plant is in DECON (USNRC definition: “all components and structures that are radioactive are cleaned or dismantled”). Therefore, these three plants show a wide range of both decommissioning activity and operator decommissioning philosophy. This will provide three very different case studies yielding a good basis for creating guidelines for decontaminated reactor safeguards.

4.1. Shoreham Nuclear Power Plant

The Shoreham Nuclear Power Station (SNPS) in Shoreham, NY was a Boiling Water Reactor (BWR) with a net output of 809 MWe. It was permanently shutdown as of May 1989. It ran for only a total of 2 effective full power days before being shutdown by political pressures. Due to its short operating period and low power history, the Shoreham site contained virtually no environmental contamination, or contamination outside of three major structures and their systems (spent fuel pool, dryer separator pit, and reactor cavity). The balance of the site and support structures remained intact with turbine facilities and diesel generators still functional right after the shutdown. In the 1990’s, the operator, Long Island Lighting Company (LILCO), totally decontaminated the site. LILCO salvaged all equipment from the site it could and removed it from the site. This included shipping the very low burnup spent fuel to Pennsylvania for use at the Limerick Generating Station, a similar BWR.

4.2. Indian Point Nuclear Power Plant

The Indian Point 1 Nuclear Power Plant in Buchanan, New York was a Westinghouse PWR rated at 250 MWe which operated from 1962 to 1974. The operator removed the spent fuel from the reactor and put the facility in the SAFSTOR. When the operator decommissions the Indian Point 2 reactor then Indian Point 1 will be decommissioned. Hence, economic and political factors as well as technical factors determined the fate of Indian Point 1.

4.3. Connecticut Yankee Nuclear Power Station

Connecticut Yankee (CY) located at Haddam Neck, CT was a 600 MWe PWR. The Connecticut Yankee Atomic Power Company Board of Directors voted to permanently close the CY plant in December 1996. Connecticut Yankee based the decision on an economic study that concluded that electric customers would, because of changing market conditions, save money if the plant was closed.

CY chose immediate dismantlement (the DECON method) because it is the most practical and environmentally responsible option for this plant. In DECON all components and structures that are radioactive are cleaned or dismantled, packaged and shipped to a low-level waste disposal site, or they are stored temporarily on site. Once the operator completes this task, which takes five or more years, and the USNRC terminates the plant's license, that portion of the site can be reused for other purposes. CY felt that the use of the present organization that is trained and knowledgeable about the facility, avoidance of long-term maintenance costs, and the availability of low-level waste disposal facilities made DECON the best decommissioning mode for the facility. CY began significant decommissioning activities in May 1998.

5. Preliminary Analysis of Decommissioned Status of RRCAs

During discussions with the operators and DOE personnel knowledgeable about the operation and disposition of the three BNL research reactors and even LWRs, we came to the following preliminary conclusions about what the IAEA safeguards department can determine regarding a facility in a decommissioned state or a closed-down facility being decommissioned.

1. The size of the core barrel is important. It indicates the power that can be generated and therefore the production capacity. This factor needs evaluation for safeguards concerns.

2. The operational status of the HVAC systems appears to be an important consideration on how easily the facility can be reconstituted. BNL managers reported that facilities that are placed in SAFSTOR can experience severe degradation of facilities and support structure due to molds and bacteria. The facility can become so unhealthy to work in within a few years that it would slow the ability of an operator to reconstitute the reactor.

3. A combination of economic and political factors influences the final status of a facility. The economics drove the USNRC to solve the problem by creating the SAFSTOR mode. This intermediate state allows radiation levels to decay down to the point where contaminate areas of the plant can be more easily and inexpensively disposed. Political factors, such as the desire of the local community to return the site to a natural state, will drive the decisions makers to return to a “greenfield” condition. The IAEA will have to evaluate the norms for a State in evaluating a specific facility’s status. If the end state of a facility is not compatible with the State’s nuclear program and the economics of that program, the IAEA may want to carry out more inspections and allocate more analytical resources to that facility so that an undeclared reconstitution of the facility can be detected.

4. The BNL staff and DOE site office felt that techniques such as incapacitating the reactor or support systems will have only a temporary effect in stopping the return of a reactor to operational status. However, these actions will affect the cost and time scale of reconstituting the reactor.

5. DIV will be required to ascertain the physical condition of the site. From the authors’ preliminary investigation, the frequency of the DIVs could be dependent on several factors, such as: 1) time since shutdown, 2) level of routine maintenance performed, 3) status of

HVAC systems, 4) existence of support services such as electricity, ‘command and ’control to control room systems, and 5) ability of a State to have a supply of NSSS equipment and nuclear fuel.

6. The use of a State Level Approach (SLA) is desirable for determining the decommissioned status of a reactor and the possibility of its restart. SLA gives the IAEA the ability to nuance its conclusion about a facility’s status with the use of open source material about the facility and the State nuclear program. This approach allows for a more quantitative and non-discriminatory analysis of the decommissioned status of a reactor.

6. Recommendations

The authors will be compiling recommendations for safeguards guidelines for decommissioned and decontaminated RRCA and LWR facilities. The case studies will include costs of decontamination and reconstitution. It is hoped that this data will shed light on the economic liability of an operator to maintain a closed-down plant in a preservation state (clean and recoverable state). Therefore, the IAEA can assess the reasonability of a State declaring a facility to be decommissioned but kept in a “Standby” condition. Such behavior may be deemed suspicious and then would warrant more frequent DIVs or other measures to verify the closed-down state of the reactor. As stated above, the use of SLA will help insure that the IAEA has a broader and more complete picture of a facility seen in the light of the entire State nuclear program. We also hope to get realistic data on the time to refurbish a reactor that is being decommissioned. The combination of all this data will provide the IAEA with not only a list of essential equipment but scenarios and the likelihood of reconstituting a decommissioned plant or a closed-down plant being decommissioned. Hence, the IAEA will receive compiled factors such as the time, cost, and practicability of reactivating a facility as compared to building a new one elsewhere. The IAEA will also receive guidance in interpreting the term "rendered inoperable” from regulator and operator standpoints.

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Down blending high enriched uranium in Kazakhstan

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Abstract. At the end of the 1990s the BN-350 reactor in Kazakhstan was permanently shut down. Unused unirradiated fuel assemblies were transported to the Ulba fuel fabrication facility in Ust-Kamenogorsk, Kazakhstan. The fuel contained high enriched uranium (HEU) with enrichment values of up to 26% ^{235}U and some depleted uranium (DU). To achieve the goal of reducing the inventories of HEU within the State, the Government of Kazakhstan arranged to have this fuel down blended at the Ulba facility. A separate material balance area (MBA) was designated, which covered the rooms where the down blending activities took place. Inside a glove-box line HEU dioxide pellets were removed from the fuel rods and converted to U_3O_8 powder by oxidizing them in air at high temperature. HEU powder was mixed with low enriched uranium (LEU) powder and finally dissolved so that the resulting LEU solution could be introduced to the parts of the Ulba plant that were certified for processing LEU with enrichment of less than 15%. The safeguards approach was based on utilizing standard containment and surveillance (C/S) measures, on-site non-destructive assay (NDA), sampling for destructive analysis (DA) and continuous inspector presence during active processing. Finally, dissolution of the mixed powders was observed to provide in-depth assurance that LEU could not be substituted to conceal diversion of HEU. This work demonstrated that the IAEA with the cooperation of the facility operator could provide assurance with a high confidence regarding the non-diversion of direct-use nuclear materials, as well as of LEU and DU during processing while simultaneously applying measures of material accounting, DA, NDA and C/S, and using safeguards inspection resources most efficiently. The IAEA has gained a great deal of experience to extend this model to future opportunities for downgrading HEU materials that are not covered under ongoing materials disposition programmes.

1. Introduction

Through the Global Threat Reduction Initiative (GTRI) [1], which was launched by the United States Department of Energy, assistance was provided to the Government of Kazakhstan that is working to eliminate unused weapons-usable HEU within its borders. In this programme, unused fresh fuel at the shutdown BN-350 liquid-metal-cooled fast-breeder reactor at Aktau was transported to the Ulba Metallurgical Plant in Ust-Kamenogorsk where HEU from the fuel was down blended to LEU in a campaign that began with a test run in March 2005 and concluded in February 2006. Since the down blending campaign was expected to be completed in a moderate time period, the IAEA determined that the most cost effective approach would be to use standard surveillance equipment, on-site NDA measurements, random sampling of nuclear material for DA and continuous inspector presence during active processing. Specially designed equipment (other than the IAEA's standard optical surveillance system) to enable unattended or remote monitoring would not have been cost effective for a simple, un-automated hands-on process. This paper discusses the essential elements of the IAEA safeguards approach and the cooperation between the facility operator and the IAEA that provided unimpeachable assurance of the elimination of weapons-usable material with acceptable impact on processing operations. Extension of this approach to future opportunities for elimination of weapons-usable materials, possibly without continuous inspector presence, is discussed.

2. Process description

Uranium enrichment in most parts of the Ulba plant is limited to less than 15% to ensure criticality safety. Usually the enrichment of the uranium being processed does not exceed 5%. Therefore the HEU oxide removed from the BN-350 fuel was blended by dry mixing with LEU powder to create batches of mixed powder that could be dissolved and processed in the main parts of the Ulba plant for manufacture of LEU fuel products. HEU was processed, as described below.

The IAEA had verified the uranium content and enrichment of the unirradiated fuel assemblies at the reactor site before they were transferred under IAEA seal to the Ulba plant. There they were stored under IAEA seal while awaiting processing. To initiate processing at Ulba, the IAEA verified and removed seals and released assemblies to the facility operator one-at-a-time so that no more than three assemblies, together containing less than one significant quantity (25 kg ^{235}U in HEU), were being processed at any time. In an initial process step carried out near the storage area, the top part of the fuel assembly was cut off without releasing any uranium-containing material to free the fuel rods, which were then transported to the HEU-processing line in criticality-safe containers. Processing of HEU was carried out in a line of nine glove boxes, shown in Figure 1. All workstations in the glove box line had transparent glass surfaces and, in some cases, access ports for manual operations so that processing manipulations could be performed from both sides and the IAEA staff could observe processing operations from both sides.

At the first workstation, a fuel rod containing HEU was fed through a fixture in the side of the glove box. There the operator cut off the end plug and tapped out the fuel pellets into collection trays and cut the cladding, as necessary, into short sections to facilitate removal of the fuel pellets (Figure 2). All fuel rods had pellets made of DU at both ends and HEU pellets in the center section. HEU pellets were annular, and DU pellets were solid. The operator segregated the pellets visually. The IAEA verified the enrichment of HEU pellets by placing five randomly selected pellets confined by a geometry-determining harness on top of a sodium-iodide gamma-ray detector calibrated for this configuration. For this verification, out of 42 trays that were filled with HEU pellets by an individual assembly, five trays were randomly selected for sampling of five pellets from each chosen tray. Empty cladding segments and DU pellets were removed from the workstation. The IAEA verified that no HEU was contained in materials removed from the glove boxes. From each assembly five DU pellets were selected three times for measurement by the sodium-iodide detector from the can in which the DU pellets were collected.

The HEU pellets were placed one layer deep in straight-sided uncovered stainless-steel trays, loaded into ovens, and oxidized in air at high temperature. This process transformed the pellets into loose oxide powder (U_3O_8). During the first weeks of the down blending campaign, the operator optimized the oxidation process with regard to temperature, amount of material per tray and time so that two loadings of each of the four ovens could be processed per eight-hour shift. The number of ovens allocated for the process determined the throughput of the entire down blending process.

Approximately 100 stainless-steel cans were available for mixing HEU and LEU powders. The appearance of an empty stainless-steel can is illustrated in Figures 3 and 4. Unique identifying numbers were stamped on the lid and body of each can. The tare weights of these cans, verified by the facility operator and the IAEA, were recorded and did not change during use.

Two trays of HEU oxide (U_3O_8) powder were emptied into a single stainless-steel can. The cans of HEU powder were removed from the glove box, weighed and placed in a storage and measurement area controlled by the IAEA. The IAEA verified the ^{235}U content of every can of HEU powder using an active well neutron coincidence counter (AWCC) and verified the gross weight of each can. Because of the relatively high purity and known stoichiometry of the HEU oxide, the uranium content could be determined from the net weight with sufficient accuracy. After verifying the HEU contents, the IAEA kept each can under containment by applying seals and/or camera surveillance, as necessary.

The facility operator loaded a number of stainless-steel cans with a calculated amount of LEU oxide (U_3O_8) such that, after the transfer of the contents of a can of HEU powder (containing two oven trays of HEU powder), the mixed powder would have enrichment in the 8-10% range. In the presence of an IAEA inspector HEU powder was added to LEU powder inside the glove box at the end of the process line. The IAEA verified not fewer than five cans from each shipment of LEU introduced by the facility operator for the purpose of blending with HEU. Since all these materials were of high purity (by accounting standards), the facility operator could apportion by weight the quantities to be mixed. The transfer of HEU from cans previously verified by the IAEA to the cans containing the pre-weighed LEU was carried out at the mixing station using funnels. The mixing area of the glove box was swept frequently. Sweepings were not accumulated; rather they were added to the can that was currently being filled. After emptying each can of HEU into a can containing LEU, the IAEA was given the opportunity to verify visually that no HEU remained in the can that had just been emptied.

The cans of mixed oxide were removed from the glove box, weighed, and both the operator and the IAEA recorded the weights. After weighing, the IAEA sealed each can containing product powder and released it to the facility operator for homogenization. Blending was accomplished by rotating the cans for half an hour about an axis perpendicular to the can's axis of cylindrical symmetry causing the contents to tumble vigorously. The cans of blended powder were placed in an IAEA-controlled room for verification of the ^{235}U content by the IAEA using the AWCC (Figure 5). The enrichment of the blended powder was calculated from the measured U^{235} content and the net weight. The operator moved the cans of mixed oxide that had been verified (100% by NDA and one DA sample taken per assembly) and sealed by the IAEA to the LEU storage area located near the loading station for the dissolver.

At the dissolution station cans of blended powder were transferred through chutes into a vertical section of stainless-steel pipe located immediately below the glove box and having sufficient volume to hold five to six cans of blended uranium powder. Nitric acid was then circulated through the pipe to dissolve the U_3O_8. One section of the acid circulation system was made of transparent glass pipe. It was possible to observe the initial dark fluid (due to the suspended U_3O_8 powder) and its gradual transition to transparent greenish-yellow-coloured liquid as the uranium oxide dissolved. For material accounting purposes, the blended powder was transferred from the HEU MBA to the LEU MBA at the blending station using the accounting data for the blended powder. As a further measure to assure that the dissolved nuclear material had an enrichment as declared, six samples for destructive analysis were randomly taken from the solution tank over the period of the entire campaign.

3. IAEA safeguards approach

The principal IAEA safeguards concern was that materials diverted elsewhere from the Ulba plant could have been substituted for or used to produce material similar to the blended HEU/LEU product to conceal the diversion of HEU. One source of material that could conceivably have been used to produce uranium with enrichment in the 8-10% range was the BN-350 fuel containing uranium enriched to 17%. However, all the fuel assemblies at Ulba with enrichment of 17% were under IAEA seal and thus could not have been processed elsewhere in the facility to obtain material that could have been used to conceal diversion of HEU.

During processing, all HEU material was under direct observation by inspectors supported by the recorded-video surveillance system. After the HEU pellets were oxidized, the resulting HEU U_3O_8 powder was transferred to cans that would be used in the mixing and down blending processes. The IAEA verified 100% of the HEU before mixing and 100% of the mixed HEU/LEU powders, and a statistically determined fraction of the cans of LEU before mixing with HEU. During verification all cans with HEU were under exclusive control of the IAEA. The IAEA kept these cans under containment by applying seals and/or camera surveillance, as necessary, before they were returned to the facility operator. The facility operator also sampled batches of HEU and mixed HEU/LEU powder selected by the IAEA and prepared the samples for shipment to the IAEA for destructive analysis to determine enrichment and uranium content at the IAEA's Safeguards Analytical Laboratory (SAL). The sampling process and the sample preparation activities were observed by IAEA inspectors. The measurements at SAL were part of the verification concept and provided backup but were not sufficiently timely for evaluating the material balance in real time at the facility.

HEU from different fuel assemblies was never mixed during processing. Both the operator and the IAEA calculated a material balance for each fuel assembly. As noted, the IAEA measured 100% of the cans with HEU powder and 100% of the cans of blended HEU/LEU powder and verified the LEU used for blending according to a sampling plan. From these measurements the IAEA could calculate the material balance completely independently of the facility operator.

The IAEA compared its measured data and calculated the material balance with the data made available informally by the facility operator immediately upon completion of the processing (but before dissolution) of the material from each fuel assembly. Since the use of data formally transmitted by the State system of accounting for and control of nuclear material (SSAC) to the IAEA would not have been sufficiently timely for this purpose, the IAEA used the facility operator's declared data with the advantage that any discrepancy between the operator data and the IAEA measurements results could have been investigated immediately. An out-of-control situation could have been detected and would have been corrected before it could have affected material from more than a single fuel assembly, which contained less than one third of a significant quantity of HEU.

Using a standard digital multi-camera surveillance unit with eight cameras (Figure 6), the IAEA monitored the following: all entrances and exits to the HEU MBA, both sides of the HEU glove box process line, the work area of the HEU process area where cans of mixed powder were rotated to ensure blending, the scales area where weighing was carried out, the IAEA's work area including the intermediate storage for HEU cans when unattended by inspectors and the path from the mixed-powder storage area to the dissolver-input glove box. Fixed interval triggered surveillance monitored the process areas during periods when the facility was not operating. Surveillance review was performed using motion-activated triggering.

The IAEA and the facility operator agreed that the IAEA would have the opportunity to verify the contents of any can with a capacity of half a liter or greater that was brought into or taken out of the HEU MBA by the facility operator without an attached IAEA seal. During working hours, the IAEA verified can movements visually. Optical surveillance provided assurance that cans were not moved during non-working hours and supplemented visual observations during working hours. The IAEA routinely reviewed the surveillance records every three days. Safeguards assurance regarding removal of HEU and substitution of other materials to conceal diversion of HEU resulted primarily from the IAEA's measurement of 100% of the items in the HEU and mixed-powder strata, its measurement of the material balance for each fuel assembly entirely independently of the facility operator, as well as that all HEU or mixed oxide materials which were either under direct physical control by IAEA personnel or were under IAEA seals when not under direct control of inspectors. Thus the optical surveillance system provided backup rather than primary assurance for scenarios involving potential diversion of HEU and substitution of other materials to conceal diversion of HEU. Together, the IAEA's material-accounting measurements, direct control of HEU by IAEA inspectors, IAEA seals on HEU and all mixed powder LEU containers and continuous optical surveillance of the HEU processing areas provided robust, redundant assurance that there had been no mechanical re-separation of the mixed powders and that all HEU removed from the fuel assemblies had been accounted for and irreversibly converted to LEU materials unsuitable for nuclear explosives.

4. Cooperation between IAEA and facility operator

The cooperation between the IAEA and facility operator was exceptional and unique throughout the duration of this project. The facility operator agreed to operate the process for only two shifts per workday, five days per week so that the IAEA could carry out its safeguards activities during operating hours with three inspectors. Facility and IAEA personnel conferred at the beginning of each work day or one day in advance to coordinate processing activities and verification activities, respectively, so that the IAEA would have the opportunity to measure 100% of certain process flows without holding up processing.

The IAEA established the material balance for each fuel assembly independently rather than verifying the operator's measurements according to a sampling plan. The facility operator agreed not to mix materials from multiple fuel assemblies in the HEU MBA and presented its material-balance data for each fuel assembly informally to the IAEA inspectors at the conclusion of the processing for each fuel assembly. The data was later formally transmitted to the IAEA through the SSAC.

In total, during the HEU down blending campaign the following nuclear material verification activities were performed:

(a) 440 uranium enrichment measurements of HEU pellets using a miniature multichannel analyzer (MMCN);

(b) 200 uranium enrichment measurements of DU pellets using a MMCN;

(c) 1550 uranium enrichment measurements of HEU powder in cans by an AWCC (100% of the items);

(d) 1550 uranium enrichment measurements of mixed product powder in cans using an AWCC;

(e) 100 samples were taken for DA from mixed product, HEU pellets, HEU and LEU powders and solutions, which were sent to the IAEA for chemical analysis; and

(f) 1750 seals were verified that were attached to cans in the HEU MBA and that were detached at the LEU MBA, not mentioning the verification of seals, which were applied at and removed from surveillance devices, doors and samples.

The facility operator supported the IAEA's measurement plan that included 100% verification of HEU and down blended product powder cans. This support resulted in greater effort and time to assist the IAEA inspector in handling the cans. The 100% verification provided an optimum balance between campaign costs and confidence regarding the non-diversion of nuclear material.

5. Safeguards without continuous inspector presence

To verify the material balance reported by the facility operator, the IAEA must have the opportunity to verify the flows and inventories associated with each MBA in accordance with the IAEA safeguards criteria. A physical inventory verification (PIV) is carried out at least annually. The IAEA verifies flows into and out of the MBA at interim inspections carried out frequently enough to satisfy the timeliness criteria for the types of materials present. For materials that are directly usable in weapons, such as HEU, the timeliness criteria require the monthly verification of the nuclear material inventory with medium detection probability for gross and partial defects. The IAEA and the facility operator negotiate an inspection plan which specifies how long the facility operator will hold materials for inspection and how frequently the IAEA will carry out inspections to satisfy the timeliness criteria.

The continuous presence of inspectors in the HEU processing area of the Ulba plant assured that the holding time for HEU-containing materials could always be less than one day and enabled the IAEA to carry out verification measurements on 100% of the HEU-containing items. Because of the high purity and uniform composition of all the important process materials, measurements of the weight and the ^{235}U content by an AWCC, supplemented by enrichment measurements using gamma-ray techniques, the IAEA measured its own material balance for each fuel assembly, rather than verifying the operator's reported material balance according to a sampling plan. The IAEA and the facility operator compared their material-balance data for each fuel assembly as processing was completed and promptly resolved any differences that might have occurred. The material being processed in the HEU MBA at any time contained less than a significant quantity of HEU. The independence of the IAEA's material accounting and the opportunity to place all HEU materials under IAEA seal when not under direct control of IAEA inspectors reduced the reliance placed on the optical surveillance system to detect the possible substitution of materials that could conceal diversion of HEU.

An approach without continuous inspector presence would be feasible by putting more emphasis on containment and surveillance (C/S) measures while performing the nuclear material verification activities using NDA and DA on the basis of a random sampling plan and testing for bias, partial and gross defects in accordance with the safeguards criteria [2]. With the cooperation of the facility operator, tradeoffs might be made that would enable the IAEA to provide an acceptable level of safeguards assurance without continuous inspector presence. The IAEA will explore tradeoffs that might be applicable and appropriate in future opportunities for elimination of weapons-usable materials.

With less frequent presence of inspectors, process materials would have to be held longer to provide opportunity for verification by the IAEA. The amount of in-process material would increase, and more cans would be needed. At Ulba about 100 tared stainless-steel cans were needed to accommodate sequential but simultaneous processing of material from three fuel assemblies. More cans would be required to hold additional quantities of oxidized HEU powder, pre-weighed quantities of depleted, natural or low-enriched uranium (DNLEU) powder for dry blending, and mixed DNLEU/HEU powder. A team of IAEA inspectors would have to verify these materials according to a sampling plan at scheduled times. The IAEA would verify the oxidized HEU powder, witness the dry blending, and seal the cans of mixed DNLEU/HEU powder. Rather than sealing each can individually, the cans of mixed powder could be placed under temporary video surveillance to maintain continuity of knowledge during the absence of inspectors at the facility until the dissolution of material from each can could be witnessed. Dissolution, which is the key irreversible step of the verification process, must be carried out in the presence of inspectors. The availability of more than one dissolver would avoid a bottleneck at this step. (Two dissolvers were available at Ulba although only one was used at any time.)

In effect, the IAEA and the facility operator would work together to dissolve a large amount of mixed powder and collect a correspondingly large amount of LEU solution in a short time. Transfer of material from the HEU MBA to the LEU MBA was based on the accounting data for the mixing of the powders so that additional measurements were not required at the time of dissolution. If dissolution of mixed powder from all cans of mixed powder could not be completed during the inspection visit, the remaining cans of mixed powder could be sealed and held until the next inspection visit when the IAEA would adjust the size of its inspection team needed to work off the backlog. If necessary, equipment for process monitoring or remote monitoring could be considered. The IAEA, State authority and the facility operator would have to cooperate in developing mutually acceptable schemes for cost and human resource allocation.

The HEU processing line at the Ulba plant remains available for use to eliminate further quantities of weapons-usable HEU. The IAEA is prepared to work flexibly with its Member States to verify the elimination of weapons-useable materials through down blending or to confirm their movement to facilities where physical security and peaceful use or disposal can be transparently assured.

FIG. 1. HEU down blending process line.

FIG. 2. Cutting of HEU fuel rods.

FIG. 3. Can for nuclear material.

FIG. 4. Nuclear material can, opened.

FIG. 5. AWCC measurement station.

FIG. 6. Digital camera installation.

Safeguards for final disposal of spent fuel in Finland

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Abstract. Olkiluoto was selected as a site for final disposal for spent fuel from Finnish nuclear power stations. At present the first underground tunnel for bedrock characterisation at repository site is being excavated. The application for the nuclear construction permit should be submitted by the operator as early as 2012. According the current plans the construction of the encapsulation plant and the final repository will start around 2014. The operation of the facility will start in 2020. The final disposal facility, i.e. an encapsulation plant and an underground repository, represents a new type of nuclear facility and new safeguards requirements need to be set. Safeguards measures of STUK are based on the national legislation and regulations as well as Finland's international commitments. The main safeguards objectives at the pre-operational phase are: 1) Generation of the verified as-build design data and 2) Assessment of all available safeguards relevant monitoring data to assure the absence of undeclared nuclear activities at or near the repository. In order to fulfil these requirements the implementing company is obligated to report the excavation plans and progress and monitoring reports timely to STUK, which assesses the information also performs safeguards inspections at the site. At operational phase verification of as-built design will continue. Concerning nuclear materials Finnish national requirement is that spent fuel will be comprehensively verified prior encapsulation followed with robust C-o-K methodologies over the whole encapsulation and disposal process. Finnish SSAC invites the IAEA to make full use of its functions and findings. It would be beneficial if the IAEA and the State had binding, but flexible, implementation arrangements to ensure that the IAEA safeguards needs are effectively met.

1. Introduction

Finnish Nuclear Power Company TVO ltd. started the final disposal research already at the beginning of 1980's. The research issues were technical concepts and country-wide screening of Finnish bedrock in order to locate suitable site for disposal. The final disposal is not purely technical process, but it also involves social and societal dimension. Another power company Fortum joined the project in 1996, when returning spent nuclear fuel from Loviisa NPP to Russia was banned. The power utilities established jointly a new company called Posiva Oy. Technically, the favoured basic concept is as follows: The spent fuel is planned to be disposed in copper canisters emplaced in bentonite clay backfilled tunnels excavated in crystalline bedrock about 500 m below the Earths surface. The details of emplacement are still subject to the underground research phase.

Posiva applied Government a Decision-in-Principle stipulated by Nuclear Energy Law to build final disposal facility in Olkiluoto. Both the local municipality and STUK as a safety authority had veto right to the application, but they approved the Posiva's plans. Government made the decision that the project and planned nuclear facility is in favour of overall good of the society. The decision was endorsed by the Parliament by a vast majority n May 2001. The Decision-in-Principle stipulates that Posiva must apply for nuclear construction license by 2012. According the present plans the construction of the nuclear facility will start in 2014. The facility will be operational in 2020. Posiva is granted a municipal construction license to excavate underground rock characterisation facility (called ONKALO). ONKALO is planned to be a part of final disposal facility. The excavation begun in June 2004 and in August 2006 altogether 1300 m of tunnel has been excavated.

In order to obtain public acceptance to this kind of project is not easy. However, Posiva did a successful job to convince local citizens, experts and local and national level politicians that their concept is pragmatic and practically feasible and involve only minimal risks to the people, property and the society. Posiva's work to gain acceptance included local hearings, a lots of technical background work, lobbying, publicity campaigns and other methods available in open and democratic society. As a consequence lots of people became involved into the process and even more became informed.

As a result of this acceptance process Finland became the first country in the world having approved final disposal plan. Moreover, as a result of ongoing construction of ONKALO, Finland is the only country in the world, where *de facto* geological repository is being excavated. One may note that ONKALO is not a nuclear facility but a research facility. For technical point of view it is of course pragmatic to use excavated research facility as a part of final disposal facility. Because the ONKALO is planned to become a part of final repository STUK have all the necessary powers to regulate the construction for the safety and safeguards assessment already in the pre-nuclear research phase.

The starting point of international safeguards is not as clear: the IAEA recommendation are given already to the pre-operational phase [1]; whereas, the EC requirement to present the BTC (Basic Technical Characteristics) to generate the Design Information is valid only 200 days before the emplacement of nuclear materials. Anyhow, the plan to construct a final disposal facility for spent nuclear fuel is declared under the Additional Protocol.

2. Present national safeguards system for the repository

The regulative power of STUK makes it possible to apply necessary safeguards measures in ONKALO. With these powers, STUK was obliged to set up safeguards requirements for the final disposal installations, in particular for the underground premises, which are *de facto* being excavated to form a part of the repository. The objective is to create the knowledgebase and maintaining of the continuity of that knowledge for the generations to follow and during the whole lifetime of the repository, especially in the area of safeguards [2].

The Finnish national safeguards system for the repository consist of generation of relevant data and information, field audits and verifications as well as review of the progress in tunnelling works and site characterisation [7]. The safeguards-related audits and verifications will be also focused on the contracted companies and their procedures. The national competent authority STUK will, in addition, use information from independent surveillance or monitoring techniques. These will include seismic monitoring and satellite imagery.

2.1. Establishment of Operator's Safeguards System

Nuclear material holders in Finland need to fulfil the regulations called YVL-Guides issued by STUK by virtue of National legislation. These guides are not binding, but the operator may suggest an alternative approach how the desired level of safety can be accomplished. Some guides stipulate safeguards issues, and STUK ordered Posiva to fulfil the applicable parts. These include:

- Nominating responsible person to take care of safeguards issues,

- Preparation of safeguards handbook (or quality manuals) which describes the processes and procedures must be performed that the safeguards relevant as-built information will be generated, processed, reported and stored. The handbook must be approved by STUK,

- Continuous performance following the accepted safeguards procedures.

The most vital part of the handbook is the reporting scheme. It is agreed that Posiva will periodically report the progress of the work and plans for the next reporting period. Posiva will also report in short notice about deviations. Typical example is if the excavation work intersects a weakness zone in the rock, which requires rapid actions for operational safety, like reinforcement of the tunnel wall that covers the virgin rock surfaces. In all cases STUK is reserved a possibility to have access to verify the tunnel wall before reinforcement. Reinforced tunnel sections will be reported separately, since after the reinforcement, the existence of any undeclared voids is very difficult to prove without disturbing the operational safety of the underground gallery and later, that of the facility.

2.2. Audits and verifications

The underground repository is mainly requlated for safety, not only safeguards. Therefore the precence of STUK in ONKALO is very regular. STUK’s safety inspectors can also make note of safeguards relevant issues [7] and report on them to the safeguards office in STUK. The tunnel sections are inspected by STUK before each of the scheduled casting campaigns. The impregnate rock walls will form the repository containment in which the facility installations can be constructed safely. The containment features of the excavated rock space are identified, authenticated and documented, thus providing the basis for license application and the Design Information at the given time in the future.

The measurement of the excavated rock spaced is carried by traditional surveying techniques as well as by laser scanning, which gives for analytical and documentary purposes a detailed 3-D space representation of the tunnel surface at the resolution of about 1 mm. Repeated surveys will be analysed as quantitative displacement measures of the safety analysis. The documentation will be complemented also by visual observations, photography etc. needed for the geological characterisation of features at rock walls. The resolution of any of these methods is considered satisfactory for safeguards purposes. The authenticity and relevance of the documentation is ensured by in-situ presence of inspectors. The national system functions include visual in-situ verification documents to be maintained for the continuity of knowledge. Short notice inspections by STUK are also possible and are by no means excluded.

These audits and verifications is primary tool in the national safeguards system. The objective is to visit ONKALO during the process of excavation at regular intervals to verify that the work proceeds as planned and reported. Therefore, basis of these verifications are reports submitted by Posiva. The visits are scheduled in co-operation with Posiva so that mapping and verifying the excavated spaces and corresponding virgin rock walls after the excavation can be physically verified. The audits are focused in Posivas planning and reporting procedures and supervision of subcoordinators. It is checked that all information which may have safeguards-relevance is really produced and stored in appropriate way. STUK may also request to have this kind of information for further analysis.

2.3. Monitoring methods

The repository performance is monitored for safety evaluations using several techniques [5]. The integrated analysis for the “safety case” produces data and conclusions for safeguards [3]. The deformation of the underground premises is monitored as a part of the timely as built design information records. In addition the hydrogeological and hydrochemical assessment increase the geological site understanding. These indirect interpretations may reveal unexpected rock properties that may also have safeguards-relevance. The rock mechanical monitoring is considered to will give the most safeguards-relevant information. Therefore, the microseismic data included in the rock mechanical monitoring [4] is applied as safeguards-relevant declaration by the operator. The excavation-introduced seismicity will also be studied using a network of microseismic stations at the investigation site. These seismic records are then inversed to estimate their source locations in the bedrock volume. The blasting, i.e., the explosions shot to proceed with the tunnels are also recorded with this seismic network operated by the subcontractor. Small scale blasts performed in excavation can be located typically with 1 - 10 m accuracy. So far the results have been well in accordance with the operators data. In addition, other activities in the area are detected owing to the active construction of new reactor and supporting installations, i.e. undeclared visitors’ centre, new accommodation village, other infrastructure. These are also visible in the commercial satellite imagery purchased and analysed by STUK.

One must not forget the societal verification. STUK is clearly part of the process. As a result of Posiva’s PR campaign for final disposal project lots of people became involved into the project. The network of sub-contractors is also wide. Being a well known and active member in this network STUK can be supposed to be informed if something covert actions would be performed in ONKALO area.

3. National safeguards requirements for the operational phase

Current plans indicate that the excavation of emplacement tunnels will continue also during the operation of the final disposal facility. Therefore the production of the timely as-built design information will remain one of the key elements of the safeguards system. During the operational period tunnels will also disappear owing to the backfilling procedures. The construction areas and nuclear material handling areas will be separated, therefore facilitating the safeguards controls.

Issue of safeguards of final encapsulation facility needs to be addressed. Only one detail differentiates the encapsulation and final disposal to traditional fuel handling facilities - reverification of the nuclear material is not possible after the process (or it is but at very high costs). At a national level STUK has concluded that prior encapsulation all nuclear fuel must be verified with the best possible technology. The generations to come deserve to have an exact picture what material has been disposed. The verification will create this information, minimising the risk that the disposal tunnels must be opened for safeguards reasons. In order this verification data to be reliable a robust C-o-K system must be employed. The issue is discussed in the report [2].

4. How to address IAEA’s present needs?

At national level STUK is confident it can fullfil its mission to prevent the proliferation of nuclear materials to unreported use. The strong mandate of national authority gives practically unrestricted short-notice access to the repository and POSIVA’s archives. The IAEA has no unrestricted access rights. The present stage of the repository (ONKALO) is not even a site in Additional Protocol declaration, so it will not provide complementary access to the IAEA. Therefore the IAEA should find other approaches to fullfil its mandate to gain credible assurance about absence of undeclared nuclear activities and materials.

One approach is to use so-called model repository approach developed by SAGOR expert group at the end of 1990's and the beginning of 2000's[1, 6]. It would require practically continuous human and instrumental presence at the site during the whole life time of the repository. The maintenance of the IAEA monitoring equipment would require also technical people to be reached at short notice. Assuming that there are at the IAEA inspectors who are competent to carry out the continuous design information verification and geo-scientific monitoring and evaluation, it is estimated that at least 1.5 – 2.0 person-years are needed annually at the site in the pre-operational phase of the repository. In addition, there is an obvious need for geo-scientific staff at the head quarters. This increases the inspection activities remarkable from the present 1 person-month spent annually in Finland, in contrast to the Integrated Safeguards aim to reduce the inspection activities. This is hardly acceptable.

If the IAEA could use the findings of national system it would be of great value. But of course the IAEA will act independently and make its independent conclusions, so the Agency can not directly tap into the Finnish national system and use it as a sole source of information. It is believed that STUK can provide the IAEA with valuable safeguards relevant information. Therefore STUK is inviting the IAEA to make full use of its findings. In order to better rely on the information provided, the IAEA is also invited to audit Finnish SSAC and its functions.

The IAEA should also examine the progress reports and as-built information. This data provides the basis against which the verifications and assessments can be made. The verification data can be obtained from three major sources:

- data from national system elements
- data from other proven sources either inside or outside the IAEA
- data from the IAEA visits and short notice accesses. During these visits the IAEA can verify in-situ the excavated premises and use proven methods.

In case of anomalies and inconsistencies the IAEA should approach the national system, which, in turn, should be able to provide clarifications needed.

The cost-effectiveness in the implementation of the IAEA safeguards measures could be based on the short notice random visits to the repository and periodic follow-up of the site characterisation and monitoring programme performance. The visits and adjoining verification activities applying mainly visual observations or surveying techniques might be carried out in concordance to routine inspections to the near-by located nuclear facilities to avoid travel costs or in an unannounced mode.

The monitoring programme performance and results are followed by the scientific expert group that also inspects quality and safety assessment of the repository annually. The participation in this group might be the most convenient and cost-effective way to maintain the required level of site understanding, thereby contributing to the safeguards evaluation. This procedure that could include the as-built design verification would require few person-days annually and would include assessment of the monitoring programme results and performance from the safeguards relevant points of view.

In general, the information needs and exchange routines are to be defined, since there are no legally binding obligations, neither in the Safeguards Agreement nor in the Additional Protocol guiding the State and the IAEA in the implementation of safeguards on geological repository in the pre-nuclear phase. The pre-nuclear phase should be declared as a general plan to the development of nuclear fuel cycle. In addition, the notification, reporting and other communication formalities as well as access procedures have to be negotiated and agreed with the relevant parties. A draft arrangement that takes into account Olkiluoto site specific aspects and the existing national safeguards system elements has been discussed and prepared [5].

5. Summary

The final disposal of spent nuclear fuel in the geological repository at Olkiluoto was accepted be the local municipality, State authorities, and finally endorsed by the Parliament of Finland in 2001. The site investigations proceeded to the underground phase in 2004 when the excavation of the underground tunnel system for bedrock characterisation at repository site began. Referring to the recommendations generated in the International Atomic Energy Agency's Programme for Development of Safeguards for Final Disposal of Spent Fuel in Geological Repositories, the implementation of the necessary safeguards measures by the national safeguards authority was initiated in 2004. The IAEA is recommended to make full use of this strengthened national system.

The main safeguards objectives at the present pre-nuclear phase are: 1) Generation and verification of the as-build design data that corresponds with the excavated rock space and its geometrical volume 2) Assessment of all available safeguards relevant monitoring data to assure the absence of undeclared safeguards-relevant activities at or near the repository.

The extensive geo-environmental monitoring data collected continuously primarily for site understanding and safety evaluation purposes are also accessible, as relevant, for the safeguards purposes. Integrated evaluation of the information generated by these monitoring tools will timely generate the updated actual lay-out of the excavated rock volumes; and moreover, can reveal indications concerning undeclared activities near or at the repository, thus providing a set of data and information that can contribute to the required assurances of the absence of undeclared activities and man-made structures of safeguards relevance.

In general, the monitoring methods contribute to the establishment and maintenance of a coherent and reliable set of data and information for the use in drawing safeguards conclusions at the geological repository site. The independent assessment fore safeguards purposes will require human and technical resoureses purhased and educated for each repository site, specifically. Taking into account the complexity and novelty of the subject matter availability of appropriate non-nuclear safeguards competencies must be secured at the inspectorates. It is recommended that the IAEA and the relevant other parties in Finland will negotiate legally binding implementation arrangements so as to ensure that the national system functions will be implemented through out the whole lifetime of the repository project in a manner that enables the IAEA to implement its safeguards obligations in Finland effectively.

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Analysis and processing tools for nuclear trade related data

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Abstract. In response to the proliferation risks from covert networks trading in sensitive nuclear technology, the IAEA has strengthened its focus on clandestine nuclear related supply and procurement activities. The analysis of relevant information constitutes an essential part of the evaluation of the correctness and completeness of a State's declarations. New working procedures and software tools have been developed to support the the IAEA's activities in this area. This paper provides a description of a secure system for processing, analyzing and storing covert nuclear trade related data.

1. Introduction

Following the revelations about extensive covert networks related to the procurement and supply of sensitive nuclear technology, the IAEA undertook to strengthen its capabilities for analysing information on such networks.[1] These revelations clearly confirmed the need for accessing and evaluating complementary information in order to draw the broader conclusion that all nuclear material remained in peaceful activities in the State.

The Nuclear Trade Analysis Unit (NUTRAN) was established in November 2004, within the IAEA Department of Safeguards, to centralize the analysis of all procurement network related information available to the IAEA [1]. NUTRAN is also the focal point for receipt of such information. In 2005, the IAEA General Conference invited all States to cooperate with the IAEA and its efforts to verify and analyse information provided by Member States on nuclear supply and procurement[2].

Information on covert networks is mainly related to actual or attempted trade transactions, involving equipment, material, technologies, companies and individuals. Trade related information appears in many different forms, the accuracy and reliability of which varies widely. For these reasons, the IAEA has developed, and is further improving, a specially designed system to enable consistent processing, analysis and reporting of such information. Further development of the tools is facilitated by the fact that all main nuclear trade analysis related processes have been identified and documented according to the Departmental quality management system (QMS) guidelines.

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[1] In his opening statement to the 49th Session of the IAEA General Conference, the Director General outlined a number of priorities for the coming years in the area of nuclear verification, including "continuing to investigate the nature and extent of the illicit procurement network". He also called for the IAEA to "explore the development of mechanisms to encourage better information sharing from States, on exports, procurement inquiries and other safeguards related issues".

[2] In operational paragraph 21 of resolution GC(49)/RES/13, adopted on 30 September 2005, the IAEA General Conference "welcomes efforts to strengthen safeguards, including the Secretariat's activities in verifying and analysing information provided by Member States on nuclear supply and procurement, taking into account the need for efficiency, and invites all States to cooperate with the Agency in this regard".

2. Features of the analysis and processing system

The system described herein was initiated in 2002, then enhanced and restructured in 2004-2005. Further enhancements are currently scheduled to improve the efficiency of the following features:

— Handling information originating from a variety of sources in different formats;
— Handling structured information and flat unstructured text;
— Enhancing information security features through managed access;
— Indexing and storing information to enable quick access and retrieval for analysis;
— Detecting unrecognized links and integrating them with existing information;
— Viewing and understanding the relevance of any discovered links;
— Generating automatic reports that highlight any gaps in current knowledge;
— Maintaining records of sources and source documents;
— Enabling information reliability ratings.

All these improvements will facilitate maintaining the IAEA's institutional memory of procurement activities related to nuclear trade in a form appropriate for analysis.

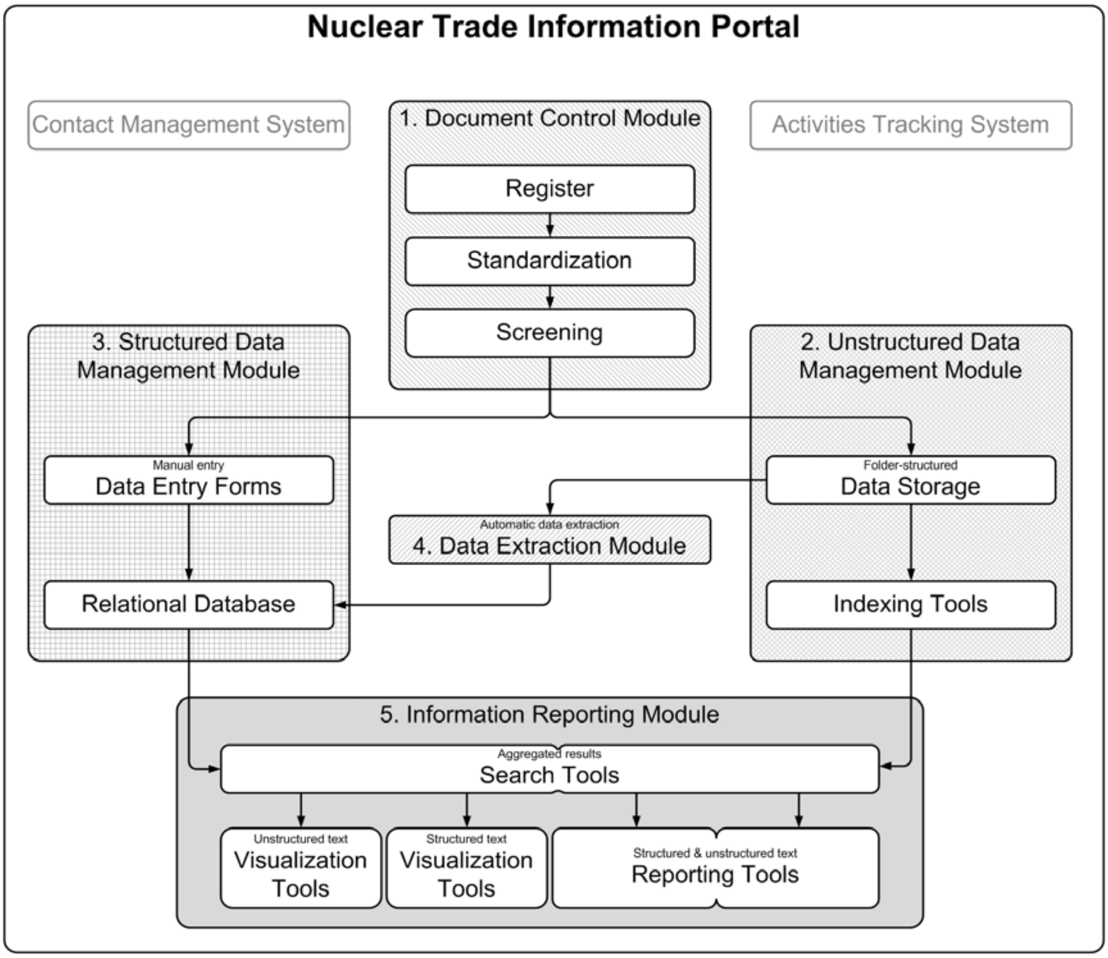

FIG. 1. An overall diagram showing the Nuclear Trade Information Portal modules and their components. The main system is operational. However, to improve efficiency, individual modules and components need further development and integration. The arrows show how different modules are interlinked.

3. System details

The Nuclear Trade Information Portal was developed and it is used by NUTRAN. The individual modules and components and the overall architecture are shown in Figure 1. The system consists of several modules each having several components. Most of the components are operational while some of the modules are only planned, waiting for development or implementation. The practical trade analysis experience gained during the past few years has helped the Unit to identify the development needs. In general, the portal modules require further integration for efficiency and secure access by additional Departmental users. The modules, which are discussed below, include the following:

— Document control module;
— Unstructured data management module;
— Structured data management module;
— Data extraction module;
— Information reporting module.

The Portal also includes two modules that are not described in this paper. They are the contact management system and the activities tracking system, which are used to control and log the daily activities of the Unit.

3.1. Document control module

The document control module enables the initial processing and tracking of incoming documents and data that are relevant to procurement networks.

Incoming documents are received in many different formats which makes the onward processing rather laborious. All documents are standardized upon receipt into a common electronic format (portable document format (PDF)). Hardcopy documents are processed using an optical character recognition (OCR) software, where appropriate. An assessment of the document reliability, based on its source and the quality, is also made at this early stage. This is important as it has a bearing on the ultimate reliability of any information extracted from this document and utilised as part of further analysis.

After standardizing, all documents are screened to determine whether the content needs to be entered into the structured data management system (see para. 3.3 below). This decision is based on whether the data contained are considered sufficiently specific and relevant for present or future use. The preliminary screening is required to carry out document prioritizing. As a minimum, all documents are stored as unstructured data (as described in para. 3.2 below).

3.2. Unstructured data management module

The unstructured data management module stores the information contained in what are called flat files. All incoming or generated documents are stored electronically in a searchable location in the standardized format to ensure that all information is available to the analyst when required. The current version of the system is called the information base and its elements are used independently.

3.2.1. Folder-structured file storage

Currently, this is a simple file storage system. Storage is organized in topic based folders to aid the user when accessing related documents. Electronic shortcuts to source documents are used when multiple storage locations are relevant. However, this classical method relies on the users' selection of the document's storage place, which is not always consistent.

3.2.2. Indexing system

A powerful, commercially available, text indexing software is used to index all the text based documents received. The system can be configured to include multiple indexes by topic or according to the analyst's needs. The system comes with its own search function. Alternatively, it can be integrated into other stand-alone search systems.

3.2.3. Search system

This is a form based search system that comes with the commercial indexing system. The system is accurate and fast; however, it requires the analyst to read though the results and determine the most relevant data to the search topic. Iterative keyword searches, although often time consuming, are performed to optimise search results.

3.2.4. Visualization system

Visual search solutions, used for optimizing document searches, are currently being tested by the Department. One system being developed and tested allows simultaneous aggregation of search results from different indexing tools and it groups relevant documents into a visual representation. In this representation, the documents are organized around the search terms in a manner that optimizes the identification of the most relevant documents related to the topic researched. This tool, still under development, can be used both on the web and on the local LANs. It will also enable the analyst to exclude irrelevant documents based on a set of automatically generated keywords relating to the documents retrieved. The system also helps to highlight any information gaps in the collection of documents retrieved [2].

3.3. Structured data management module

Data are entered into the system if they are assessed during early screening as being sufficiently specific and relevant for analytical use or they are recognized at a later stage as being relevant. Another method of data entry is to use an automatic data extraction system. Data entry is a daily routine activity. It requires systems that facilitate efficient, secure and consistent entry. The main elements of the structured data management system have been specially designed within the IAEA and are called the knowledge base. They provide consistent data that can be easily interfaced with commercial visualisation tools. However, further enhancement through the integration of the various system elements is required to improve efficiency.

3.3.1. Relational database

Analysis and processing tools make use of a specially designed database. It is used to store structured data, record and show complex links between entities in covert nuclear trading networks. The database was developed by the IAEA and is hosted on a secure (air-gapped) SQL server. It has built in security and authentication functions to ensure authorized access to all data and source documents.

3.3.2. Data entry screens

The screens are optimized for consistent manual entry of trade related information. They provide guidance to the user when entering information related to sources, cases, entities, trade transactions, equipment and events. They also reduce duplication, increase the efficiency of the user and improve the reliability of the data entered. The screens are linked directly to the relational database application, currently called the Procurement Tracking System (PTS).

3.3.3. Search engine

A custom made text-based search engine is used to search the relational database and produce appropriate automatic text reports. The database holds information on multiple level relationships.

Because of their complexity, reporting is confined to one level only. The analyst has the option to direct searches through a search screen. The search results coming back in the form of hyperlinks could be used to navigate and explore all relevant information.

3.3.4. Visualization tool

Due to the complexity of multiple layer relationships, a commercially available data visualization tool is used to allow the analyst to extract information from the database and display it in a visual format. The tool enables the depiction of all relevant information held in the database, allowing possible hidden links to be uncovered. It also allows the analyst to identify and interrogate specific nodes and networks of interest.

The planned integration of the search engine, reporting system and the visualization tool are described in the information reporting module (see para. 3.5 below).

3.4. Data extraction module

To decrease the workload of manual data processing, the feasibility of various automated data extraction capabilities has been investigated. Such tools appear to offer interesting possibilities. However, no system has been deployed yet, but several commercial tools could be considered. The tool to be selected needs to be able to extract contact and product information related to entities into a structured data set that can be imported into the relational database.

Whatever system will be chosen, the quality of the data extracted and imported into the database will not be as reliable as that entered through a manual process. Therefore, certain precautions are needed. It is envisioned that the tool will only be used with documents and data suitable for this type of processing. In addition, information extracted will be tagged with a specific reliability and confidence rating, which identifies the automatic mechanism used to enter the data. These ratings may be upgraded by an analyst, if appropriate.

3.5. Information reporting module

This module, yet to be developed, will integrate the searching and reporting of both the structured and unstructured data management modules. Currently, both the unstructured and the structured data management modules have their own search engines and visualization tools.

The information reporting module will be isolated from both the structured and the unstructured data management modules and will belong to a different security ring. Separating the reporting tool from the data entry tools enhances the security of the system as it separates raw data from analysed information.

The information reporting module will have three components: First, a search tool will be used for entering the search term and displaying the results. Second, a visualization tool will display the search results in a visual interface to increase the analysts' accessibility to information and decrease the time required for searching. Third, a reporting tool will generate reports that combine the results of visualization and automatic text reports from the database.

3.6. Security and access rights management

All of the modules mentioned above are controlled by a security and access rights management system. It consists of two components. The first component, the security management, ensures that only authorized personnel have access to the system. This is currently arranged by encrypting the file storage system and using the entire system on a stand-alone, air-gapped LAN that is not connected to other networks by either wired or wireless means. The second component, the access management, controls access to the database itself and its records through a two-layer system. The first layer allows access to the complete database and access is limited to authorized personnel only. This is currently deployed. The second layer limits access of authorized personnel to the defined record level of the particular database. This function is yet to be developed. It will allow different levels of secure access to the database and, thus, additional Departmental users.

3.7. System integration

Reliability and rapid access of and to data are major requirements of an analytical system. Reliability requires, amongst other things, avoiding data duplication. Duplication quickly increases the amount of noise in a data set and decreases the quality and speed of analysis. The integration of system modules will improve this aspect because of the consistent application of process tools used to prevent such duplication. The portal access will provide a better overall view of the information available to the user for analysis or work with the data. However, the integration of all the modules described above will require significant software development.

4. Conclusion

In line with the Departmental Quality Policy Statement, the continual improvement of nuclear trade analysis processes, tools and systems is an essential part of the endeavours to detect early signs related to covert trading of nuclear related goods or services. The majority of the modules described herein are currently functional but deployed as stand-alone modules that require further integration. The amount of information related to covert nuclear trade is increasing while the Unit responsible for the analysis of covert trade related issues is relatively small. Any enhancement of the overall system and tools supporting secure handling, storing and analysis will allow access to additional Departmental users, thereby improving efficiency and increasing the chance that covert nuclear trade related activities be discovered on time.

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Duplicate management in mining open source literature for knowledge and intelligence

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Abstract. In the collection and processing of Open Source information for knowledge discovery and other intelligence purposes, no matter how clever the search algorithm or analyst, a persistent problem is the management of the sheer volume of information identified. This information comes in the form of unstructured, semi-structured and structured data. One of the key issues that exacerbate this information overload is the production of duplicate or near-duplicate information. Another is the collection of data that has little relevance or utility to the subject matter of interest (referred to as Spam in this document). This paper focuses on the identification and removal of duplicate, near-duplicate information and Spam in the context of a state-of-the-art Technology Data Analysis System that is specifically designed to organize information around an organization's technology strategy. To that end, the paper provides the analysis of the design and algorithmic infrastructure developed to create a mechanized system that virtually eliminates the duplicative and irrelevant information resulting from Open Source data gathering activities. This system, the Automated Knowledge Discovery System (AKDS), is designed to automate the identification, retrieval, analysis, and organization of scientific, technical, and business data and provide a continual surveillance of the current state of the technologies of specific interest to an organization. This paper focuses on the text analysis modules that enable the identification of duplicate and irrelevant information. In this paper, these concepts are referred to as duplicate and near duplicate detection and irrelevant information (Spam) filtering.

Introduction

In the collection and processing of Open Source information for knowledge discovery and other intelligence purposes, no matter how clever the search algorithm or analyst, a persistent problem is the management of the sheer volume of information. In 2004 Google indexed over 3 billion web pages and the Internet was growing at a rate of more than 10,000,000 new pages a day. [1] In 2005 over 17 million new sites were added. [2] Yet this phenomenal growth does not fully reflect the complexity as much of the information of value is not indexed by Google or other search engines. In fact, most information on the Web is not indexed by any of the major search engines but resides in the deep web and in database services such as Factiva, LexisNexis, and Dialog. [3] To add to the complexity and technical challenges of gathering useful data, this information also comes in the varied forms of unstructured, semi-structured and structured data. One of the key issues that exacerbate the retrieval and information overload challenges even further is the production of duplicate or near-duplicate information; a related issue is the collection of data that has little relevance or utility to the subject matter of interest.

In addition to the escalating amount of information being published, an undesired effect that impacts organizations is the amount of time knowledge workers expend on searching for the information they seek to enable them to perform work related tasks. Information technologies that can address this facet of their work and reduce the amount of time that is needed to find the right information can certainly improve the bottom line in dramatic ways; it is estimated that knowledge workers spend 20-30% of their work time searching for online information but only successfully find the information they need 50-60% of the time; this level of effort translates to $18-20K per knowledge worker per year of unproductive effort [4].

Automated Knowledge Discovery System (AKDS)

A variety of information solutions for providing targeted, competitive intelligence analysis can be enabled through automated systems that virtually eliminate the duplicative and irrelevant information resulting from Open Source data gathering activities. In the context of state-of-the-art Technology Data Analysis Systems, a set of data processing tools has been developed that can illustrate the effectiveness of new technologies. One system that highlights opportunities to apply technologies to address information overload is the Automated Knowledge Discovery System (AKDS), a hybrid toolset that is designed to automate the identification, retrieval, analysis and organization of scientific, technical and business data. The information targeted by the AKDS is determined by the topics in an organization's strategic plans. Therefore, each organization would have a customized AKDS that provides continual surveillance of the current state of the technologies of specific interest to that organization. The research and development of the AKDS was funded by the US Department of Commerce's Advanced Technology Program; improved data mining features have also been funded by the National Science Foundation Small Business Innovative Research program. AKDS entered commercial markets in early 2006.

Many organizations have strategic objectives that are documented through a variety of methodologies; for technology intensive organizations, strategic planning often includes the development of a technology roadmap. This roadmap articulates the focus and a timeline for the development of each of the capabilities that must be met in order for the organization to reach its future objectives. These capabilities are often related to developing or maturing specific processes or, as in the case illustrated in the manufacturing sector, improved goods and products. Yet, over the lifetime of the strategy, organizations are challenged to identify and continually track the potential array of technology solutions that may exist for each of the objectives that are on the critical path.

Although a key reason why technology strategies fail to be implemented is related to organization cultures and the difficulty in successfully moving a strategy through all levels of a complex organization, access to needed information is one of the highest ranking technical reasons why organizations fail to meet strategic objectives defined in their planning stages. The AKDS was designed to address this deficiency; the system can easily be initialized for a specific organization's technology focus based on content derived from the strategic objectives. For organizations that do not have formally documented strategies, a set of defined topics of interest can be used to initialize a system.

As illustrated in the figure below, the AKDS is based on the development of a hybrid approach to leverage semantic-based technology with traditional text and data mining processes. The AKDS is composed of three subsystems that collectively enable identifying, analyzing and organizing targeted data. The architecture includes:

- The Search Subsystem – a set of modules responsible for submitting queries to multiple data sources, retrieving relevant documents, metadata extraction, and passing the documents and metadata into the knowledge repository. The design of the Search Subsystem includes the capabilitiy for adaptive query formation; when query terms or logic do not prove useful in specific sources, the query will be adapted when searching the source in future search sessions.
- The Automated Knowledge Discovery Subsystem – a set of modules that applies the knowledge contained in domain ontologies to identify the key concepts represented in the technology roadmaps and the documents retrieved from the Search processes. The text analysis pipelines which are the focus of this paper are located within this subsystem.

- The Mapper Subsystem – a set of modules that evaluates the key concepts in the documents and associates relevant documents to topics identified in an organization's technology roadmap thus providing a current state view of every topic in the organization's strategic pipeline.

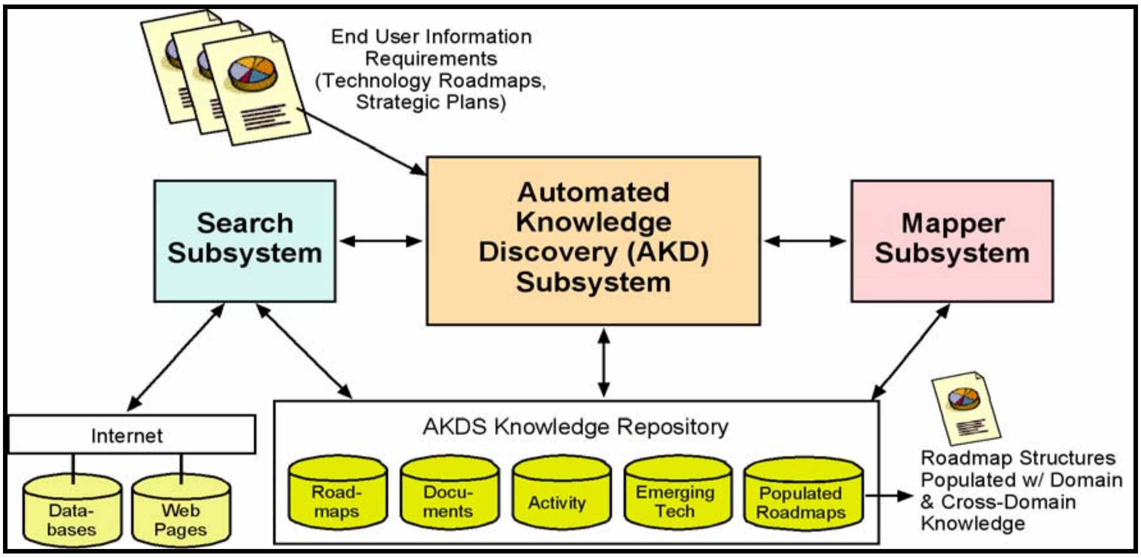

FIGURE 1. The AKDS is based on a service-oriented architecture that includes separate web services for enabling automated search, text and data mining, classification, and content management.

Initially an AKDS is a blank slate awaiting the specific definition of the topics for which the system will focus. Once the technology topics of interest are identified, the automated retrieval, analysis and classification processes described above will manage the full spectrum of system requirements. A set of classifiers are developed that includes all of the potentially useful terminology to retrieve information on the topics; expanded query terms are derived from a set of domain specific ontologies that encompasses the technical language of interest to the organization. The query terms are then continually submitted to targeted data sources as well as the open web; the resulting search sets are retrieved and stored in the AKDS knowledge repository.

Each of the targeted data sources has a set search schedule; thus, the AKDS will continually return to the data sources and refresh the information available on a daily, weekly or other basis as warranted for a specific source. Organizations can then have a constantly updated "current state assessment" of the technologies of direct interest. This current state assessment can provide business intelligence, competitor analysis and can also trend research activity across the entire spectrum of technologies identified in the strategic plan. Since one AKDS can monitor thousands of technologies, the level of effort of knowledge workers retrieving information across the organization – as well as the timeframe for information delivery - can be impacted in dramatic ways.

The AKDS Document Utilities

Within the text and data mining web services, located in the AKD subsystem illustration above, the AKDS design includes a text analysis pipeline that is designed to evaluate the information utility of the retrieved documents. Often, a query term represents several distinct concepts (*polysemy*) and so it is anticipated that the search system will retrieve inappropriate word-to-intended concept mappings. A set of machine-learning-based tunable filters were developed to assure that the information that is ultimately retained in the system potentially has high value to end users. The pipeline includes web services that are designed to identify duplicate, near-duplicate, or irrelevant information.

A duplicate document is one that is an exact replica of an existing document that has been previously retrieved by the search subsystem and that also includes the same record designator from the original source, such as a URL or accession number. A near-duplicate is a document that does not compare exactly to an existing document in a character-by-character analysis; however, the discrepancies between the two documents are so minimal that statistically there is a very low likelihood that the subsequent document will provide additional utility to a knowledge seeker. Irrelevant information is a broad category of documents that, for a given system, provide no information utility to the population that the system or query is designed to serve.

Once the low value information is identified, a separate pipeline is initialized to determine if the data should be kept in the knowledge repository or removed from the collection. If the data is kept in the system, a data clustering web service evaluates documents' similarity to existing documents already stored in the repository and will collapse the documents into a cluster. When the system user submits queries to access documents, only the most representative document in the cluster will initially be presented to the user; however, links to the remaining documents in the cluster will be available for system users to access all related documents. Through this combination of data processing steps, all documents are archived in the system, yet end users no longer retrieve duplicate information and, at the same time, have more direct access to unique, high value information.

The selection of the most representative document in the cluster that is initially provided to users in search query results is performed at the time the end user submits a query. The most representative document is selected based on the end user's specific query terms. The system analyzes the terms and then identifies the document that contains the nearest matches within the document cluster. For documents that have equal concept distribution, additional analysis is required. In the AKDS, each data source is given a trust level to segregate data that has a high pedigree from other data sources for which the pedigree is unfounded. This trust level is integrated into the algorithm where multiple documents have equal indicators of being identified as the most representative. In addition, some metadata fields are also included in the computation. For some organizations, the most representative document in a cluster may be the document that has the most recent publication date. For other AKDS implementations, the most representative document may be the document that has the most complete set of metadata, documents derived from a specific data source, or a combination of factors.

The illustration below shows the major functionality included in the document analysis pipeline.

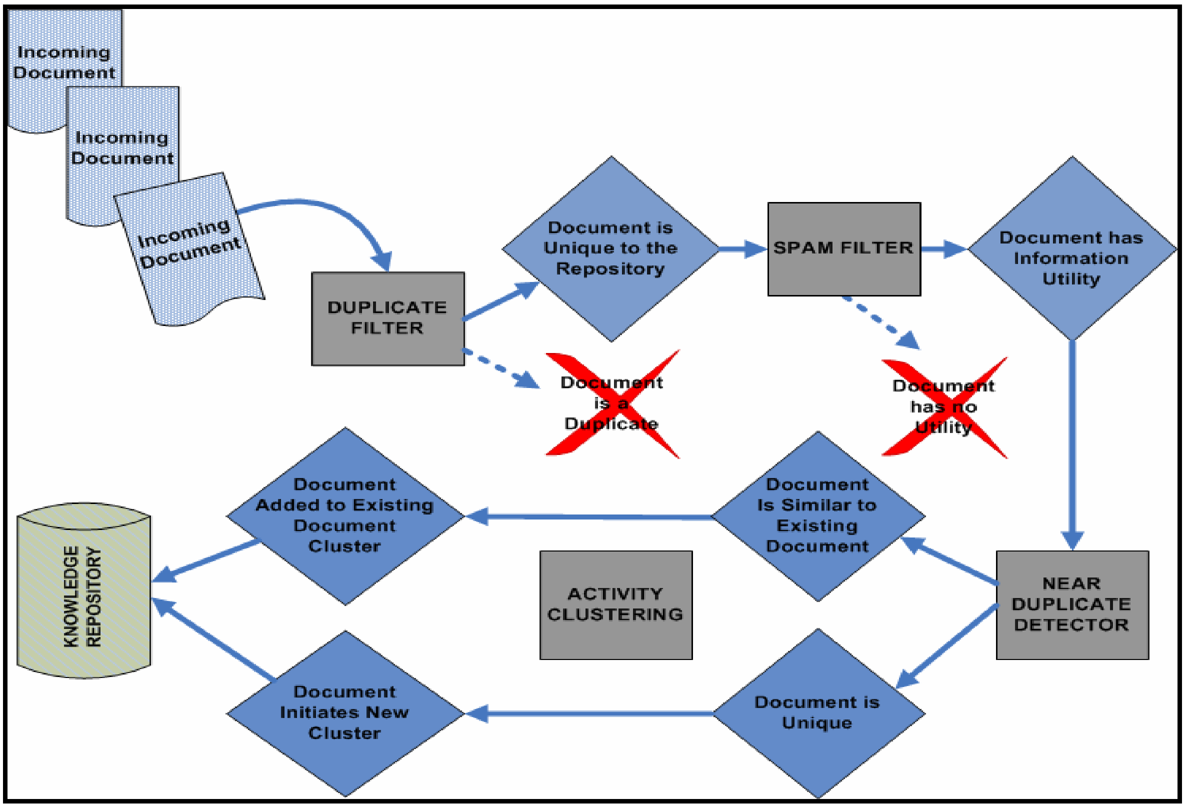

FIGURE 2. The AKDS Data Evaluation Pipeline is a web service that includes tunable filters that can be modified based on an organization's determination of what constitutes "information of value".

Exact Duplicate Detection

After an AKDS system has been initialized and populated with a current data set, the search subsystem of the AKDS is designed to access specific, targeted databases and websites to extract new information being added to the data sources. A search specific plug-in is designed to address the nuances of each source; for retrieving information from open websites, standard search engines are accessed through a broader-based plug-in. Query terms are derived from the specific technology topics identified for the AKDS customer that have been represented in the domain ontology; derivative terms, synonyms, and related concepts are provided for query formation; additional logic is applied based on the development of a global search schema that enables reusing code from one source to another.

Once the search queries are launched, documents matching the search criteria are identified. However, before retrieving the documents into the system, two separate filtering process are invoked. The first duplicate filter is launched to remove any documents for which the exact document has been identified and retrieved by the system in previous search sessions; this filter is perfomed at the document URL level. The search plug-ins are designed to re-retrieve duplicates after some configurable amount of time; the amount of time to which the variable is set will vary from source to source. This allows the system to pull in updated documents that may contain the same URL as the original document.. For specific data sources that are set to retrieve a specific number of documents per query session, the plug-in will return to the search session and retrieve additional documents until the predetermined number of unique documents have been extracted - or until no documents remain at the targeted source that match the query terms.

Once URL filtering has been completed, a more powerful filter is invoked to insure that the exact same document is not retrieved from a second data source. This is a commonly occuring problem when a single document is published on multiple sites. This process computers a SHA1 hash of the document file and compares the hash to every document previously retrieved. If there is no match, the document is then pulled into the AKDS. If the document is rejected, a link is created between the document and the query that retrieved the document in order to identify the data source for a specific document.

The identification of exact duplicates is a vital step in the information retrieval pipeline. A single technology topic is typically composed of many concepts for which a set of queries is developed. A query to retrieve information on a specific subject might include dozens of concepts in various expanded configurations that result in dozens of queries per topic. As a result, the likelihood of retrieving the same document in more than one query session is high. Therefore, the ability to continually access the data source to retrieve unique documents for each query is an important facet of automated search processes as well as a key enabler to support more streamlined processes for data management.

Duplicate and Near Duplicate Detection

Once documents are identified as new to the system, the duplicate assessment modules are invoked which are based in part on the IMatch algorithm devised by Abdur Chowdhury [5]. Identifying duplicate, or near-duplicate, information is essential for minimizing the number of documents that system users retrieve. The technical objective of this part of the pipeline is to identify which documents are so identical to existing information that a user would gain no information advantage by the subsequent documents. Duplicate documents are often a legacy of data management practices that allow multiple copies of a single document to exist in a system or where a single document is published in multiple data sources without modification from data source to data source. Documents that are determined to be duplicates can be removed from the system based on system owner requirements for document management.

For system owners that are required or desire to maintain all variant copies of all documents in the collection, the documents that are determined to be near-duplicates are kept in the system and collapsed into clusters; these documents often have very slight variances in content such as a formatting feature or minimal character differential between documents. For these near duplicates, a user would gain little new knowledge from accessing each document; however, the slight differences suggest that one document may have some important distinction that could be relevant to an end user; therefore, these documents are typically not discarded.

The duplicate assessment algorithm operates by taking a series of normalized ‘slices’ of terms from a frequency ordered bag of words document and applying the SHA1 hashing function. The process then evaluates the document hashes to identify documents that have nearly similar content and for which subsequent documents do not have any new or unique information. Both the size and offset of the frequency ‘slice’ that is hashed determines the level of discrimination that is applied. For instance, if a broad slice offset at the least frequently occurring document terms matched between two documents, one can conclude that the documents are nearly *exact* duplicates of each other – differing only by line terminating characters or some other subtle difference. Conversely, a hash derived from a narrow slice taken from the most frequently occurring terms would match in spite of more significant differences – indicating a subset, superset, or revision of the same document.

Based on the specific application and how the system owner defines the information of value, the identical or nearly identical information (near duplicates) can be kept in the system and further processed by submitting them to a clustering web service; for the AKDS, clusters can be an effective way to minimize how much information is presented to system users. When a user’s query returns documents that are members of such a cluster, the most representative or ‘best’ document may be displayed to the user and the near duplicates would be initially hidden from the user.

Conversely, if there is no perceived value in maintaining all of the documents, the near duplicates can also be deleted from the system. This can be accomplished in an automated manner by comparing the number of coinciding hashes while accounting for their slice size and offset in the term frequency based representation from which they were derived. The user may then specify a threshold-based degree of hash matching, above which documents may be automatically rejected.

This module is particularly useful for systems that are evaluating large volumes of unstructured textual data such as Internet search engine queries that have a tendency to return the same or similar documents from multiple sources. This module has also proven to be effective in identifying documents where multiple versions have only slight modifications for which the end user would gain little utility beyond the initial document reviewed. Current metrics to assess the effectiveness of the module indicate that, after processing through the system, less than 2% of the remaining data could be considered redundant. In other words, once the search results have been processed through the dupcheck module, more than 98% of the data is non-duplicative.

Irrelevant Information (Spam) Filtering

The AKDS system also evaluates document features to identify documents that contain little, if any, utility. This module utilizes a process similar to the See5 decision tree algorithm developed by Ross Quinlan [6] to determine whether the prevalent features (or lack thereof) indicate that a document has a high potential for useful informational content. The features used to assess documents fall into three major categories: HTML tags, parts of speech (POS), and phrase analysis. HTML features consist of the density of a given tag within a document. POS analysis utilizes Word Net [7] to determine the potential parts of speech individual tokens may have. WordNet is an online lexical resource developed by the Cognitive Science Laboratory at Princeton University. It provides synonym sets and relationship linkages between nouns, verbs, adjectives and adverbs in the English language.

In addition, stop words (e.g., a, an, the, therefore, etc.) are counted. The phrase tagging data analysis determines the density of noun and prepositional phrases as well as complete sentences, fragments, proper nouns, and other named entities. The densities of all of these features are applied to a decision tree to determine whether the document is of good quality or a web form, advertisement, or list of links or other unwanted data. The most significant challenge of this development was to determine which features contribute significantly to an assessment and which contribute only noise.

Documents that are determined to be low value by their decision tree classification are removed from the system. This prevents unnecessary subsequent document processing and removes Spam or list documents that contain large numbers of ‘buzz’ words and are falsely returned as high ranked documents in user queries. Additionally, this module can be fine-tuned to filter specific types of documents from a collection based on the user’s needs.

Current categories evaluated by the decision tree processes include forms, lists of abstracts, lists of linked items, lists of unlinked items, lists of news items, lists of glossary terms, and full text articles. Each category is tagged as either good or bad prior to training and constructing the decision tree. The addition of categories and refinement of existing categories allow the module to be calibrated to include or exclude a variety of types of data including advertisements, class syllabi, foreign language documents, and other types of information not traditionally considered as Spam.

It is also important to note that the assessment of features does NOT include the *content*, but rather only calculates the density of types of occurrences such as tag type, POS type, phrase type, and sentence type. This allows the algorithm to perform well in any domain. That is, it seeks to find documents that contain large segments of naturally written text that is part of a contiguous, grammatically correct body of work. This process achieves an overall accuracy greater than 90% in practical application and is derived from a cost-effective level of effort as it requires only ~800 human-classified training documents when substantially altering the parameters.

Information Clustering

To further reduce information overload, documents that discuss similar technologies, processes, or other content can be grouped by examining their *conceptual* similarity. By employing an ontology, documents can be defined in terms of concepts rather than terms. The advantage of this approach is that it reduces the document content description to unified, unambiguous concepts rather than term tokens that can vary greatly in describing a topic and introduce ambiguity if analyzed out of context. The ontology allows for the contextual analysis of the text of a document and the assignment of these unambiguous concepts to represent the content. Once conceptually defined, documents can be grouped into clusters based on their prevalent topics discussed and other required metadata. For example, for the AKDS, two documents may discuss the same research project, but the performing research organizations in the two documents are different universities. Based on the tunable clustering process, the two research projects could be placed in separate clusters even though the research itself was identical.

The clusters are also used to form the basis of summary level reports that the AKDS can generate based on criteria determined by system users. For example, if a user would like to identify the top ten performing organizations for a specific technology, the user can create the criteria for a report in a similar way as one would search the knowledge repository. However, instead of providing documents as the results of the query, the AKDS is designed to analyze the metadata associated with clusters that contain documents that match the user's criteria; the system then generates a graph showing the trend or analysis the user specified. The graph includes links to the actual documents from which the values were derived.

This activity clustering can certainly be useful in limiting the number of documents that a user or analyst must look through to determine whether the results are relevant to the subject of their query. When an interesting result is viewed, the user can expand the cluster to see other relevant information related to the query topic. This allows documents that might have otherwise been poorly ranked due to the nature of the query to be presented within a document cluster that was highly ranked – thus 'finding the needle in the haystack' and putting the query into context with similar publications.

System Integration

To integrate this complex array of processes, a stable underlying architecture was required. Initially designed as an event-driven configuration, the broad requirements for the subsystem modules moved the system through various development stages ultimately to a service-oriented architecture which provided the level of stability and scalability necessary.

The architecture enables the integration of new functionalities and modules over time without costly redesign for integration efforts for future systems. Additional modules can be developed as a single application or could effectively be integrated into existing services or pipelines without requiring redesigning the basic functionality or architecture.

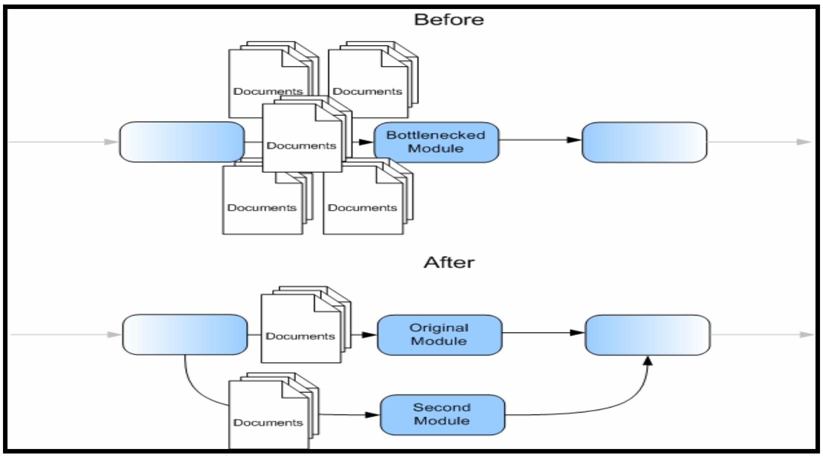

FIGURE 3. In response to increases in the load on a particular module, AKDS automatically creates additional instances of the module.

The AKDS architecture enables near-linear scalability. The loosely coupled nature of web services, combined with a management system that addresses network load balancing, allows AKDS to gain nearly 100% of the benefit of adding additional hardware. Thus AKDS can respond in real time to increased demand on a particular module by instantiating multiple instances of the module, thereby increasing the width of the processing pipelines automatically.

Summary

For technology-intensive organizations, filtering the vast amount of available information into manageable knowledge about capabilities, projects, personalities, and technologies results in many benefits for the knowledge workforce. In the AKDS information value model, organizations benefit from having greater visibility and better alignment of how well their research and development (R&D) investments match their strategic, organizational objectives.

Knowledge workers benefit by having a system that enables greater efficiency in identifying and retrieving information of value by continually being able to monitor the technology and business landscape for their research areas of interest. AKDS system users can access the records stored in the knowledge repository, but the AKDS model for information delivery also includes components to enable users to more effectively gain the benefit of automated processing. Through the duplicate, near-duplicate and Spam filters designed in the document analysis pipeline, end users can access information that has been highly filtered; thus end users can focus on ***using*** the information being sought rather than expending their valuable resources filtering through vast amounts of data.

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Integrated information portal

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Abstract. The Integrated Information Portal (IIP) is a web-based information portal to support the needs of decision-makers responsible for nuclear safeguards. The emphasis of IIP is to provide a seamless interface for accessing multi-source information in a timely and intuitive manner. With proper access control, open source information can be integrated with internal confidential databases, through an open-architecture design. The paper discusses the open architecture approach from an information technology perspective. To demonstrate the concept, a prototype was created using Microsoft .Net, AJAX enabled technologies and a customized ESRI ArcIMS web mapping solution in an n-tier client/server architecture. The benefits of implementing and using a solution orientated in this manner are presented.

1. Introduction

The first phase of our work on 'Integrated Information Portal' (IIP) was presented at the 2005 INMM meeting in Phoenix Ref. [1]. The ongoing evolution of the IIP concept has been sponsored by Canadian Safeguards Support Program in support of the International Atomic Energy Agency (IAEA). The previous paper discussed the concept of IIP, as well as demonstrating example technologies that illustrate aspects of IIP. Building on the concept, further work was carried out in this project to enable the integrated part of the vision, moving the concept into a prototype version. In the IIP prototype, users of all roles involved in nuclear safeguards would access a Common Operating Picture (COP) user interface having appropriate privilege levels. The visual nature of the interface would support the ability to search for information geographically and connect to various open and internal information sources in a seamless, integrated manner, as well as having access to open analytical tools. The intended purpose of IIP is to provide decision makers with online situation awareness and collaboration in the area of nuclear safeguards, while enabling technical personnel to leverage the interfaces to support in-situ analysis and monitoring.

To demonstrate IIP, the prototype was set up as n-tier client/server architecture with server-side processing to provide web access. Technology details are discussed in the system architecture section. The main goal was to demonstrate the feasibility of using existing web technology to support information exchange in a manner that is easy to use and incorporates interoperability with respect to technology and data. A combination of in-house data and open source freely available data were used in this demonstration. Interoperability is demonstrated through data exchange formats and using available web mapping services.

The prototype is an evolving concept shaped through the input from industry users. In this project, a case study related to the delivery of spent nuclear fuel to a storage facility was used as an illustration. Included in the case study is open-source information readily available from the web, complimented with examples of in-house data. It has been envisaged that available open-source information on the web could be a valuable source of information for the IAEA, as many of the data sources are readily available, generally reliable, up-to-date, and cost-free or at nominal cost.

The details of the system architecture used to design the system, including interoperability, access to information, and the interface layout, are discussed in the next section.

2. System Architecture

The technology to enable IIP into an operational setting is readily available and can support a rich fusion of information that is openly accessible. System architecture was used as a design step to determine the appropriate technology to form the backbone of the IIP. The design criteria are defined by the business requirements and users needs, in balance with the strategic objectives. In another words, longevity of the system, maintenance, interoperability (data & software), governance, functionality, and future needs are accounted for as part of the design planning.

The particular implementation for this prototype solution leveraged Microsoft .Net framework and ESRI ArcIMS solution for web mapping. Web components, including mapping, are wrapped around AJAX enabled technologies using ASP.NET for server side asynchronous processing. The solution allowed the users to experience dynamic viewing of multiple layers within a series of panels, all simultaneously and interactively.

The system architecture is an n-tier architecture that consists of presentation tier, business tier and information tier. IIP is distinctly composed of the IIP client and IIP server. The IIP client represent the presentation tier and is the user interface contained within the web browser that renders map display, triggers the request for open source internet data, and provides dynamic content of information. IIP server role is represented as the business and information tier. The business tier retains the design queries and functionality. The information tier provides software objects to broker and enable database connectivity. The database can be any Relational Database Management System (RDBS) from proprietary vendors such as Oracle, SQL Server, and DB2 to open source like PostGresSQL and MySQL. Spatial and non-spatial data can be store within separate databases or together in one. The mapping engine for IIP is controlled by the ArcIMS server. A .Net connector was created to interface with the web browser and ArcIMS application server to manage the client’s web mapping. Figure 1 shows the system architecture.

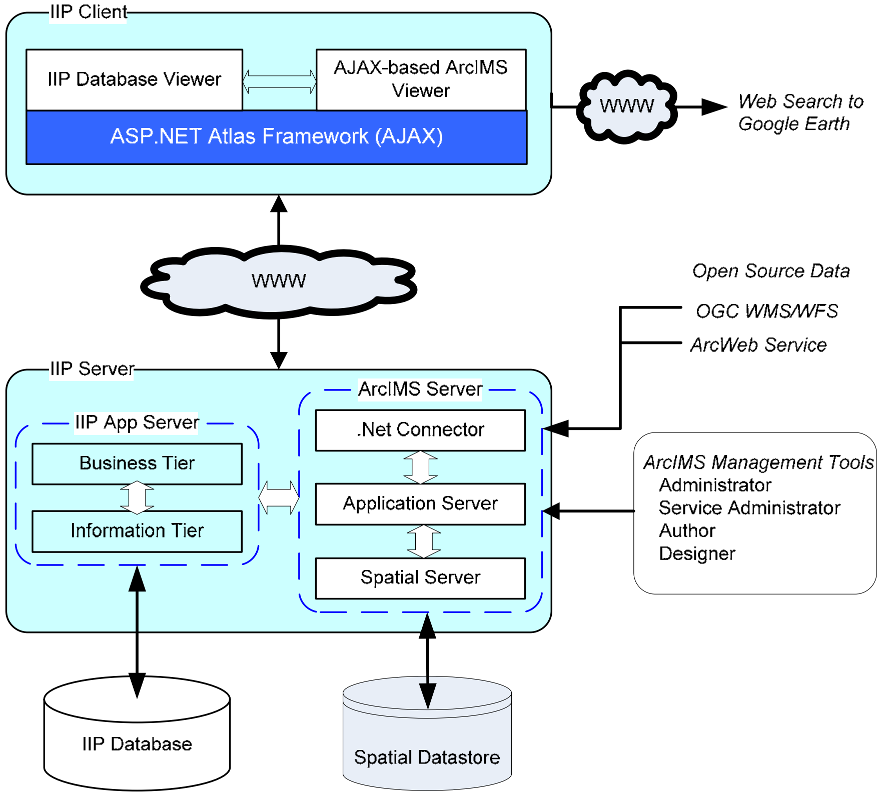

Figure 1. Integrated Information - System Architecture.

In Figure 1, open source Web Mapping Service (WMS) or Web Feature Service (WFS) features are requested through the browser. Web services use standard protocols to support interoperability between software solutions and data formats. Permission to access the information is also handled at this level to facility an exchange of data. The requests are fulfilled with appropriate privileges. Other open source services have been integrated such as Google Earth, to provide an example of how external sources of imagery and additional mapping information can be leveraged. The system architecture also provides the ability for unidirectional access of secure data in a controlled manner while integrating with open source data. The benefit of such an approach is an increase in user capacity of information and reduction in cost for data acquisitions. The key feature of the architecture is its design as a modular oriented approach where functionality is added through software libraries. The benefit of the modular design is to improve maintenance and expanded functionality including different GIS software, outside services for data, and other types of software to support visualization, network analysis, and safeguards related applications.

The IIP will provide users with an easy to use interface to access data world-wide, allowing filtering by attributes and geography based on queries for information by geographic areas, such as by country or location in a country. The proposed open architecture would allow utilization of existing and future tools, and integration with the ISIS Re-engineering Project (IRP).

3. Data Access & Acquisition Strategy

A fundamental aspect of the IIP is its ability to rapidly acquire diverse information through web services as well as internal database. The IIP approach is to move away from the silo mindset into new paradigm towards a distributed data services approach with the appropriate controls. The benefits of such an approach is minimizing the duplication of data, providing access to a broader range of information, and reducing costs through buying only data sets that are needed. Web services between departments can be established as both an internet and intranet to share information in a manner in which the database custodian maintains control. As an example, in some cases there is a need to display boundary maps by many different users. It is not always necessary to physically acquire the data; rather, a presentation of the data is what is needed. Web Mapping Services allows for the distribution of geographic data via pictures that can be used to visualize a map in the browser. The distribution of the data is significantly smaller in size than the actual physical data, reducing bandwidth requirements. Furthermore, appropriate online tools can supplement the preliminary analysis by accessing available online analytical tools, such as the Gaussian model for calculating dispersal of contaminant following a RDD event.

4. IIP User Interface

The IIP user interface is the display seen in the browser of the user. The design of the interface optimizes balance functionality, information content, and practicality, making it intuitive for users to operate. The primary use of the IIP is for decision makers to facilitate access to supporting information and collaborate with other users. A COP display layout was selected as the means to provide situation awareness in the form of location of interest, communication tools, pictures displays, and reference information on screen. A main control panel at the top of the browser allow the user to select appropriate panels. The panel concept organizes the information in modular structure. The individual display of panels can be manipulated by the users allowing them minimize and maximize features of interest. Two example illustrating the user interface is shown for the search engine to discover data and the user display showing panels that contain vector mapping, imagery, web cam, video, online text messaging, audio, documents, and tabular data in Figure 2.

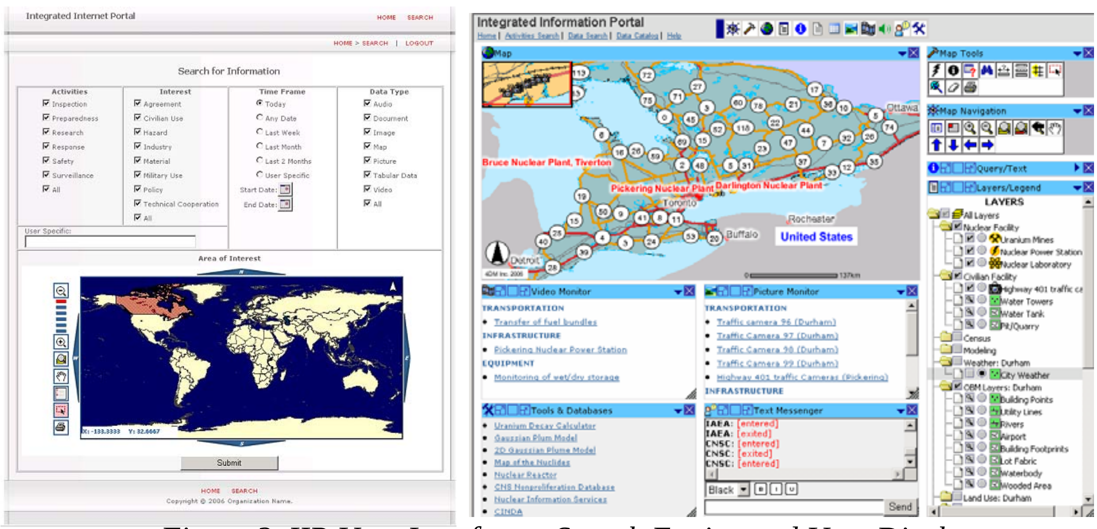

Figure 2. IIP User Interface – Search Engine and User Display.

IIP has the ability to establish multiple user interfaces that can be customized to specific features related to nuclear safeguards activities such as surveillance, inspection, and monitoring. Figure 3 illustrates a possible scenario of monitoring the delivery of fuel to a nuclear facility. In this scenario, the inspector is located on site while other users with appropriate security privileges can access the information in real time from their locations around the world. The map panel displays the facility location buffered at 1km as a precautionary security perimeter. Web cam panels provide monitoring information on the delivery route linking the user interface to live data to identify potential traffic related issues. Another panel receives text messages from the Inspector who is on-site as the online observer. A picture of the facility is displayed for reference allowing users to become familiar with the structure. At the same time, audio communication is exchanged with online experts and collaborators. Information is transmitted back and forth in real time.

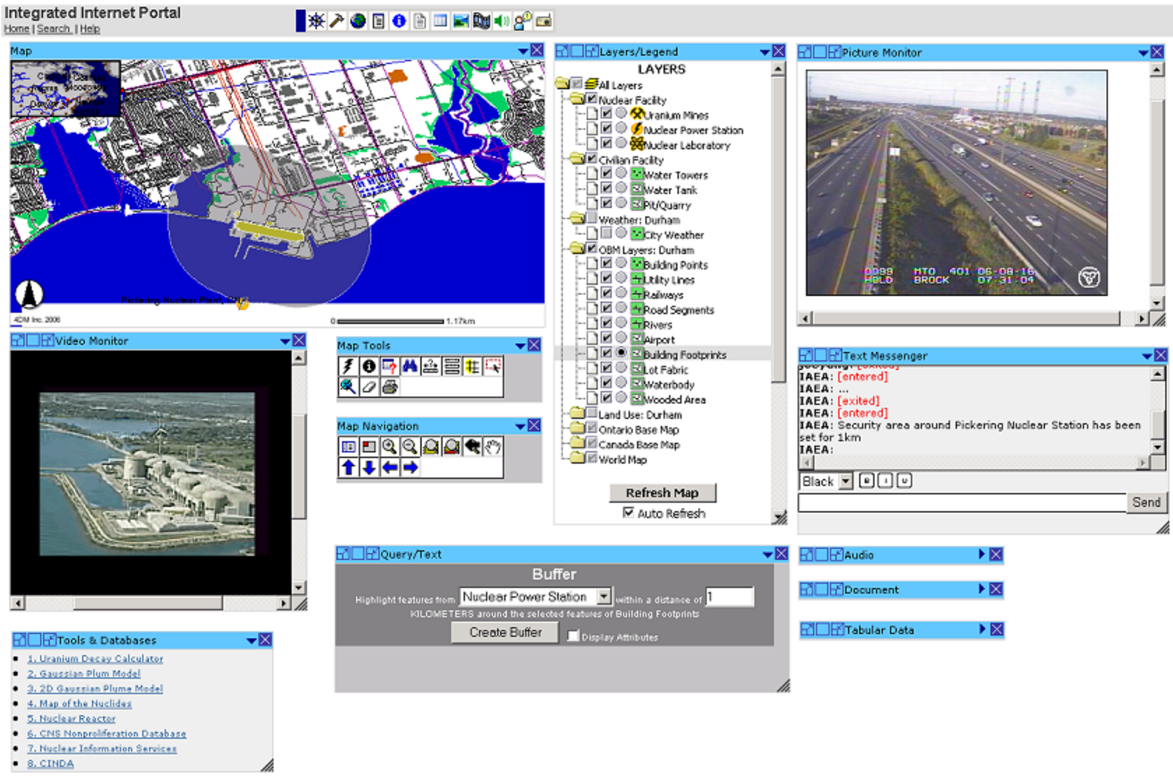

Figure 3. Monitoring of Nuclear Fuel to Facility.

In Figure 4, the user interface for IIP is able to link to external online tools such as a plume dispersion models and directly access open data sources such as Google Earth to build an extensive environment of information contents for decision makers and support personnel.

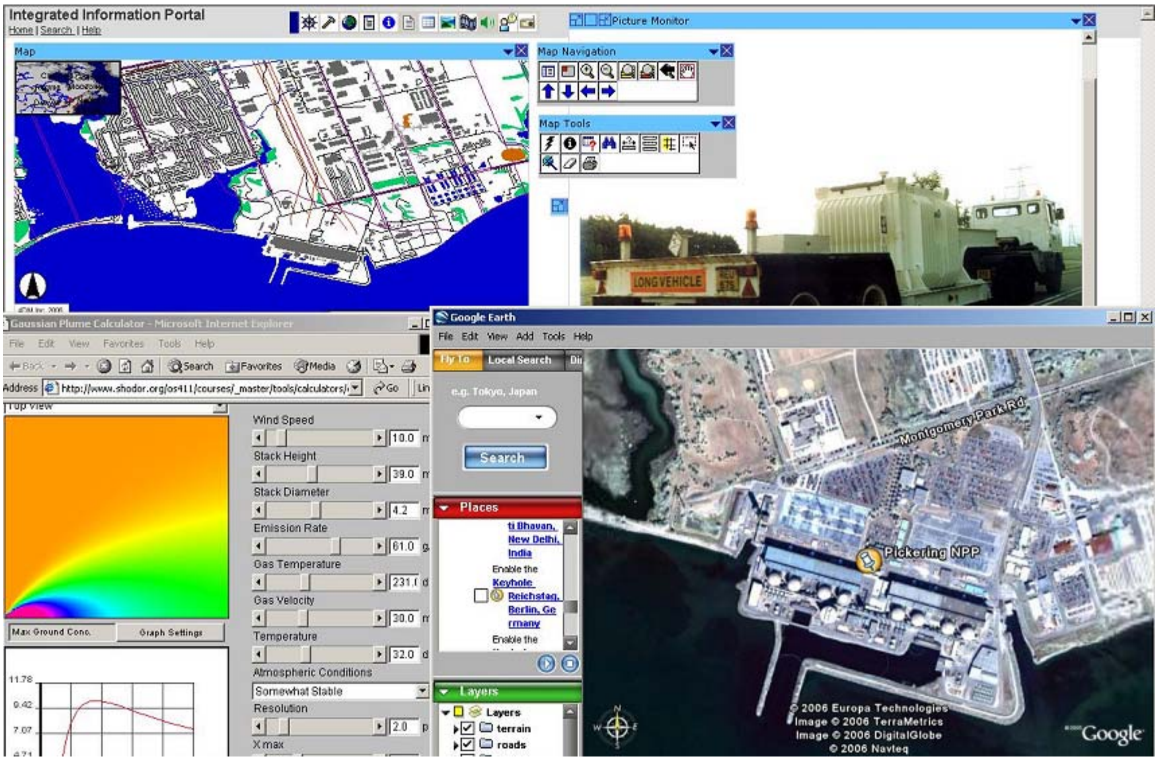

Figure 4. Integrating Online Tools and Data.

5. Discussions

The IIP portal delivers a number of significant benefits to the IAEA. Leveraging the open architecture approach can provide flexibility in accessing information and integrating a range vendor technology. Modular design allows the IIP to evolve with new functionality to meet the changing needs of the users. Accessing data and information on the internet compliments existing internal database and reducing cost, purchasing what is needed. A one-stop visual approach using mapping and imagery with ancillary data provide decision makers with a comprehensive perspective. Users from different locations can access the portal and view the same information simultaneous while being in constant communication. Wireless communication devices will allow field inspectors with satellite or land base networking to have in-situ support. Technology is available to implement IIP and customize for different users.

6. Concluding Remarks

The concept of IIP was created into a prototype using ESRI ArcIMS and Microsoft .Net technology to demonstrate the open architecture approach. Other web mapping technologies are also available that can provide similar capabilities such as Integraph GeoMedia, Autodesk MapGuide and open source applications such as Map Server. For the IIP, a user interface that linked web mapping, communications (text messaging, radio, audio), documents and tabular data was enabled to provide decision makers with an easy to use tool. Open source data was integrated demonstrate the rich content readily available to supplement existing data. Data sources such as Google Earth, Virtual Earth, web cams, OGC standard map services, and other online sources can be integrated into IIP. The technology exists to move this prototype forward into an operational tool at low risk. The data sources can be extracted freely and at minimal cost to support rapid and comprehensive information. The blending of technology and data into a portal that is easy to use and provides accessible information at one's fingertips is the emphasis of IIP. However, as discussed above, access to up-to-date external data and services (often cost-free) when needed would provide good leverage for available resources. The benefits of this approach can be seen through enhanced collaboration and communication, leading to an increase in knowledge sharing and credible decision making.

ACKNOWLEDGEMENTS

Helpful suggestions by Dr. Vincent Tao and assistance provided by Zia Haider are gratefully acknowledged. Google Earth and associated data presented in this paper, is for demonstration purposes only and is bounded by terms and conditions applied to its use.

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The IAEA and State declared safeguards information
Progress, plans and issues

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Abstract. The provision of information by States constitutes a key element for the implementation of IAEA safeguards. Reports on the changes, inventories and balances of safeguarded nuclear material have been received since the 1960s, beginning with INFCIRC/66-type safeguards agreements and followed by INFCIRC/153-type safeguards agreements.

The primary information provided by States is with respect to nuclear material accounting and the additional protocol. Nuclear material accountancy, while remaining a cornerstone of international safeguards, is now being combined with additional protocol information to provide a more complete and thorough picture of a State's declared nuclear programme. As a result of the evaluation of State declared information and all other safeguards-relevant information available (including the results of in-field verification activities and open source information), the IAEA is able to draw soundly based conclusions regarding the non-diversion of declared nuclear material and, if an additional protocol is in force, regarding the absence of undeclared nuclear activities in the State as a whole.

The needs of strengthened safeguards mandate accurate, complete and timely reporting, necessitating maximum use of the various types of State declared information. To these ends, the IAEA is in continuous contact on reporting matters with States. Furthermore, the IAEA trains Member State personnel on the associated reporting mechanisms and provides software to increase the quality of nuclear material accounting reports and a system to assist States in the preparation of additional protocol declarations. As the IAEA looks to the future, a major project is currently in progress for the complete re-engineering of all safeguards IT applications, including the nuclear material accounting system. The gains expected by updating the hardware and software are increased information utilization and correlation through the implementation of state-of-the-art systems.

This paper describes the progress, plans and issues associated with State declared information. It explains that, although processing of such information is commonly referred to as 'nuclear material accounting' this involves far more than just 'balancing the books'. It underscores the fact that processing a large diversity and volume of State declared information under strict requirements requires special skills, deep knowledge and vast experience in the application of safeguards. The paper also argues that the nature of State declared information is such that the skills mentioned above are essential not only for meeting the day-to-day needs of safeguards, but also for more in-depth analysis including extracting and analysing information needed for drawing conclusions within the context of the State evaluation process and the reporting in the Safeguards Implementation Report. Looking ahead, it stresses the importance of maintaining the necessary high levels of expertise for ensuring the optimum utilization of State declared information.

1. Introduction

State declared information is at the core of international safeguards and provides the IAEA with an extensive picture of States' nuclear activities. The combined existence of information obtained under safeguards agreements and other arrangements and from open sources and imaging is unique to international safeguards. This provides the IAEA with the necessary resources for drawing soundly based conclusions regarding the non-diversion of declared nuclear material and regarding the absence of undeclared nuclear material and activities.

What makes the totality of information available for IAEA analysis stand out are the components of the State declared information and the information obtained in the course of inspections and complementary access. Therefore, it is critical that the associated databases reflect the highest reliability, credibility and accuracy. An important element of processing State declared information is nuclear material accountancy which is a cornerstone of the IAEA safeguards system. This paper describes the progress, plans and issues related to State declared information and its importance for IAEA safeguards analysis and evaluation.

2. Progress and activities

2.1. Analysing State declared information

The IAEA has performed analyses of State declared information since the beginning of nuclear material accountancy to the present day, using the broad variety of information available. The work continues to expand and build on what has been done over the past years. Nevertheless, nuclear material accountancy retains its critical importance for the core business of safeguards. The IAEA is in the unique position of having available information provided by States, including nuclear material accounting reports, additional protocol declarations and data reported under other arrangements. Information supplied by States is analysed together with information obtained by the IAEA inspectorate, from open sources and any additional available information. The categories of State declared information on which analysis is performed are shown in Figure 1.

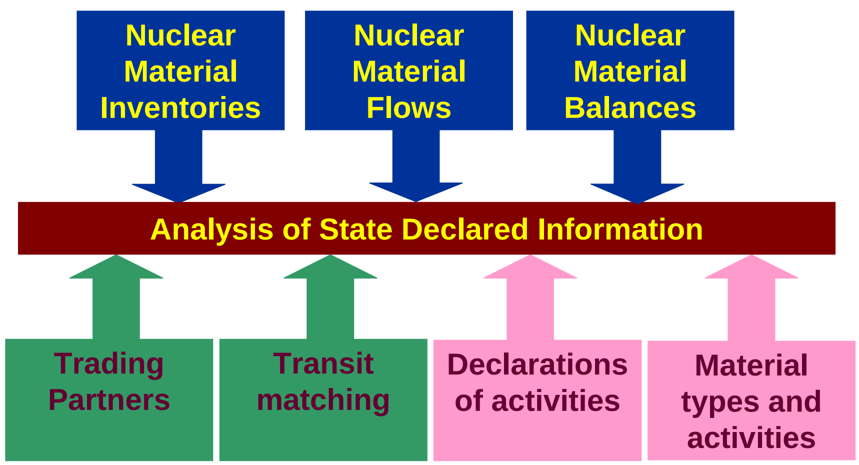

FIG. 1. Analyzed State declared information.

In the process of analysing State declared information, the task is to extract, analyse and correlate all pertinent information from a wide variety of sources. At the same time, the various mechanisms by which the data were reported must be kept in mind, to ensure the highest value of the analysis results. New analysis approaches have been developed, for example, based on the type of event occurring or where there is interest in a specific type of nuclear material or a desire to follow nuclear material as it relates to defined activities. Figure 2 illustrates a representative example (using artificial data) of analysis results following a portion of nuclear material in a State. In addition, by comparing State declared information with that obtained by other means such as open sources, it is also possible to make decisions on the relative value of a related source. This can be important in that not only is other information substantiated (or not) but there is also a level of accuracy and credibility that can be assigned to a compared open source.

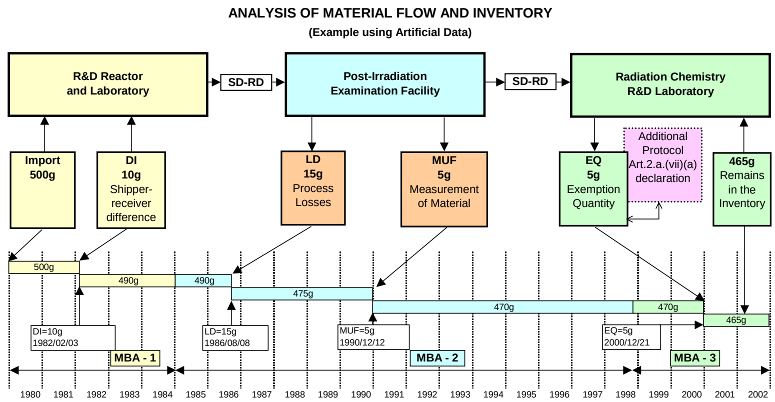

FIG. 2. Example of State declared information analysis.

The provision of information for use in the implementation of safeguards has increased significantly over recent years, reflecting the demands for more analysis and newly defined interpretations of State declared information. Information requirements that existed prior to strengthened safeguards have also continued, with some of those needs being enhanced as well. The routine provision of information based on that declared by States is summarised in Figure 3.

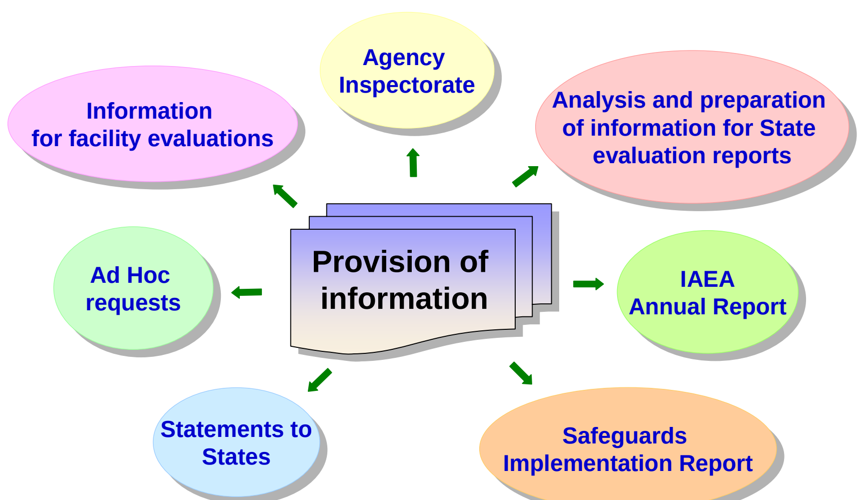

FIG. 3. Provision of information.

2.2. *Ensuring information accuracy and reliability*

The accuracy and confidence in the analysis results are directly related to the reliability and completeness of the relevant information. For nuclear material accounting information on balances, inventories and changes in material quantities and form, extensive analysis is required to ensure the correctness of the information declared by States. In addition, the correlation of imports, exports and domestic transfers requires extensive efforts and is a key added value component resulting from IAEA analysis. This ‘transit matching’ is one of the activities that only the IAEA can do on a world-wide basis.

Responsibilities for State declared information began in the late 1960s, with nuclear material accounting reports for safeguards implemented under INFCIRC/66-type agreements. The next major increase in the amount of State declared information occurred in the early 1970s, with the beginning of safeguards implementation under INFCIRC/153-type agreements pursuant to the Treaty on the Non-Proliferation of Nuclear Weapons (NPT). Since then, there have been increasing tasks related to State declared information, with the implementation of the additional protocol and the initiation of State Evaluation Reports (SERs) being recent major additions. While responsibilities have increased, the existing tasks have remained essential. Figure 4 illustrates the increase in these responsibilities. Clearly, while the increasing need for information has to be met, tasks that have been done earlier must continue at the same (or even higher) level.

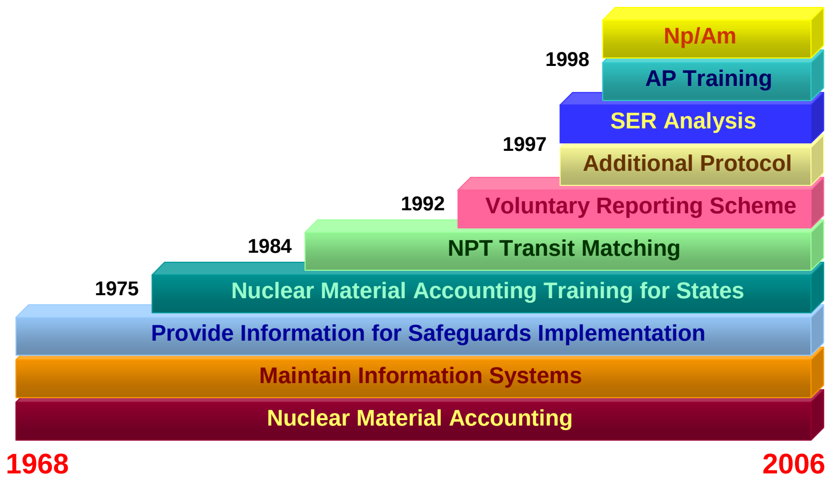

FIG. 4. Responsibilities since start of reporting.

All data provided by States over time are kept online in the IAEA Safeguards Information System (ISIS). Several databases on different platforms are maintained and kept up to date with respect to quality and content. It is necessary to ensure the quality of recently reported data and also to continually guarantee, to the highest degree possible, that the data, even the oldest, are at the same high level of usability.

2.3. Increasing data quality in State reports

As regards information accuracy and reliability, the IAEA actively takes steps wherever possible to improve the quality of State reports. The most direct mechanism used for ensuring reporting quality is the day-to-day communication with States on information received. This is accomplished by letter, facsimile, e-mail and telephone.

Another approach to addressing report quality is through the individual State systems of accounting for and control of nuclear materials (SSACs). Although the SSACs are responsible for the quality of data sent to the IAEA, through a Member State Support Programme project a software system — Quality Control Verification Software — was developed to perform a series of quality control checks on State data before dispatch to the IAEA The system was made available in 2006 to all States that provide nuclear material accounting reports.

With respect to the implementation of the additional protocol, a system — the Protocol Reporter — was developed to assist States in the preparation and submission of declarations to the IAEA. The software was distributed to those States who wished to use it, and it has been of significant assistance in the State implementation of the additional protocol. A project is underway to upgrade the system and a new version is expected to be available for distribution in 2007.

2.4. Training of State personnel and IAEA staff

Given the large variety of information that States are required to correctly provide to the IAEA, ongoing efforts are needed to maintain a high level of expertise at the State level as well as at the IAEA. Shortly after the beginning of reporting under agreements pursuant to the NPT, training courses were held for initial instructions on reporting using Code 10 for INFCIRC/153-type agreements. Just as responsibilities for continuing nuclear material accounting analysis have not ceased, the requirement remains to regularly provide training to State personnel and IAEA staff on nuclear material accounting and reporting.

With the advent of the additional protocol, it was necessary to develop lecture and workshop materials for training on the requirements for Article 2 declarations and on the timelines for those declarations. This has been the most significant recent addition to State declared information training responsibilities.

Each year, personnel responsible for State declared information participate in regional and international courses that are held to provide training on nuclear material accounting, reporting and the additional protocol. Training is also given to IAEA inspectors and other staff at headquarters. In addition, State representatives visiting IAEA headquarters receive training on nuclear material accounting and additional protocol declarations, when requested.

Another aspect of training is accomplished through the regular daily contacts between IAEA staff and State personnel. These contacts, usually made in the context of resolving a specific accounting, reporting or declaration query, offer an excellent opportunity for the IAEA to give detailed instructions and, on some occasions, explain general concepts and principles. Ongoing communications with States on reporting issues are very productive, serving all parties well.

3. Plans and enhancements

3.1. Expand analysis approaches

Analysis with respect to State declared information began with nuclear material accounting balances, inventories and transfers. As can be seen from the above discussion, analysis has increased significantly since the advent of strengthened safeguards. With increased requirements for the diverse use and interpretations of State declared information, there is a corresponding need to expand analysis activities as well. As experience is gained, the differing interpretations that can be taken, along with the conclusions that can be drawn, are enhanced. This is an ongoing process that will continually be refined as new and changed needs become known.

3.2. Providing expert input to ISIS Re-engineering

The ISIS Re-engineering Project (IRP) will result in the current ISIS applications and other systems being moved to a common operating system and hardware platform. For systems and databases developed over a long period of time, there is the need to migrate all data to new database systems and to implement existing applications on a new platform.

This project requires extensive input from the analysts of State declared information, to provide developers with the specifications and criteria needed to ensure that new systems are able to provide the necessary databases and tools to permit continuing the responsibilities for State declared information. The IRP is divided into several project levels, and the expert knowledge on State information is directly related to the reverse engineering of existing applications and to specifying user needs for new systems. Also of major importance is how State declared information is included in the Safeguards enterprise data model. Other areas of involvement are with information security, definition of a common user interface and selection of the hardware/software platform.

As a result of the IRP, all major applications and databases will be implemented on a single platform. This is expected to provide easier means to correlate and analyse reported information of various types, and a logical result is that responses to information requests will be addressed differently. Likewise, a significant improvement in analysis approaches and reduced efforts could be realised. A very active user community will be needed to ensure a smooth transition to the new system.

4. Issues and challenges

4.1. Utilizing large volumes of diverse information

The two primary factors affecting the analysis process are the large data volumes and the diversity of information. There are nuclear material accounting reports dating back to the late 1960s; however, most of the data are for NPT-type reports starting from the early 1970s until the present. Not only is the volume of data large, over the years there have been changes in reporting practices and procedures that have affected the use and interpretation of the retrieved information. Obviously, at all times the nuclear material accounting reports met the requirements for the implementation of safeguards, but when analysing data over such a long period of time, there are factors within the data that need to be considered. For example, States have been divided or combined, material balance areas have been changed, combined or divided (some more than once), modifications have been made in the codes used for reporting material descriptions and policy clarifications have been made on the use of certain inventory changes codes. A major challenge is selecting the most relevant data, applying the necessary factors over time and assimilating them with respect to other data sources.

The diversity of information has become more of an issue since the initiation of the Voluntary Reporting Scheme in the early 1990s. Since then, the number of databases, on various platforms, has increased and they all need to be utilized in the analysis process. Not only are additional protocol declarations in a separate database and platform, the recording of information (e.g. textual) is such that additional efforts are needed when relating declarations to nuclear material accounting reports.

4.2. Meeting complex information requirements

Over the years, the safeguards criteria have changed and with these changes the necessity for other interpretations of the data. A good example of this is the need to define inventories and transactions in terms of high enriched uranium (HEU) or low enriched uranium (LEU). Nuclear material accounting as defined in INFCIRC/153 (Corrected) does not specify separate accounts for HEU and LEU; all enriched uranium is accounted for in a single nuclear material balance. Even though nuclear material accounting reports provide information that can be used to correctly determine the enrichment in most cases, reporting formats as well as a lack of a specific indication if material is HEU or LEU are limiting factors. This results in the need for additional considerations when analysing and presenting information on HEU and LEU.

Because types of State data have different reporting requirements, correlating information is affected. Some of the information is in a structured form while other information is more of a textual nature. Also, even though the IAEA encourages commonality in reporting formats and codes, the voluntary nature of some information lends itself to deviating modalities and reporting mechanisms among States. Those differences need to be considered in the State declared information analysis process. Even with tightly structured data, such as that for nuclear material accounting, there are sufficient variations that affect the analysis process.

When an SSAC is established and operated in a State, not only do international obligations have to be met but also domestic regulations need to be implemented. Therefore, all SSACs differ in one way or another, which results in the differences in the way information is supplied by the State. This includes reporting modalities, data transmission mechanisms, speed of communication with the IAEA, quality of reporting and knowledge by the State of the nuclear material status in the individual facilities. All of this adds to the complexity of meeting the quality, reliability and timeliness requirements for data provided to the IAEA.

The various types of State declared information are shown in Figure 5, noting that some non-NPT information has been reported since 1968. As with the increase in responsibilities noted above, additional State declared information has been added to the repertoire of databases while, at the same time, responsibilities and tasks for "earlier" databases continue.

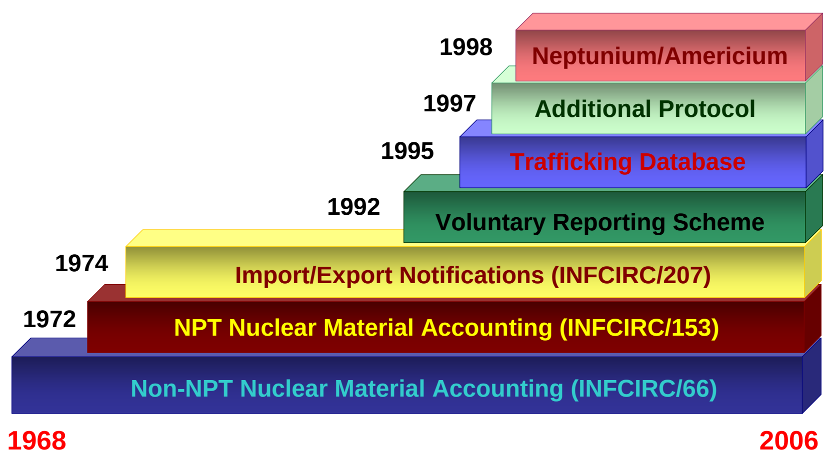

FIG. 5. Types of State declared information.

4.3. Managing knowledge and experience

For the IAEA staff who analyse State declared information, high levels of knowledge and experience are necessary — levels that can only be obtained through many years of working in the safeguards information field. To begin with, an in-depth knowledge of safeguards agreements and reporting arrangements is absolutely necessary, to be fully able to ensure the confidence level of the analysis results. In addition to understanding the differences in State reports and declarations, it is critical that staff also understand the interrelationships of the corresponding databases and the closeness of the related information.

A unique blend of advanced safeguards information skills and experience is critical for obtaining the best analysis results of State declared information. Beginning with good knowledge of the nuclear fuel cycle, the staff need to have extensive information technology experience, be able to work on various platforms and utilize a variety of software tools, as indicated in Figure 6. In addition to all of the above skills and knowledge, the analyst must be able to apply all of the factors that are "behind and within the data" — i.e. those aspects and caveats critical to the analysis but which cannot be readily built into software tools or databases.

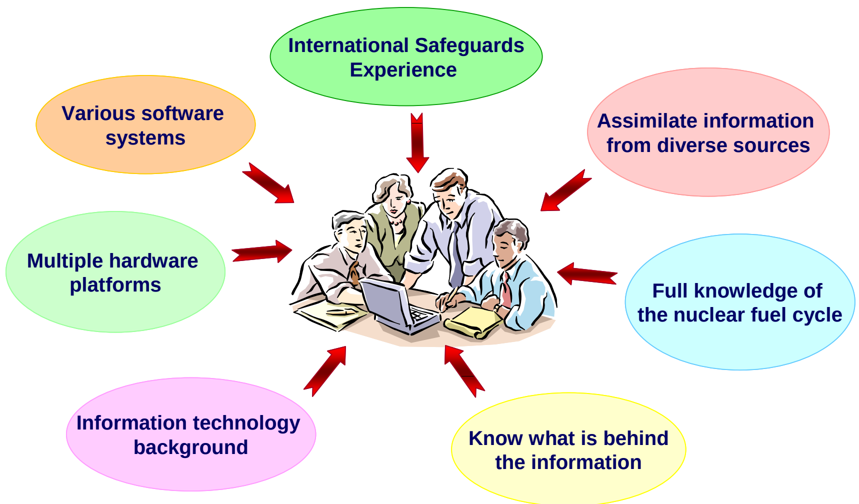

FIG. 6. Unique combination of skills and knowledge.

IAEA experience in this area confirms that it takes a number of years before one fully attains the level of expertise required for the type of analysis needed. Thus it is important that knowledge preservation be a key factor in personnel management decisions, to ensure the continuity of knowledge and to provide time for further development of expertise. As analysis demands increase, it is also necessary to have an active programme of expanding technical abilities, from both the safeguards and the IT perspectives. The exceptional level of expertise needed for analysis of State declared information mandates a centralized function whereby knowledge management, documentation and experience can be concentrated.

With all of the added information sources and the changes with respect to strengthened safeguards, it is imperative to continue emphasising the critical importance of nuclear material information, accounting and accountancy. These are a cornerstone of safeguards.

5. Summary and conclusion

State declared information is an essential component of safeguards analysis. The responsibilities, roles, tasks and available information have increased greatly over the past 15 years. Clearly, information analysis has progressed far beyond that performed in the 1970s, and represents much more than just "balancing the books".

The processing and analysis of State declared information need to be centrally located within the Department of Safeguards in order to guarantee a continued high level of expertise for the future. Highly qualified and experienced IAEA staff are absolutely necessary to ensure the long term, in-depth and viable use of the information.

Nuclear material accountancy system in Brazil

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Abstract. The Nuclear Material Accountancy System (NMAS) in Brazil is being redesigned in order to ensure higher automation levels in the inventory changes performed by the Material Balance Area (MBA) operators and, to increase the control of the State Authority over the nuclear material present in the country. The new project is based on a web system that is being developed right now in Brazil by the staff of CNEN. This system will be installed in a dedicated server under secure environment maintained at CNEN headquarters. Each user of the system will be provided with the necessary hardware and software to remotely access the system, inputting or retrieving data according to the access profile permission given to that specific user.

1. Introduction

CNEN is the governmental entity in Brazil responsible for the licensing of nuclear installations, the establishment of rules and regulations concerning the nuclear area and the control of all activities involving nuclear and radiological material.

The Safeguards and Physical Protection Coordination (COSAP) integrates the structure of CNEN, and is responsible for the accounting and control of nuclear materials, for the physical protection of facilities handling nuclear or radioactive materials and for the control of illicit traffic of nuclear and radioactive materials. COSAP is the State Authority for safeguards matters in Brazil and controls all the facilities by a NMAS and routine inspections.

In addition to the control of all nuclear material accomplished by the state safeguards system, the nuclear material after the starting point of application of international safeguards is also subject to regional safeguards of the Common System of Accounting for and Control of Nuclear Materials (SCCC) implemented by ABACC – Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials, and to international safeguards implemented by IAEA – International Atomic Energy Agency. Brazil is signatory to a comprehensive safeguards agreement (INFCIRC/435) that includes four parties: Brazil, Argentina, IAEA and ABACC.

Under this scope, COSAP maintains the knowledge of nuclear material inventory in all nuclear facilities in the country based on data collected from the MBAs and reports to ABACC, which is responsible to provide accounting information to IAEA, under INFCIRC/435.

In the following sections of this paper, a description and comparison of the present system and the upcoming project is made. It must be pointed out that the system is under development and it is possible that new ideas, not described in this paper, could be introduced later on.

2. The Present System

2.1 General Overview

CNEN regulations [1] state that a Material Transfer Authorization (ATM) shall be requested to CNEN with sufficient time in advance to the transfer of nuclear material, in order to give opportunity to the State Authority to inspect the material. The ATM is mandatory in case of international transfers, non-regular national transfers and those regular transfers that exceed one effective kilogram.

All inventory changes of nuclear material involving Brazilian operators are notified to the State Authority. The notification of nuclear material transfers is made through the Material Transfer Notification (NTM) issued by the shipper MBA, confirmed by the receiver MBA and sent to COSAP. In case of inventory changes other than transfers between MBAs, the notification is made by means of the general ledger, which is submitted monthly, to COSAP for auditing purposes.

2.2 Auditing Procedures

The NMAS is based on collections of MS-Excel spreadsheets for support documents, general ledgers and inventory lists that are prepared by the operators and submitted to the State Authority by conventional mail, including a floppy disk protected by password and/or encryption with those XLS files.

COSAP establishes the timeliness required for the receipt of those documents from the operators as the fifth working day of the following month in which the inventory change took place. The auditing of the files is performed manually, checking the consistency of the support documents against the general ledgers and the transfers between two different MBAs.

Today, all data sent by all operators spread over Brazil are kept in separate XLS files stored in a file server with no communication with Internet for security reasons. This means that activities such as the matching of shipper receiver data on nuclear material transfers are not automatically done and, all files received at COSAP must be manually copied into the file server.

2.3 Reporting to the Agencies

In order to unify information, the NMAS adopts the same terminology defined in the Common System for Accounting and Control of Nuclear Materials (SCCC) and in Code 10 [2] of the Subsidiary Arrangements to the INFCIRC/435 Agreement.

After the auditing of the data submitted by the operators, the State Authority has the obligation to send accounting reports to ABACC on all the nuclear material subject to INFCIRC/435. This information is sent by encrypted e-mail, using the 80 columns fixed format TXT files. Some macros in Excel were developed to transform the XLS files into the fixed format TXT files.

During inspections to MBAs with a great amount of inventory changes and/or a large list of inventory items, the State Authority prepares a file in TXT format with the inventory changes not yet informed through Inventory Change Reports (ICR) and/or the itemized list in CSV format and give them to the ABACC and IAEA inspectors, in order to help the inspection development.

2.4 Weaknesses of the System

At this point it is already possible to identify some weaknesses in this system such as redundancy of information, possibility to introduce errors, manual auditing of operators data and preparation of management reports.

Another difficulty faced by this system is the lack of standardization and integration of the documents format. The support documents and the itemized lists do not follow the same pattern for all MBAs, resulting in poor automation capability of the system. Management reports or data analysis based on the documents are also difficult to accomplish since the information stored is not integrated.

3. The New Project

Given the confidential nature of the safeguards information, the analysis and modeling of the nuclear material accounting procedures were conducted by the System Analysts Team of CNEN itself. This initial study pointed to the development of an entirely new system called E-GAMMA, the main features of which are described as follows:

3.1 Secure Access

The development of a WEB system with Internet remote access was chosen due to the need of on-line access to the information by both operators and COSAP. In addition, taking into account that the information dealt by the system is confidential, the data security in different levels of the application is being carefully studied.

One possible solution is presented in Figure 1. The device (D) works as a Firewall, IPS (Intrusion Prevent System), VPN (Virtual Private Network) and Anti-X (Spyware, Virus, etc.). The client requests access to the application using the Internet browser and D requests the client identification through some digital certificate (a Smart Card and its reader for example). Once this client is certified, D creates a virtual machine in the client station and establishes a VPN where encrypted data will transit, and finally turns available the initial page of E-GAMMA.

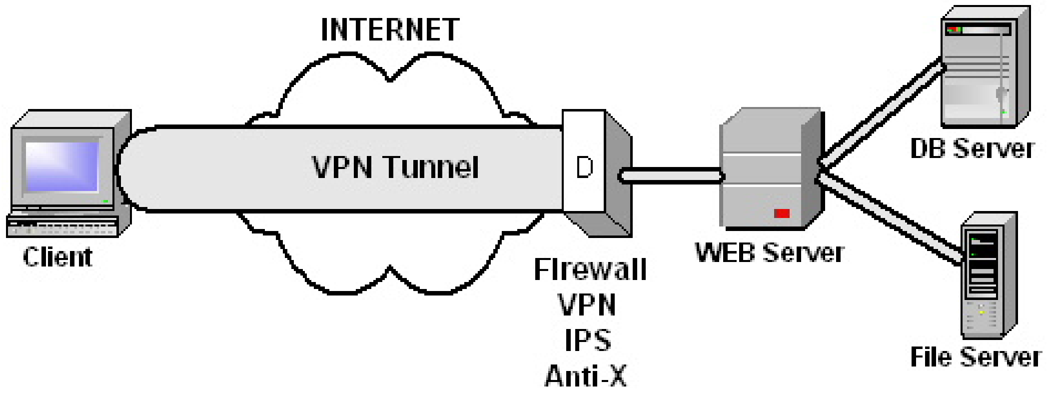

FIG. 1. Scheme of remote access to E-GAMMA.

Once the initial page of the application is accessed, new identification of the client is necessary, based on an internal system of strong passwords with creation/expiration dates and historic of changes controlled by the application itself. At the time the client logs out the application, D undoes the virtual machine and erases any traces of the application in the client station.

The application will run on three dedicated servers (Web, Database and Files) maintained at CNEN headquarters using modern security and optimization techniques.

3.2 Access Profiles

COSAP previously registers each user in E-GAMMA and associates this user to one or more access profiles. Each one of those profiles grants access to specific function menus that are summarized as follows:

(a) *Administrator:*

- manages the access to the system and maintains the users' record and the links to the different profiles,
- enables or disables profiles, users and passwords, e-mail delivery, management queries, etc.

(b) *Safeguards:*

- maintains the records of institutions, facilities, MBAs and inspectors in the system,
- updates the Table of Rules of each MBA according to information contained in the Facility Attachment (FA) and Design Information Questionnaire (DIQ),
- maintains the inspection agenda and performs queries and management reports,
- turns available the accounting reports files to the users of the ABACC profile.

(c) *Inspector:*

— performs queries to support documents of MBAs and prepares management reports,

— updates and manages the inspection reports and runs other queries to the system.

(d) *ABACC:*

— performs queries to the support documents, general ledgers and accounting reports,

— downloads accounting reports files (ICR, MBR and PIL) and turns them available for IAEA downloading.

(e) *IAEA:*

— performs queries to the support documents, general ledgers and accounting reports,

— downloads accounting reports files that were turned available from ABACC.

(f) *Operator:*

— prepares the support documents recording all inventory changes in the MBA,

— updates the itemized list of the MBA in the system,

— performs queries and prepares management reports.

3.3 Support to Different Languages

E-GAMMA is originally being developed in Portuguese using XML files to store labels, texts and error messages. The translation of those XML files content is foreseen to English and Spanish, in order to make it easier for ABACC and IAEA to access the system. At any time users will be able to change between languages.

3.4 Standardization of Support Documents

The support documents used by the operators were redesigned and standardized for each type of facility. These documents are generated using electronic forms available in E-GAMMA for recording and keeping track of all inventory changes occurring in Brazilian MBAs. The main documents are described as follows:

— *DTD – Domestic Transfer Document*: used for recording and reporting receipt at starting point, nuclear material transfers between MBAs in Brazil and eventual shipper-receiver differences.

— *DIE – Import/Export Document*: used for recording and reporting nuclear material transfers between Brazil and other countries and eventual shipper-receiver differences.

— *DMC – Category Change Document*: used for recording and reporting changes in the category of the nuclear material, resulting from blending, burn-up or enrichment processes.

— *DIDT – Exemption/De-Exemption/Termination Document*: used for recording and reporting these inventory changes. From this document E-GAMMA will print out the requests for those authorizations to the agencies.

— *DPPN – Nuclear Loss/Production Document*: used for recording and reporting these inventory changes.

— *DOVI – Other Inventory Changes Document*: used for recording and reporting all other inventory changes such as measured discards and accidental loss or gain of nuclear material.

3.5 Tables of Rules

E-GAMMA implements Tables of Rules for each MBA, which are updated by COSAP using the Safeguards access profile, based on information from FAs and DIQs. These rules may be for example: codes for nuclear material description, measurement basis, key measurement points (KMP), material category, maximum enrichment, etc.

This set of dynamic rules together with other rules already implemented in the system, such as Code 10 [2], will compose the knowledge basis needed by E-GAMMA to validate each support document created by the users of the Operator access profile. This way it is possible to reduce typing errors by filtering the information that can be input into the system for each MBA.

When beginning the preparation of a DTD, for example, the system limits the universe of possible receiving MBAs for that specific transfer, based on a compatibility analysis between the rules of both the shipper MBA and the receiver MBA.

3.6 Management of the Support Documents

The inventory changes taking place at the MBAs are entered into E-GAMMA by means of the electronic registration of the support documents. All documents possess an attribute representing its state on different phases of its life cycle, during the process of recording the inventory change.

For each document state, there is a specific functions menu available for each user profile that can be accessed. Figure 2 represents the States Diagram for a DTD since its preparation from the shipper MBA up to the final receipt by the receiver MBA, going through approval and correction phases if necessary.

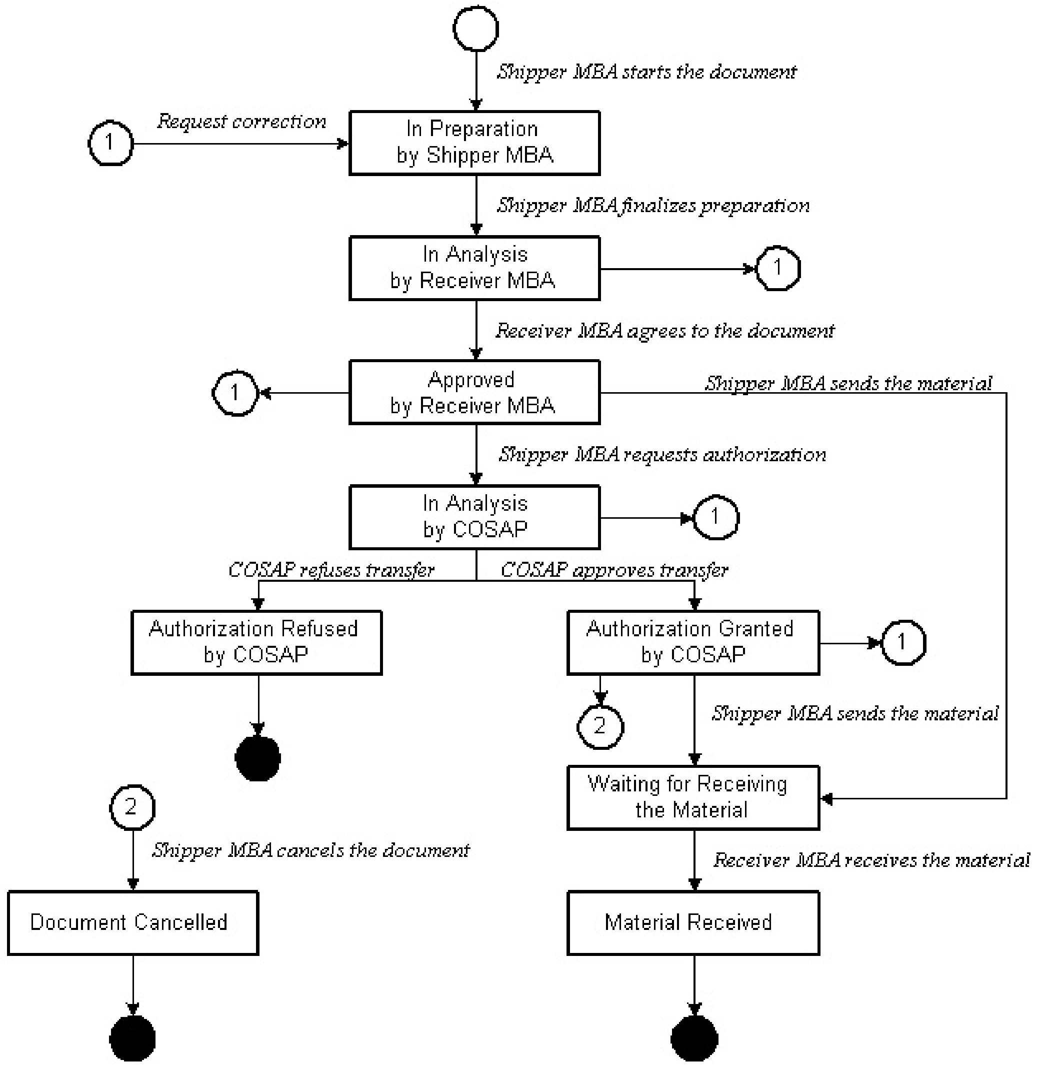

FIG. 2. States Diagram for a DTD showing different document status.

COSAP will be able, for example, to verify which documents are awaiting for authorization or agreement from the receiver MBA. E-GAMMA also uses alerts by e-mail that are sent to the users involved in the transfer, in order to signalize the state changes of the documents, making this management a lot easier.

3.7 Reporting to the Agencies

The basis for all information needed to prepare accounting reports is proceeding from the support documents. E-GAMMA will permit the issuing of reports from previous inspection periods to assist accounting record inspections, for eventual adjustments and corrections. At any time it will be possible to view the general ledgers of the MBAs by category of nuclear material, which is nothing else than the compilation of the inventory changes occurred in the period and ordered by date.

Once the auditing of the data by COSAP is finished, the access to the information (support documents, general ledgers and reports) will be granted to IAEA and ABACC for verification. After that, the accounting reports for the period will be concluded and concise notes may be attached to the reports by COSAP. At this time these files are then turned available for ABACC to download them in labeled format, instead of the fixed format that is used today.

The system includes also a command that can be used by ABACC to make the information available for IAEA download.

4. Conclusion

As a general conclusion, E-GAMMA will be far more secure, reliable and easier to work than the present system. Some advantages of the new project can be pointed out:

— Since all data will be stored in the system, there will be no need, as it happens today, for the operator to send documents in paper and floppy disks by conventional mail to the State Authority. This action reduces a lot the transit of data enhancing to a great extent the overall security and improving the response time of the system,

— There will also be a minimum amount of record errors thanks to the rules and filters based on the FAs and DIQs for each facility. These documents will also be stored in the system, in PDF format, and can be accessed at any time,

— There will be no more data redundancy. Transcription errors between support documents, general ledgers and reports will no longer exist since the information is entered just once,

— More elaborated queries to the information will be “just one click away”,

— The State Authority will have much more control and traceability on the inventory changes and, will also have some statistical analysis tools to evaluate SRD and Material Unaccounted For (MUF),

— Considering that both ABACC and IAEA will also access the system to retrieve information, a condition would be reached where the agencies could access the system previous to an inspection and do all the accounting verification remotely. This way the field inspection would be used just for the physical verification of the material and perhaps solve some accounting follow up actions. The consequence would be a significant reduction in costs and inspection efforts of the State Authority and the agencies, for the whole country.

As a final remark, CNEN also recognizes that the implementation of the new system will change a lot the accounting procedures existing at the facilities today, which means that the MBA operators will need training on dealing with the system.

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Implementation of IAEA safeguards agreement: Bangladesh perspective

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Abstract. Since long Bangladesh has been using atomic energy and related technology in different fields of health, industry, agriculture, research and education solely for peaceful purposes for socio-economic development of the country. For the purpose of international safeguarding Bangladesh has joined the Non-proliferation Treaty (NPT) in September 1979. In this connection, Bangladesh has also signed IAEA Safeguards Agreement (June 1982) and Protocol Additional (March 2001). Bangladesh is fully committed and transparent in its nuclear activities for ensuing safety and security of all radiation sources, nuclear materials and practices. Bangladesh, as a matter of policy, has incorporated in its legislation the NPT as well as other relevant international/bi-lateral conventions & agreement for compliance by any person or practice dealing with radiation sources & nuclear materials. Convention on physical protection of nuclear materials (1980) and CTBT (March 2000) has also been signed by GOB. Bangladesh has established and maintaining a system of accounting for and control of all nuclear materials in the country and monitoring the safeguarding activities within the framework of NSRC Act (1993) and Rules (1997). Bangladesh is always providing full co-operation to all safeguards related activities conducted by IAEA. So far no mismatch was found and no unwanted nuclear material was detected in Bangladesh by IAEA inspectors. In Bangladesh activities under safeguards agreement are functioning smoothly without any constraints. However, for further improving the effectiveness of safeguards activities proper attention should be given to increase the supervisory activities and technical capability of the regulatory body. Adequate infrastructure should also be developed to cope with the newly emerging situations like the illicit trafficking of nuclear materials and suspected nuclear terrorism that are of great global concern now-a-days. In this regard Bangladesh is going to consider ratification of a new international convention for the suppression of acts of nuclear terrorism.

1. Introduction

Bangladesh has long since been using atomic energy and related technology in the field of health, industry, agriculture, research and education. Their uses are increasing steadily for the socio-economic development of the country. Bangladesh as a signatory of both IAEA Safeguards Agreement and Protocol Additional in connection with NPT (Non-Proliferation Treaty) is fully committed for peaceful uses of atomic energy in the country for ensuring safety and security of all radioactive sources, nuclear materials and practices. Bangladesh, as a matter of policy, has incorporated in its legislation the NPT as well as other relevant international/bilateral conventions and agreements for compliance by any person or practice dealing with radioactive sources & nuclear materials. The nuclear regulatory body under the BAEC monitors safeguards activities within the framework of Nuclear Safety and Radiation Control Act (1993) & Rules (1997). Bangladesh has established and maintaining a system of accounting for and control of nuclear materials in the country under a national program. Government of Bangladesh provides full co-operation to IAEA safeguards inspectors for accounting and verification of any radiation source or nuclear material in the country. Bangladesh provides IAEA on regular basis requisite information as identified and required by the Protocol Additional through prescribed forms. So far statements of conclusion on the verification activities conducted by IAEA inspection teams are quite satisfactory and no mismatch was found in this regard. As nuclear program of Bangladesh is solely aimed at peaceful purposes, designated safeguards inspectors from IAEA enjoy adequate freedom for complementary access to any place on a site or a practice as per article 4 and 5 of the Protocol Additional under Safeguards Agreement. In Bangladesh activities under Safeguards Agreement are functioning smoothly without any constraints.

2. The Nuclear Regulatory Body of Bangladesh

2.1. The Bangladesh Atomic Energy Commission (BAEC)

The Bangladesh Atomic Energy Commission, a statutory body, was formed by the Presidential Order No. 15 of 1973. Section 6(1) of the order states that, “the function of the Commission shall be to do all acts and things, including research work, necessary for the promotion of the peaceful uses of the atomic energy in the field of agriculture, industry, development of related technology and appliances and for the execution of development projects involving nuclear power stations and the generation of electrical power there at...”. This order provides only promotional power to the BAEC. Simultaneously, the existing Nuclear Safety & Radiation Control (NSRC) Act (1993) also confers necessary power to the Commission to conduct all regulatory activities in the country.

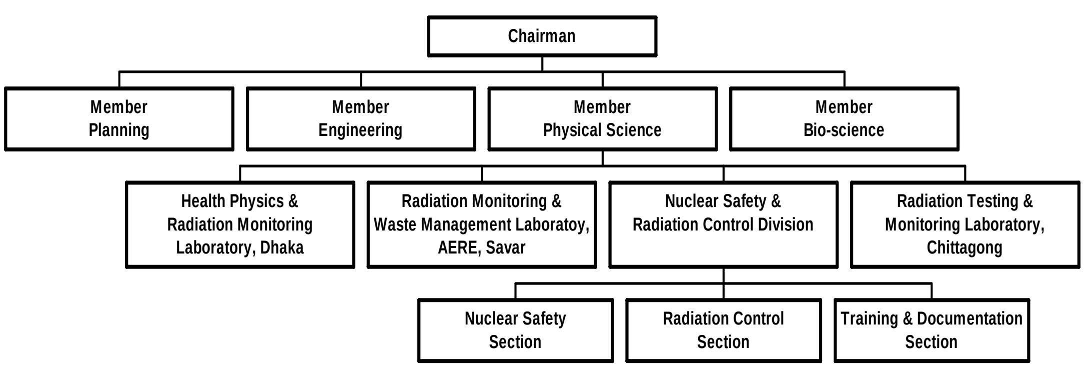
2.2. Regulatory Body

The Bangladesh Atomic Energy Commission (BAEC), vide section 3 and 16 of the Act, is the Regulatory Body and has the power to make necessary rules, formulate policies and implement the regulatory control in the country. This provision has further been illustrated in the Rules (1997) vide section 15 of the Act. The Commission may delegate any or all of its powers or responsibilities to any of its members. As a practice member (physical science) has been delegated such power. However, Chairman, BAEC is the head of the regulatory body. It is to be noted that the regulatory body is not yet fully independent.

2.3. Nuclear Safety and Radiation Control Division

The Nuclear Safety and Radiation Control Division (NSRCD) vide rule 4, is responsible to the BAEC for facilitating implementation of the provisions of the Rules (1997). It is in fact working as the secretariat for the regulatory body for performing all regulatory activities under the NSRC Act & Rules. The organogram of the regulatory body is shown below.

ORGANOGRAM OF REGULATORY BODY (BAEC)



3. Nuclear Safety and Radiation Control Act & Rules

3.1. Nuclear Safety and Radiation Control Act

It took considerable time to establish a legal basis for the control of uses of nuclear energy, related technology and radiation practices in the country. After long and concerted efforts, the NSRC Act (No.21) was promulgated on July 22, 1993 to provide for ensuring nuclear safety and radiation control in the country. The Act confers all necessary powers to the BAEC to regulate uses of atomic energy, radiation practices and the management of radioactive waste.

3.2. Nuclear Safety and Radiation Control Rules

Pursuant to Section 16(1) of the NSRC Act, BAEC formulated draft rules in 1995. The NSRC Rules, after review of the concerned Ministries and Establishments and vetting of the Ministry of Law, Justice and Parliamentary Affairs, were notified in the Bangladesh Gazette on September 18, 1997. The rules are quite comprehensive for the control of radiation sources and nuclear materials in the country.

4. Policy of the Government of Bangladesh (GOB) regarding Safeguards

The Government of Bangladesh is fully committed for peaceful uses of atomic energy and related technology in the country. As a gesture of this commitment Bangladesh has signed NPT in September 1979. It is to be mentioned here that Bangladesh is the only country in the region to sign the above treaty. In this connection Bangladesh has further signed IAEA Safeguards Agreement in 1982 and consequently signed Protocol Additional in 2001. By signing up of these agreement & protocol Bangladesh has entered into the international obligations for safeguarding of all radiation sources, nuclear materials and practices. Bangladesh is fully transparent for any of its nuclear activities or practices for physical inspection and verification in accordance with the statute and safeguards system of IAEA. Bangladesh, as a matter of policy, has incorporated in its nuclear regulatory legislation the NPT as well as other relevant international/bi-lateral conventions and agreements for compliance by any person or practice dealing with radiation sources and nuclear materials. Convention on physical protection of nuclear materials has also been signed by GOB in 1980. Bangladesh has further signed CTBT (Comprehensive Test Ban Treaty), which has come into force in March 2000. The prime objectives of the above treaties and agreements are to ensure safety and security of all radiation sources, nuclear materials and practices in the country thus the implementing the provisions of safeguards. The Government provides free access and full co-operation to all designated IAEA safeguards inspectors for Bangladesh for monitoring, accounting & verification of any radiation source or nuclear material in the country. Bangladesh is going to consider ratification of a new international convention for the suppression of acts of nuclear terrorism.

5. Safeguards Inspection & Verification by IAEA

Safeguards inspectors are designated for Bangladesh following the procedures of Article 85 under the safeguards agreement. Upon receipt of advance notification by IAEA in accordance with the Article 83 of the agreement proper arrangements are made for the inspectors to perform their functions of safeguarding like verification of previous records, visit and physical inspection of the facility and material balancing area according to the provisions of the agreement and requirements of the Protocol Additional. As nuclear program of Bangladesh is purely aimed at peaceful purposes safeguards inspectors enjoy adequate freedom for complementary access to any place on a site or a practice as per article 4 & 5 of the Protocol Additional. The inspectors use their own equipment & techniques for the purpose of safeguards. Sometimes they take help of the services available in Bangladesh including the use of equipment when it is required. Any new problem and creation of a new material balancing area are elaborately discussed to address them with adequate data and information. Inspectors also call on the regulatory body at the final step to discuss the safeguards related matters to improve the system and communication in this regard. IAEA inspection teams get full co-operation during their stay in Bangladesh from all relevant corners. So far statements of conclusion on the verification activities conducted by IAEA inspection teams are quite satisfactory and no mismatch was found in this regard.

6. Safeguards Activities in Bangladesh

Bangladesh has established and maintaining a system of accounting for and control of all nuclear materials in the country. As a part of national safeguards program the facilities in the country using nuclear materials have been divided into two material balancing areas differentiating by two separate codes namely, BD-A for facilities within Atomic Energy Research Establishment (AERE), Savar and BD-Z for all other locations or facilities outside AERE. To maintain records of all nuclear materials for these two separate entities two senior officials from Bangladesh Atomic Energy Commission (BAEC) have been designated to perform their responsibilities in this regard. Information is gathered from these two areas to prepare separate reports, which are then combined together to make one report for sending to IAEA through authorized communication channel of BAEC. The nuclear regulatory body under BAEC also monitors safeguards activities within the framework of Nuclear Safety and Radiation Control Act (1993) & Rules (1997) specifically for compliance of the rules 88 through 90. The Government provides full co-operation to IAEA safeguards inspectors for accounting and verification of any nuclear material in the country. Bangladesh provides IAEA on a regular basis requisite information as identified in article 2 and in prescribed periods as mentioned in article 3 of the Protocol Additional through prescribed forms. So far statements of conclusion on the verification activities conducted by IAEA inspection teams are quite satisfactory and no mismatch was found in this regard. As nuclear program of Bangladesh is purely aimed at peaceful purposes, designated safeguards inspectors from IAEA enjoys enough freedom for complementary access to any place on a site or a practice as per article 4 and 5 of Protocol Additional under safeguards agreement. In recent years several environmental samples from different sites of the country were tested in IAEA laboratories and no unwanted nuclear material was detected.

7. Suggestions for improvement of the safeguarding systems

In Bangladesh activities under the safeguards agreement are functioning smoothly without any constraints. For further improving the effectiveness of safeguards activities attention of the appropriate authority is being drawn to the following points:

a. State regulatory control over inventory of nuclear materials under the safeguards agreement should be strengthened by increasing the supervisory activities. Technical capability of the state regulatory authority should also be increased in respect of safeguards.

b. At present Bangladesh has technical deficiency specifically in respect of environmental monitoring. For conducting such monitoring procurement of modern NaI(Tl) detector(s) coupled with MCA can be very much useful.

c. Amidst global concern on illicit trafficking of nuclear materials and suspected nuclear terrorism, respective authority should take proper steps including the introduction of border gate monitoring systems to control the above clandestine activities.

d. Any data received by satellite imagery system of safeguarding requires to be verified by the state regulatory authority before arranging of any special trip by the inspectors for avoiding any kind of misinterpretation of the data that might occur and thus making inspections cost effective.

e. Venders would be required to provide full information (quantities, composition and other specifications) of any nuclear material integrated with an equipment or machinery during transfer into and out of a state.

8. Conclusion

The nuclear program of Bangladesh is solely dedicated to the peaceful purposes for the socio-economic development of the country. For the fulfillment of national commitment safeguarding activities are functioning smoothly within the framework of NSRC Act and Rules. In terms of international obligations under different agreement and conventions Bangladesh is fully transparent and co-operative to any inspection and verification of its activities and practices by international body like IAEA. To further improve the effectiveness of safeguards activities the regulatory body should be fully independent. To overcome the technical deficiency of the nuclear regulatory authority of Bangladesh IAEA help and co-operation is necessary in this regard. Proper attention should be given to develop adequate infrastructure to cope with the newly emerging situations like illicit trafficking of nuclear materials and suspected nuclear terrorism which are of great global concern now-a-days. In this regard Bangladesh is going to consider ratification of a new international convention for the suppression of acts of nuclear terrorism.

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The Euratom Nuclear Material Accountancy System for the implementation of the safeguards agreements, the Euratom Treaty and Euratom agreements with third States

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Abstract. As early as 1957 the founding members of the European Atomic Energy Community established the Euratom Treaty with the aim of promoting the peacefull uses of nuclear Energy, while at the same time introducing controls aiming at enhancing safety and security and preventing uses of nuclear material other than the ones declared. Chapter VII of the Euratom Treaty foresees the implementation of Safeguards on nuclear material used in the Community and establishes details of the Euratom Safeguards system. Euratom safeguards is entrusted to the European Commission and is based on two pillars: (i)Assurance that Nuclear Material is not diverted from its declared use [Art. 77(a)] (ii)Compliance with the obligations that the Community has undertaken as a result of agreemnts with international organisations the third states [Art. 77(b)]. Further, Art. 101, gives to the European Commission the mandate to negotiate, sign and implement International agreements. The agreements with the IAEA and the main supplier states are such examples. The cornerstone of Euratom Safeguards is the Euratom Nuclear Material Accountancy system, which, according to Art. 79 of the Treaty, is based on a Commission Regulation. The Commission Safeguards Regulation, is the legal basis and gives practical guidance as to the contents, format and frequency of the records and reports that the operators of Nuclear Installations have to submit to the Commission. These reports form the adequate basis for the Commission in discharging its responsibilities vis-à-vis the IAEA and third states. The paper describes the Euratom NMA system and its recent overhaul to take into account additional reporting requirements stemming from the Additional Protocol and also from the IT developments.

1. Introduction and Legal framework

The Euratom Treaty, signed in 1957 by the six original Member States of the European Union, is one of the three founding Treaties of the European Union, the other two being the European Economic Community and the European Coal and Steel Community.

In the above Treaties, the Governments of the European Union Member States (the original six and every new member state) are committed to ensure the economic and social progress of their counties by common action to eliminate obstacles and barriers in Europe.

It is within this context and in the early years of Nuclear Technology developments that the Euratom Treaty came into force, with the aim of creating the appropriate conditions for the rapid development of nuclear industry in Europe and the advancement of the cause of peace.

The European Atomic Energy Communities' (Euratom) task is fulfilled by:

- Promoting research
- Establishing uniform safety standards

- Facilitating investment and the establishment of nuclear installations
- Ensuring regular supply of nuclear fuel
- Making certain that nuclear materials are not diverted to purposes other than those for which they are intended
- Establishing with other countries and international organizations relations, fostering progress in the peaceful uses of nuclear energy.

The last two idents are explicitely addressed in the Euratom Treaty, namely in chapters VII (Safeguards) and X (External relations) and their implementation is trusted to the Commission of the European Community.

Chapter VII establishes a safeguards system, to assure "… that nuclear material is not diverted from its intended use as declared by the user". A Commission Regulation (currenctly No 302/2005) describes the nature and extend of the requirements on the operators of nuclear installations, so that the above assurance can be obtained.

At the same time the Community has entered into obligations with the IAEA, for the implementation of non-proliferation safeguards, and with third states for co-operation and facilitation of nuclear material trade (Art. 101 of Euratom Treaty).

The Euratom Nuclear Material Accountancy stystem was developed in order to facilitate the implementation of all above obligations without putting extra burden on the operators and/or the authorities of the EU member States.

2. Description of the requirements

2.1. Assurance that nuclear material is not diverted from its declared use [Art. 77(a)].

Under this Article, the operators of nuclear installations are obliged to declare to the Commission the use of nuclear material, to submit advance notifications of changes of use, of imports and exports, and regular reports of inventory changes and inventory lists. The assurance is obtained by analysing the information/reports submitted by the operators, and by comparing it with the physical reality during inspections.

It should be noted that in the Euratom system the notion of nuclear material extends from ores to waste, and that the same safeguards requirements apply to the civil nuclear material of the nuclear weapon member states.

2.2. Compliance with the obligations that the Community has undertaken under the Agreements with the IAEA [Art. 77(b)].

Under this Article, the Commission has to discharge a number of obligations, as detailed Safeguards Agreements (INFCIRC 193 for the 23 non-nuclear weapon member States, INFCIRC 290 for France and INFCIRC 263 for the UK), as supplemented by the corresponding Additional Protocols.

These obligations are equivalent to an INFCIRC 153 type of agreement, and are accomplished via a set of communications/reports from Euratom to the IAEA and by coordinated Euratom and IAEA inspections.

Due to their nature, some Additional Protocol requirements are accomplished by the Member State authorities either via Euratom or directly to the IAEA with information/copy to Euratom.

Obligations under INFCIRC415 (information on production, inventories and international transfers of certain nuclear material) are also accomplished by Euratom.

2.3. Compliance with the obligations that the Community has undertaken under Agreements with third States [Art. 77(b)].

The Community has concluded nuclear co-operation and supply agreements with a number of countries (i.e. USA, Canada, Australia, Japan, Ukraine, Uzbekistan, etc). In these agreements the European Commission guaranties the apporpriate use of the material transfered via the implementation of safeguards throughout the European Union. A system of advance notifications, prior consent and annual inventory reports is implemented by the Commission, based on declarations originating either from the supplier of the material or from the operators in the European Union.

3. Description of the system

In order to fulfill the requirements described above, the European Commission relies on its Nuclear Material accountancy system. This system is implementated by a Commission Regulation, describing the nature and extent of the requirements on the operators that will permit the discharging of all Community responsibilities.

The main elements of the system are:

3.1. The Basic Technical Characteristics (BTC) of an installation. The BTCs, which are equivalent to the DI questionnaire and include a description of the use of the nuclear material, are sent from the operators to the Commission, are checked for completness and are subsequently forwarded to the IAEA.

3.2. An annual outline of the programme of activities, which enables the Commission to plan its safeguards activities and co-ordinate them with the IAEA.

3.3. The operating and accounting records kept by the operators containing information on the quantities, category, form and composition of nuclear materials, their actual location and the particular safeguards obligation as provided in the agreements with third countries, together with details of the recipient or shipper when nuclear materials are transferred. These records serve also the requirements of the IAEA under the Safeguards agreements.

3.4. The accountancy reports (i.e. ICR, PIL, MBR) which are sent from each operator to the Commission.

These reports, representing a yearly volume of 1,3 Mio records, are loaded in the Euratom nuclear material date base (called CMF-3).

The reports are checked by the Euratom accountants for consistency and completness and form the basis for the reports required by the IAEA and the third countries.

3.5. The advance notifications of exports and imports of source and special fissile materials, sent by the operators to the Commission within prescribed dead lines, and which contain all information relevant to the implementation of the system of prior consents and notifications required by the agreements with third countries and the IAEA.

3.6. Concerning the declarations under the Additional Protocol, it should be noted that the Commission is responsible for all declarations related to nuclear material, and the site declaration, while for the other declarations a special, country specific, distribution of responsibilities between the Commission and the Member State, is implemented.

For the nuclear material related AP declarations (i.e. ores, source material,waste and exempted material) the same Euratom Accountancy system is used.

4. The recent overhaul of the Euratom Accountancy system.

As explained in the introduction the legal basis for the Euratom Nuclear Material Accountancy system is a Commission Regulation. The first legal documents to describe the declaration requirements of Euratom Treaty safeguards were Regulations no 7 and 8 of 1959. In 1976, Regulation 3227/76 was issued with the aim of adapting the reporting requirements to enable the Commission fulfill its obligations under INFCIRC193. Regulation 3227/76 underwent a couple of minor amendements, and until 2005, when it was replaced by the new Regulation, it served its purpose well.

The basic reasons for issuing a new regulation were three:

- The necessity to improve the data reporting and data treatment quality
- The wish to integrate in the regulatory text provisions of a Euratom “waste policy”
- The need to adapt the content to the requirements imposed by the expected entry into force of the Additional Protocol

Further it was felt that several long stnading wishes for small changes in some of the reporting requirements could be addressed.

The drafting of the new Regulation started in 2000, and after a lengthy approval discussion in the European Council in which Member States’ experts also participated resulted in a document including significant changes compared to the original Commission proposal. The Regulation was published under the number 302/2005[1].

The Regulation was accompanied by a second document including implementation guidelines. These guidelines which were published later as a Commission recommendation with the document number “2006/40/Euratom”[2], were not given a binding legal status, yet they include advice for operators that “if followed, it would satisfy all Regulation reporting requirements”.

4.1. The new reporting format

One important requirement of the new Regulation is that operating records should be kept in electronic form, and reports must be transmitted in electronic form using XML labelled format. Limitations of field content caused by the fixed field length (e.g. batch-id, weight) no longer exist. This format is insensitive to positional changes and null data elements do not have to be reported at all.

A number of new data elements were introduced (e.g. transaction id, line number, check-sum) to improve internal consistency and easier follow up of accountancy corrections, while others (e.g. document reference, advance notification reference, campaign identifier) allow cross referencing between operators’ records and reports. New inventory change codes were introduced (e.g. for waste transactions, for nuclear transformation, for corrections) in order to reflect current reporting needs.

The Commission recognized from the onset that the implementation of a new reporting format would require substantial effort by the operators. For this reason a transitional period of 3 years, extendable to 5 years was provided, but also a number of projects were initiated with the aim of assisting the operators to cope with the new requirements.

For the 10 states that acceded to the European Union in May 2004, the Commission developed the software package called "ACCESS" with the following functionalitities:

- Local NMA database (Oracle)
- Data input via Editor or Xloader
- Local Rulebook validation
- Automatic calculation of MBR and stocks
- Electronic reporting to Euratom/rapid feedback
- Secure data transfer (encryption)

The Commission decided to also provide assistance to the operators of Nuclear installaitons in the 15 old Member States towards meeting the requirements of the new reporting format, and has initiated the project ENMAS, mainly addressed to small and medium sized operators.

- For **small operators**, "ENMAS light" is a stand alone application, with manual NMA editor, validation of the good syntax of the declarations, some type of Rulebook, XML report generation and viewing. Encryption and sending by email can be independent. Testing of ENMAS light has been completed, and from October 2006 is available for use.

- For **medium sized operators**, who currently produce the accountacy reports as a by-product of their NM management system, ENMAS will use the ACCESS features in a simplified platform. ENMAS is expected to go in production early in 2007.

4.2. The Euratom waste policy

As stated earlier the Euratom Treaty covers all nuclear material from mining to waste disposal. In order to take into account the above and the reporting requirements for waste introduced with the Additional Protocol, a number of novelties were introduced, in particular:

- The declaration of BTCs of Waste treatment or Waste storage installations when these are separate from other installations

- Definition of three Waste inventory change codes "discards to the environment", "transfer to conditionel waste" and "transfer to retained waste"

- Extension of the IC code TU (Termination of Use) to quantities of nuclear material diluted to concentrations below those of Uranium in Ores.

For the declarations of shipments and receipts of conditional waste, which is done outside the standard ICR format, special forms were included in annexes.

4.3. The Additional Protocol related provisions

As explained in paragraph 3.6 above, the Commission is responsible for reporting to the IAEA all AP related information concerning nuclear material. In order to cope with the above requirement, a number of changes were introduced in the new Regulation:

- For mines and concentration plants, a simplified BTC form and an annual reporting Annex were introduced

- For source material, the only missing data element ("Chemical form") was introduced in the ICR declarations

- Material exempted from IAEA safeguards, is still under Euratom safeguards. In order to cope with this, the derogation mechanism was introduced, whereby MBAs granted a derogation continue to report to Euratom using simplified reporting annexes. Euratom has then the responsibility to request exemption from IAEA safeguards

- For waste containing HEU, Pu or U-233, simplified reporting annexes were introduced.

5. Conclusion

The Euratom Nuclear Material Accountancy system integrates all the requirements imposed on operators by international legislation, and offers to the Commission the possibility to discharge its responsibilities stemming from the Euratom Treaty and the Agreements with International organizations and third countries.

The implementation[3],[4] of the recently published new Euratom Regulation is advancing well, introducing thus a modern accountancy and reporting standard along the 25 member states of the European Union.

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Development of safeguards technology for lab-scale advanced fuel cycle facility at KAERI

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Abstract. KAERI(Korea Atomic Energy Research Institute) has been developing the DUPIC (Direct Use of PWR spent fuel in CANDU) fuel cycle and ACP (Advanced Spent Fuel Conditioning Process) technology for the purpose of a spent fuel management. A safeguards system has been applied to the R&D process for fabricating DUPIC fuel directly with a PWR spent fuel material. Safeguards issues to be resolved were identified in the areas such as international cooperation on handling a foreign origin nuclear material, technology development of an operator's measurement system of a bulk handling process of a spent fuel material, and a built-in C/S system for an independent verification of a material flow. All these safeguards issues have been resolved. The lab-scale DUPIC facility (DFDF) safeguards system was successfully established under an international cooperation program. The ACP has been under development at KAERI since 1997 to tackle the problem of an accumulation of spent fuel. The concept is to convert spent oxide fuel into a metallic form in a high temperature molten salt in order to reduce the heat power, volume, and radioactivity of the spent fuel. The main objective of the ACP is to treat the PWR spent fuel for a long-term storage and eventual disposal in a proliferation resistant and a cost effective way. Moreover, the electrolytic reduction method of the ACP can contribute to an innovative nuclear energy system as a key technology for the preparation of a metallic fuel. Since the inactive tests of the ACP have been successfully implemented to confirm the validity of the electrolytic reduction technology, a lab-scale hot test will be undertaken in the ACP facility (ACPF) to validate the concept. This paper summarizes the main features and the current status for developing the safeguards system of the ACP and DUPIC. It is expected that a deployment of these advanced safeguards technologies would be useful for an engineering-scale fuel cycle facility.

1. Introduction

To become a developed country in the 21st century, Korea is facing a Trilemma of its economics, energy and environment caused by the expanding public conflict between an energy security for a national economy development and environmental problems. In Korea, the energy consumption rate is steadily increasing, however, most of the energy resources are from foreign countries. Therefore, nuclear power, which is economic and free from green gas emissions, is highly recognized in a country with insufficient natural resources like Korea. Currently the electricity generation by nuclear power is approximately 40% in Korea and it is expected to increase to 60%. However it is inevitable that such an increase in the nuclear power portion will result in a concern for a management of the spent nuclear fuel. Therefore it is a priority to develop a fuel cycle technology that can utilize spent nuclear fuel as a semi-domestic energy resource, reduce the amount of fossil fuel imports and drastically curtail the amount of high level waste.

DUPIC fuel technology has been developed by Korea, Canada and the United States in order to utilize the PWR spent fuel in the CANDU reactor.In 2000, KAERI refurbished a part of a hot-cell of the irradiated material examination facility and established the DUPIC fuel development facility (DFDF) to process the PWR spent fuel and fabricate the DUPIC fuel on a laboratory scale. In this facility, about 25 pieces of remote fabrication equipment are installed including safeguards equipment such as a DUPIC Safeguards Neutron Counter and neutron monitoring equipment [1].

The advanced spent fuel conditioning process (ACP) has been under development at KAERI since 1997 to tackle the problem of the accumulation of spent fuel. The concept is to convert the spent oxide fuel into a metallic form in a high temperature molten salt in order to reduce the heat power, volume, and radioactivity of the spent fuel. The main objective of the ACP is to treat the PWR spent fuel for a long-term storage and eventual disposal in a proliferation resistant and a cost effective way. Moreover, the electrolytic reduction method of the ACP can contribute to an innovative nuclear energy system as a key technology for the preparation of a metallic fuel. Since the inactive tests of the ACP have been successfully implemented to confirm the validity of the electrolytic reduction technology, a lab-scale hot test will be undertaken in a couple of years to validate the concept. A preliminary study on the safeguardability of a pilot-scale ACP facility was performed. As a result of the study, our conceptualization of the facility features and material flows across the ACP facility lead us to conclude that a safeguards system could be designed to meet the IAEA's detection goals and to provide an independent verification scheme [2]. Based on the results of a safeguards implementation in the DFDF hot cell, the reference safeguards design conditions are established for the ACPF.

This paper addresses the main features of the DUPIC safeguards development status and also the future prospects of a safeguards implementation to the ACP facility at KAERI.

2. DUPIC Safeguards System Development and Implementation

2.1. Technology Development of DUPIC Safeguards System

In 1999, KAERI established the DFDF. Safeguards system of the DUPIC facility faced many issues, namely: 1) there was no directly applicable NDA measurement technology for the DUPIC process material due to a high radiation interference, 2) there was no DUPIC specific safeguards criteria available in IAEA reference publications, and 3) there were no appropriate C/S methods for a positive identification of a material movement in and out of a hot cell. DUPIC safeguards R&D group summarized the technologies necessary for a process material accounting as follows: (1) neutron coincident counting methodology could be an alternative fabrication facility, (2) Near Real Time Accountability system would be required due to the inaccessibility to a process material under a high radiation environment, and lastly (3) unattended continuous monitoring system comprising of a radiation detection and image recording would be built into the system for maintaining a continuity of the knowledge of a material flow as a dual C/S system.

2.2. Material Accounting System

Normally item facilities need to account for a nuclear material based on burn up data and code calculations. Bulk facility needs to account for a nuclear material by a weighing, chemical analysis of the representative samples, and/or an NDA measurement. But in the case of the DUPIC process the NDA method is adapted to account for the entire process material. In order to maintain a data coherence and material balance between item facilities and a bulk handling facility, Curium Ratio method is introduced to account for the DUPIC process material, in which the accounting data from item facility is normalized to accounting data of the DFDF NDA measurement system authenticated by the IAEA. All the process material is measured with one NDA instrument called DSNC (DUPIC Safeguards Neutron Counter) employed in the Curium Ratio technique.

A drawing and photo of the manufactured DSNC are shown in Fig. 1. The DSNC, a well-type neutron coincidence counter, is for inferring the amount of Cm-244 from measuring spontaneous fission neutrons at various process stages in the DFDF. The DSNC design focused on all types of DUPIC process material that are remotely measurable (CANDU type bundle, powder, rod-cut, hulls, and wastes) in a hot cell during a lab scale operation [3].

The DSNC together with the DUPIC Safeguards Neutron Monitor (DSNM) have been approved by the IAEA as the official nuclear material measurment and monitoring deveices of the DFDF and have been used for IAEA and national inspections.

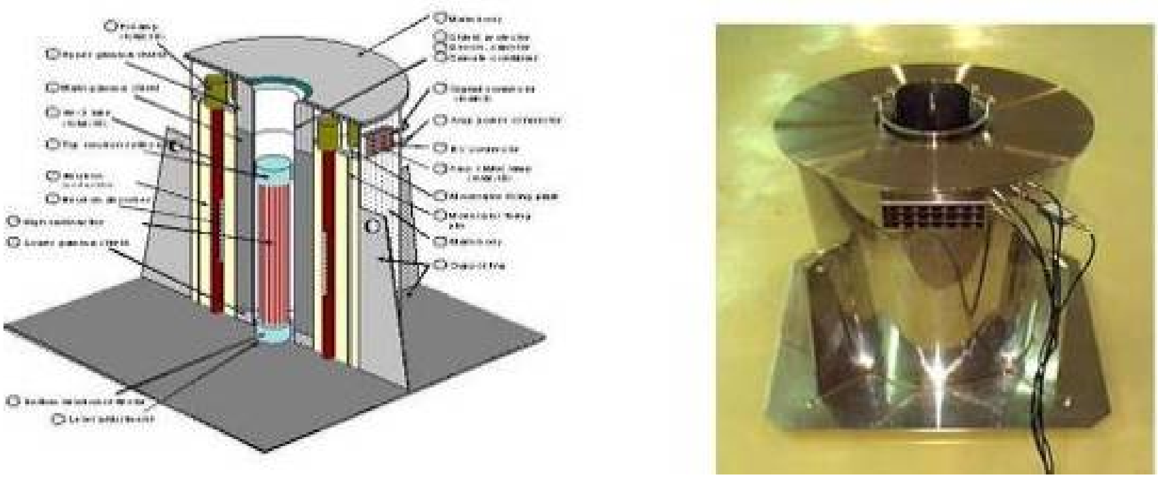


FIG. 1. Drawing and photo of DUPIC Safeguards Neutron Counter.

3. ACP Safeguards System Development and Implementation

3.1. ACPF Establishment

A cold demonstration of the electrolytic reduction concept was performed in a lab-scale mockup system (5 and 20 kgU/batch of natural uranium) in 2003 and 2004 and the recovery yield for uranium was more than 99%. For the active demonstration of the ACP technology, a lab-scale hot cell (ACPF) construction was started in early 2005 and completed in the same year. The ACPF hot cell is composed of two cells: a process cell (M8a) and a maintenance cell (M8b). All the equipment of the process is located in the process cell, as shown in Fig. 2.

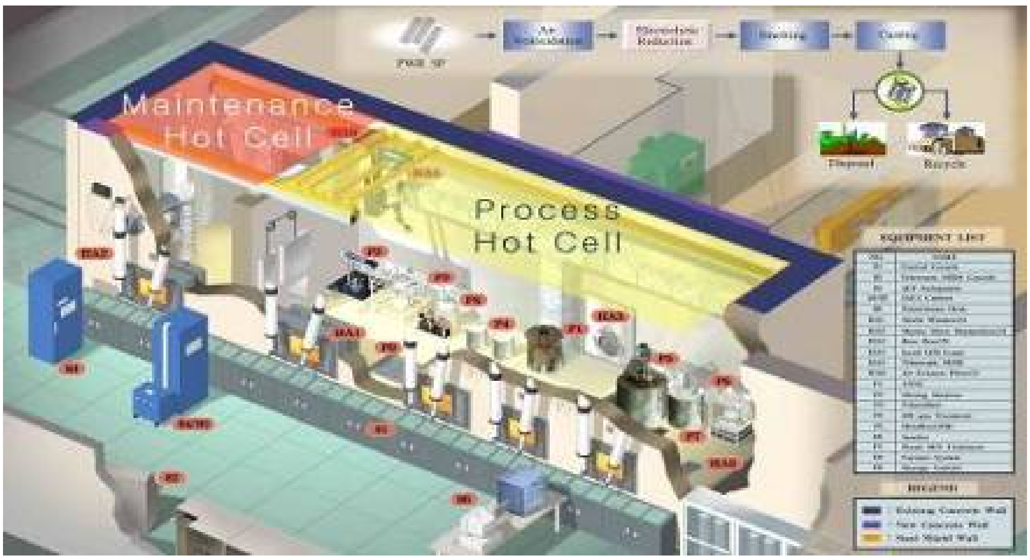


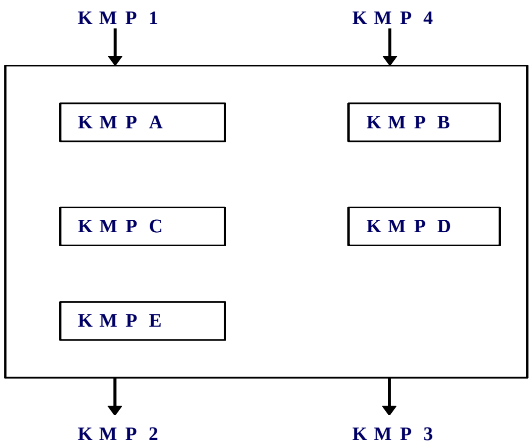
FIG. 2. ACPF.

3.2. Key Measurement Points

For the material control and accountability of the ACPF, the material balance area (MBA), material flow pattern, key measurement points (KMPs), and inventory during a material balance closing were defined as shown in Fig. 3.

The facility operator does a material accounting based on some declared values for the feed materials; destructive chemical analyses for mixed oxides; and NDA measurements for U metals, recyclable scraps, and disposable waste streams. IAEA verification is preceded by a shutdown and cleanout of all the major process areas and an accumulation of the inventories at a few locations shown in Fig. 3 as KMPs.

IAEA verification will employ attributes and variable measurements, preferably NDA measurements including the use of a remote monitoring data transmition. CoK will be maintained by seals at the transfer points, radiation gate monitors and a surveillance within the cell, on the transfer route and surrounding areas.



Inventory KMPs
KMP A : Rod-cuts storage
KMP B : Pellet storage
KMP C : Powder storage
KMP D : Metal Ingot storage
KMP E : Scrap & Waste storage

Flow KMPs
KMP 1 Receipt NM, de-exemption, accidental gain
KMP 2 Shipment, exemption, accidental loss
KMP 3 Transfer to RW
KMP 4 Transfer fr.om RW

FIG. 3. Material Flow and Key Measurement Points at the ACP Facility.

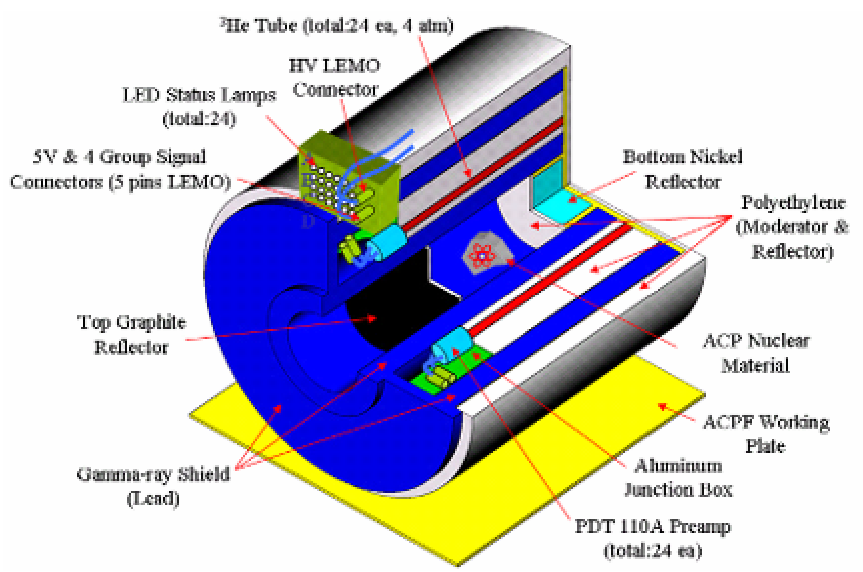


FIG. 4. MCNPX 3D model of the ASNC.

3.3. *NDA Equipment*

Based on the results of a safeguards implementation in the DFDF hot cell, the reference safeguards design conditions are established for the ACPF. Basically, the nuclear material accounting will be performed by ASNC(ACP Safeguards Neutron Counter), which is the same concept as adopting a curium ratio technique [4]. The ASNC has been developed by upgrading a well-typed NDA system by considering a high-efficiency, high-accuracy and remote friendly manner for the hot cell operation. The prototype design is shown in Fig. 4.

3.4. *Material Control and Accounting*

The nuclear material accounting procedure of the ACPF is shown in Fig. 5. An initial PWR spent fuel assembly in the pool of the PIEF (Post-Irradiation Examination Facility) is selected and pulled out from the pool. The assembly is moved into the hot cell of the PIEF and then a gamma scanning is performed. The cutting points of the fuel rod are marked after this scanning and then the rod is cut in the PIEF hot cell. The burnup data of the fuel rod is obtained through a gamma scanning and the contents of elements such as U, Pu, Cm, Cs, Sr, etc. are calculated by using the ORIGEN code. The rodcuts are put in to the Padirac cask and moved into the ACPF hot cell through the rear door of the process cell. Not only the rod cuts but also the processed materials such as the U_3O_8 powder, metal powder, smelted metal ingot, waste salt ingot, and hulls are measured with the NDA equipment of the ACPF to perform a nuclear material accounting of these materials.

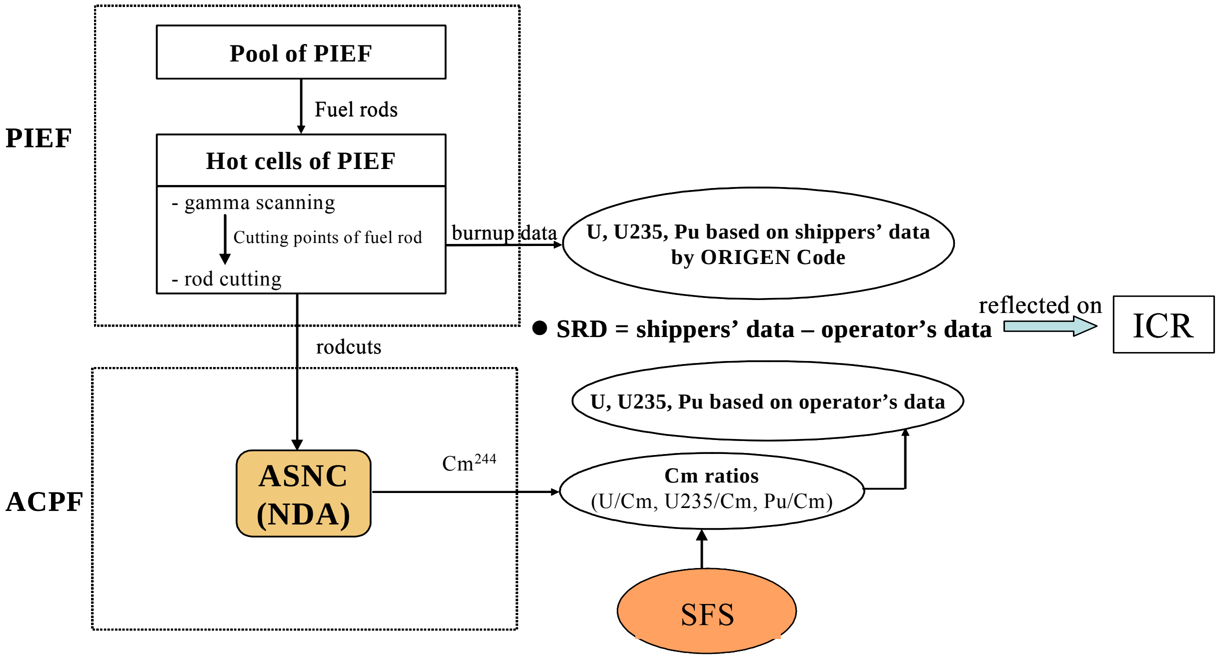


FIG. 5. Nuclear Material Accounting Procedure of the ACPF.

3.5. *ACP Nuclear Materials*

ACP nuclear materials can be categorized into 3 groups according to their characteristics for a neutron leakage multiplication ***M*** and (α, n) to a spontaneous fission neutron ratio ***α***: (1) metal ingot, (2) waste salt, and (3) spent fuel rod-cuts, UO_2 and U_3O_8 powder, and hulls [5]. Table 1 shows the parameters of the ACP nuclear materials calculated by the MCNPX, SOURCES and ORIGEN-S codes. Based on this result, all the ACP nuclear materials can be accounted for with singles and doubles rate measurements by the ASNC. The dominant source of a spontaneous fission neutron from ACP nuclear materials is the ^{244}Cm element. The mass of ^{239}Pu and ^{235}U contained in each nuclear material can be obtained from the ratios of ^{239}Pu/^{244}Cm and ^{235}U/^{244}Cm, called the *Cm ratio*, and the measured ^{244}Cm mass of each nuclear material [6]. The Cm ratio of each material can be established by a chemical analysis. Though the samples of group 3 have different values of ***M*** and ***α***, only one calibration curve will be produced for the nuclear materials of group 3 because there is no significant difference among the values of ***M*** and ***α*** of the nuclear materials of group 3. Therefore three calibration curves are required to measure the mass of ^{244}Cm of all the ACP nuclear materials. Doubles rate for a nuclear material is measured by the ASNC and the mass of ^{244}Cm of that material is calculated from the calibration curve. Then finally, the mass of ^{239}Pu and ^{235}U contained in that nuclear material is obtained from the Cm ratio for that material. However the Cm ratio calculated by using the ORIGEN-S code will be used until the real calibration samples are prepared and the chemical analysis data is available.

Table 1. Parameters of ACP nuclear materials.

Group	ACP nuclear materials	Net leakage multiplication, ***M***	(α, n) to spontaneous fission neutron ratio, ***α***	Criteria for grouping
1	Metal ingot (20 kg)	1.178	0	$1 < M < 2$ and $\alpha = 0$
2	Waste salt (10 kg)	1.001	0.656	$M = 1$ and $0.5 < \alpha < 1$
3	UO_2 powder (100 g)	1.019	0.016	$M \approx 1$ and $0 < \alpha < 0.5$
	U_3O_8 powder (100 g)	1.019	0.230	

3.6. Performance Test Rresults of the ASNC

The neutron detection efficiency, spatial efficiency profile in the cavity, HV plateau characteristic curve, deadtime coefficients, die-away time, predelay, and gate width were determined by measurements with two ^{252}Cf spontaneous fission neutron sources. One of the sources is a weak source (CVN101) whose activity was about 1.6 μCi (09/02/06) and the other was a strong source (C7-427) whose activity was about 0.9 mCi (15/02/06). The strong source was cross-checked by the LANL N-1 group. Table 2 shows the performance test results of the ASNC. From the test results, we can see that the efficiency is high enough to obtain reasonable values for singles and doubles rates of the spontaneous fission neutrons.

Table 2. Performance test results of the ASNC.

Item	Specification	Item	Specification
Cavity diameter (Inner diameter of basket)	20 cm	Efficiency (HV @1650 V)	21.04%
Outside diameter	72 cm	Flat counting zone	33 cm (σ: ±1.32%)
Gamma-ray shield	Inner lead: 6 cm Outer lead: 6 cm	Gate	64 μs
Preamp	PDT110A® (24 ea)	Predelay	4.5 μs
^{3}He tubes	Reuter-Stokes®	Die-away time	63.0954 μs
(a) Number	24 ea	Double gate fraction, f_d	0.4921
(b) Active length	20 inch	Deadtime coefficient	A = 0.8227E-6 B = 0.1679E-12
(c) Diameter	1 inch	Operation HV	1650 V
(d) Gas fill	4 atm		

According to the present schedule, the PWR spent fuel will not be brought into the ACPF until 2008. Instead, a simulated fuel will be used to fulfil the ACP experiments during the cold test period. Preliminary calibration curves for the metal ingot, waste salt, and UO_2 powder are produced to perform the nuclear material measurements in the early stages until real calibration samples of the ACPF are prepared. These samples are going to be produced from the first ACPF hot operation. Preliminary calibration curves are produced from the measurement data for the ^{252}Cf sources [7]. These curves are also generated by using the MCNPX code. The preliminary calibration curve for the measurement of the ^{244}Cm mass of the ASNC is shown in Fig. 6.

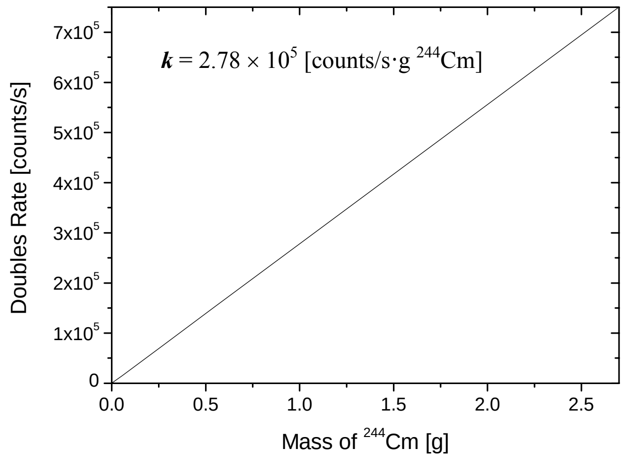


FIG. 6. Preliminary calibration curve for the measurement of ^{244}Cm mass by the ASNC.

3.7. *C/S System for the ACPF*

The C/S system layout of the ACPF is shown in Fig. 7. There is one neutron monitor for the rear door of each hot cell to monitor a movement into and out of the ACPF. The IAEA cabinet will be installed after a completion of the facility attachment for the ACPF. ACP material measurement system and surveillance system will be integrated into a single safeguards system in the next R&D stage.

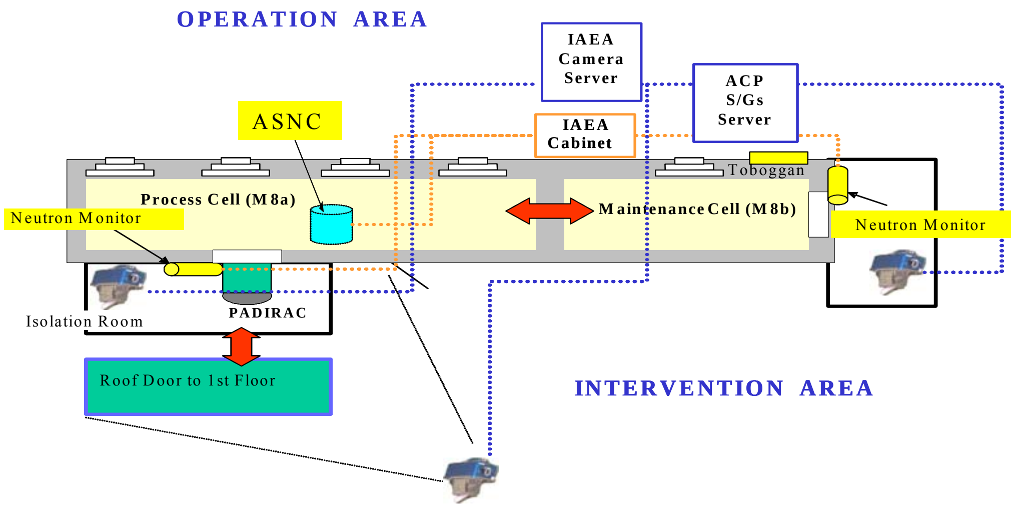


FIG. 7. C/S System Layout of the ACPF.

4. Summary

KAERI has been developing the DUPIC fuel cycle and ACP technology in a spent fuel bulk handling facility. A lab-scale DUPIC facility safeguards system was successfully established under an international cooperation program. With the implementation experience gained from the lab scale facility and a further advancement of the safeguards technology development, a model safeguards system for an engineering scale bulk handling facility is foreseen in the future. Based on the performance test of the ASNC, it seems that the Cm accounting method is reliable enough to be used for a measurement of the ACPF process materials. Preliminary calibration curves were generated from the measurement data of the ^{252}Cf sources. These calibration curves will be used until real calibration curves are produced with calibration samples from the early stage hot operation of the ACPF. An on-site experiment for the preliminary test of a remote monitoring system is being carried out and implemented as a part of a safeguards system of the ACPF. The R&D efforts for the ACP safeguards system will be continued to stabilize and enhance the performance of the system through the integration of a material accounting system and a remote monitoring using VPN.

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Development of the Epithermal Neutron Multiplicity Counter (ENMC)

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Abstract. Japan Atomic Energy Agency (JAEA) has developed the Epithermal Neutron Multiplicity Counter (ENMC) under a joint study program with Los Alamos National Laboratory (LANL). The ENMC was developed in order to mitigate the measurement uncertainty of plutonium in impure MOX generated from the MOX fuel fabrication process. The ENMC can measure not only thermal neutrons but also epithermal neutrons. The ENMC has a high measurement efficiency of neutrons and a very short die-away time which lead to a considerably lower measurement uncertainty. JAEA performed functional tests to confirm the high measurement performance. The test results showed that the ENMC total measurement uncertainty could be reduced to 0.5-0.7% by optimizing of measurement conditions. JAEA considers that by utilization of advantages of nondestructive analysis (e.g. small sampling error), the ENMC will be applicable as an effective measurement system for material accountancy of the impure material which is difficult to take a representative sample. Furthermore, by application of the ENMC to the safeguards, the efficiency of inspection activities will be improved. This report describes the results of the functional tests and applicability of the ENMC.

1. Introduction

In JAEA's MOX fuel fabrication plant, scrap materials are generated during the process (e.g. pellets rejected in the pellet inspection). Almost all the scrap materials are recycled into the process as part of the feed material after the dry recovery treatment. But a small amount of scraps recovered by clean-up activity of the glove box and equipment which includes relatively high light-element impurity. These impure materials are stored in the storage and recycled in the future. The amount of plutonium in pure scrap materials is determined for material accountancy by destructive analysis (DA). While the Plutonium Scrap Multiplicity Counter (PSMC) which is a nondestructive analysis (NDA) system is used for determination of plutonium amount in the impure scrap materials. However, long measurement time was needed to mitigate the plutonium measurement uncertainty. In 2002, the ENMC was installed in the plutonium fuel center plutonium fuel fabrication facility (PPFF) where MOX fuels for the ATR 'Fugen' and the FBR 'Joyo' were fabricated in the past. All fuel fabrication campaigns have been completed and treatment of the scrap materials generated from them has been done there.

2. Features of the ENMC

JAEA developed the Epithermal Neutron Multiplicity Counter (ENMC)(see FIG.1) under a joint study program with LANL in order to mitigate plutonium measurement uncertainty of the impure scrap materials.

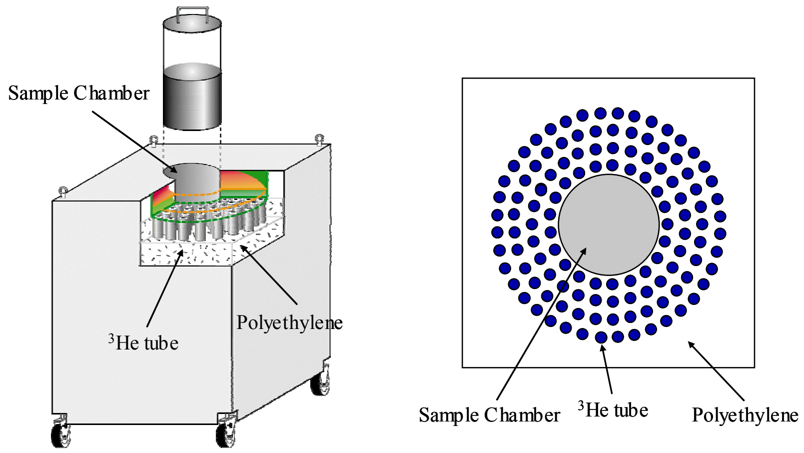


FIG. 1. Layout of the ENMC.

The ENMC can measure not only thermal neutrons which are measured by a conventional NDA system (eg. PSMC, INVS and HLNCC) but also epithermal neutrons. The PSMC for measuring plutonium amount measures the thermal neutrons by a coincidence and multiplicity technique. When it measures high impurity scrap materials, measurement uncertainty becomes large because of the increased number of (α,n) neutrons. This issue can be overcome by the short die-away time which will be attained by measurement of the epithermal neutrons.

The thickness of high-density polyethylene (HDPE) of the ENMC was reduced to allow detection of the epithermal neutrons. The cross section of the epithermal neutrons on ^{3}He is smaller than that of the thermal neutrons, so the detector efficiency is reduced. To overcome this issue, the number of ^{3}He tubes and the pressure of ^{3}He gas in them were increased in comparison with the PSMC (Table 1). These improvements give the ENMC a high efficiency of neutron measurement, 64%, and a short die-away time, 20μs [1]. The measurement of epithermal neutrons before complete moderation contributes to considerable mitigation of the measurement uncertainty.

Table 1. Comparison of ^{3}He tubes of ENMC and PSMC.

	PSMC	ENMC
Number of ^{3}He tubes	80	121
Pressure of ^{3}He in tubes	4 atm	10 atm

3. Initial calibration test of the ENMC

The initial calibration test of the ENMC was performed and measurement parameters, neutron detection efficiency, gate fraction, dead time correction, were determined. Furthermore, a comparison with PSMC measurement performance was carried out. The test results confirmed that the ENMC could measure impure scrap materials in about 1/40 – 1/10 of the measurement time with the same measurement uncertainty as the PSMC. The measurement uncertainty for the impure materials was 1/7

– 1/3 less for the ENMC than the PSMC for the same measurement time. Thus, the original purpose for developing the ENMC was attained.

4. Functional tests of the ENMC

JAEA confirmed the ENMC mitigated measurement uncertainty of impure MOX scrap materials in the initial calibration test. It was also confirmed that the ENMC had high measurement performance, and measurement uncertainty would be further lowered by optimization of measurement conditions.

JAEA carried out functional tests to identify how to utilize capabilities of the ENMC to their full potential. These functional tests were divided into two phases. In the phase I functional test, actual nuclear materials stored in the facility were measured to confirm measurement errors. In the phase II functional test, many kinds of samples were measured to find the optimum measurement conditions. The results of these functional tests are described below.

4.1 Phase I functional test

The purpose of the phase I functional test was to evaluate the measurement error of the ENMC. The systematic error components and values are shown in Table 2. The total systematic error of the ENMC was 1.3%. The statistical error depends on Pu mass, measurement time and α value. The statistical errors and measurement uncertainties of ENMC are shown in Table 3.

For the pure MOX sample (α=1), the statistical error was 0.1 – 0.3%, and the total measurement uncertainty was 1.4 % with 100 minutes measurement. The dominant error factor was the systematic error and the major component for it was sample positioning error. For the impure MOX sample (α=11), the major component for measurement error was statistical error, and it was 3%. The total measurement uncertainty was 3.3% [2].

Table 2. Systematic error components for ENMC.

Error component	%
Container effect	0.5
Positioning error of nuclear material (Variation of nuclear material distribution in container)	0.9
Positioning error of container	0.1
Calibration curve	0.8
Isotopic ratio (DA)	0.3
Cross talk (affection of background neutron)	~0
Cosmic ray effect	~0
Total systematic error	1.3

Table 3. Statistical errors and total error of ENMC.

α	Measurement time (min)	Pu Mass(g)	Systematic error (%)	Statistical error (%)	Measurement uncertainty (%)
1 (pure MOX)	10	1 - 10	1.3	1.1	1.8
		10 - 100		0.3	1.4
		100 - 1000		1.0	1.7
	100	1 - 10		0.4	1.4
		10 - 100		0.1	1.4
		100 - 1000		0.3	1.4
11 (impure MOX)	10	～500		10	11
	100			3	3.3

4.2 Phase II functional test

The purpose of the phase II functional test was to lower the total measurement uncertainty of the ENMC. In this test, the optimum measurement conditions were evaluated to minimize the systematic and the statistical errors. It was confirmed that the total measurement uncertainty of ENMC would be reduced to 0.5% in case of the pure sample (α=1) and it would be reduced to 0.7% in case of impure sample (α=11). Evaluation of each measurement error of ENMC is shown in Table 4.

Table 4. ENMC error estimation for storage items and small samples.

	Error components	Error for storage items	Error for small samples
Systematic Error	Container wall effect	0.5	0.0
	Positioning error of nuclear material (variation of nuclear material distribution in the container)	0.9	0.1
	Positioning error of Pu container	0.1	0.0
	Calibration curve	0.8	0.3
	Isotopics analysis error (DA)	0.3	0.3
	Total systematic error	1.3	0.5
Stastical Error	α≒1	0.3 (1.0kg Pu) (100min)	0.1 (20g Pu) (100min)
	α≒11	3 (0.5kg Pu) (100min)	0.4 (1g Pu) (400min)
Measurement Uncertainty	α≒1	1.4 (1.0kg Pu) (100min)	0.5 (20g Pu) (100min)
	α≒11	3.3 (0.5kg Pu) (100min)	0.7 (1g Pu) (100min)

The detailed evaluation results are described next.

a) Evaluation of container wall effect

The container wall effect was evaluated by placing a californium source in a plastic bottle, an SUS can and an aluminum can. These containers are used for the storage of nuclear materials and all have different wall thicknesses. The results of this experiment are shown in FIG. 2. The maximum difference of doubles rates was about 0.5%. However, this error could be neglected when using the same type container for calibration and measurement. By considering the cost and the easiness of disposal, the plastic bottle was selected for measuring small samples.

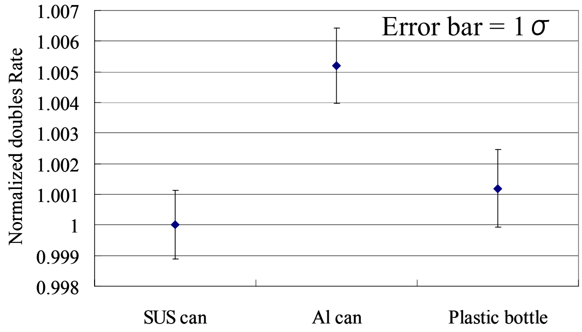


FIG. 2. Variation of doubles rates.

b) Evaluation of positioning error of nuclear material (variation of nuclear material distribution in the container)

The variation in sample location changes the neutron detection efficiency and is the main reason for sample positioning error. The ENMC radial and axial doubles profiles are shown in FIG. 3. The variation of the doubles profile was less than 0.9% for a practical sample container made of SUS (SUS can: height, 20cm; diameter, 12cm). To reduce this error, the nuclear material was placed at the most uniform region of detection efficiency. For a plastic vial (height, 5cm; diameter, 3cm), the error by nuclear material positioning was 0.1% because the exsiting range of nuclear material was limited.

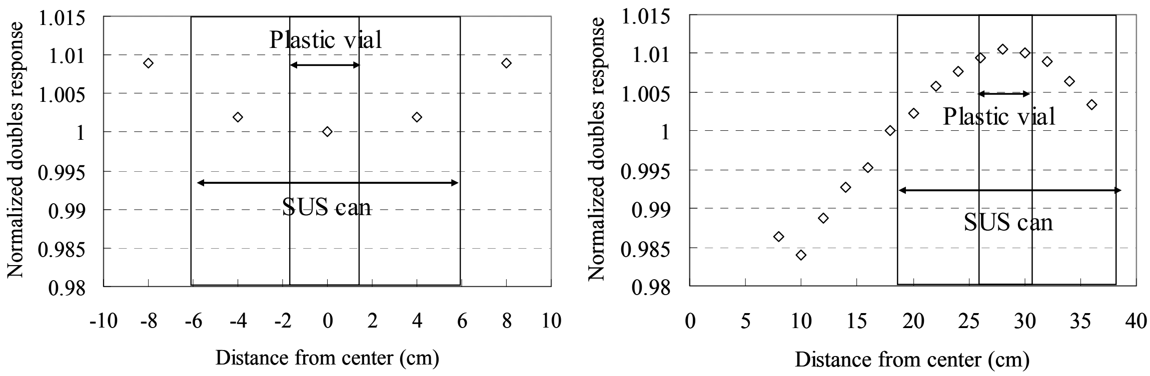


FIG. 3. Normalized doubles response profiles (Left: axial, Right: Radial).

c) Evaluation of positioning error of the sample container

The same sample was measured repeatedly to evaluate the positioning error of the plastic bottle in this experiment. A sample holder was made in order to keep the sample at the same position each time. The sample (20g Pu, α=1) was measured nine times for 100 min by three different operators. The results are shown in FIG. 4. The standard deviation of all measured results was 0.13%. This standard deviation was almost the same as the mean of the standard deviation of each measurement result. Therefore, the positioning error of the plastic bottle was included in the statistical error.

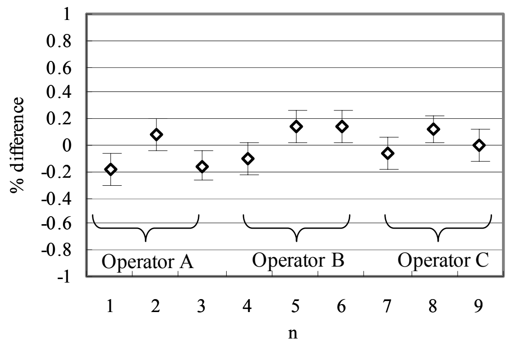
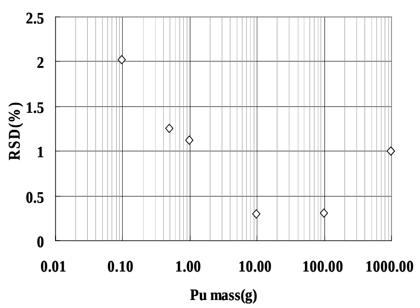


FIG. 4. The repeat measurement results for the same sample.

d) Evaluation of optimum plutonium mass to minimize the statistical error

The optimum plutonium mass to minimize the statistical error was evaluated. This experiment was done by using pure samples (α=1) and impure samples (α=11) with different plutonium masses. FIG. 5 shows the statistical error depended on plutonium mass of the pure samples. FIG. 6 shows the results of impure samples. The plutonium mass that made the counting statistics the smallest was 20g Pu for the pure sample, and 1g for the impure sample.



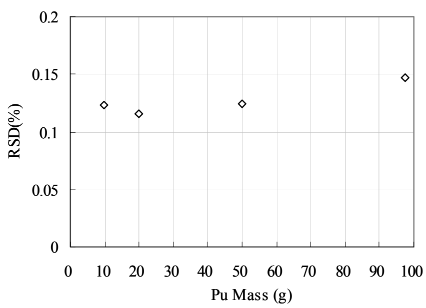


FIG. 5. Pu mass vs. RSD for pure MOX sample (left: Pu mass range 0.1g -1000g, α=1, meas. time = 10 min, right: Pu mass range 10g - 100g, α=1, meas. time = 100 min).

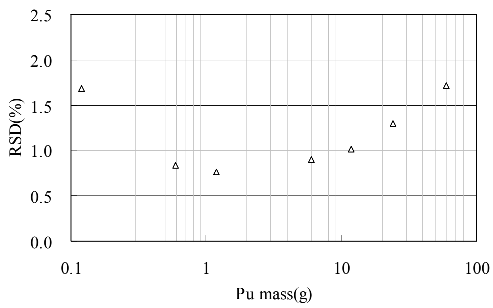


FIG. 6. Pu mass vs. RSD for impure MOX samples.

(Pu mass range: 0.1g – 100g, α=11, meas. time = 100min)

e) Evaluation of the optimum measurement time

FIG. 7 shows the statistical error depended on the measurement time. In this experiment, the pure sample (α=1, Pu mass=20g) and the impure sample (α=11, Pu mass = 1g) were measured with different measurement times. For the pure sample, the best statistical error was 0.12% in a 100 min measurement. For the impure sample, the best statistical error was 0.42% in a 379 min measurement (maximum counting time in a work day). For the pure sample, the statistical error in a 100 min measurement was sufficiently small compared to the systematic error of ENMC. On the other hand, for the impure sample, the optimum measurement time was longer than one day, but it was limited to one day from the viewpoint of safety.

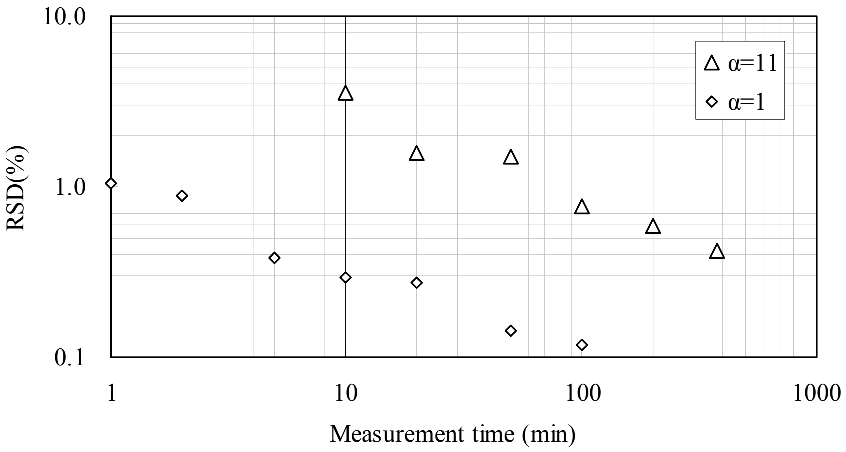


FIG. 7. The measurement time versus precision for the ENMC.

f) Evaluation of the calibration error

The calibration error of ENMC consists of Pu concentration error (by DA), container wall effect and counting statistics. Lessening of these errors was confirmed in the phase II functional test. The calibration error was reduced from 0.8% to 0.33% by optimization of measurement conditions (Table 5).

Table 5. Calibration error.

Calibration error	Large sample	Small sample
Pu concentration (DA)	0.3	0.3
Container wall effect	0.5	0
Counting statistics	0.5	0.1
Total	0.8	0.33

5. Applicability of the ENMC to material accountancy and safeguards

DA and NDA have advantages each other. DA is effective measurement method for the material accountancy and the quality control of the homogenized material such as the product material because of high measurement accuracy. On the other hand, NDA has also advantages for measuring of impure scrap material because of small sampling error. It also has advantages of not generating liquid waste for the NDA measurements and it does not need standard samples for each measurement.

By optimizing of measurement conditions, measurement uncertainty of the ENMC becomes just 0.5% - 0.7%. Furthermore, the ENMC can get measurement results within one day even for high Pu impurity scrap materials. Therefore, the ENMC can be an effective tool for a material accountancy system. In particular, for impure scrap materials generated from treatment processes and from dismantling of old equipment, the ENMC can provide better results than DA.

The ENMC would be applicable for safeguard. During inspections, some DA samples are taken for verification of bias defects. These samples are treated in the facility to allow their transfer to an IAEA analytical laboratory in Vienna. This activity requires an inspector work for a week to accomplish this. By application of the ENMC as a tool to verify bias defects, DA sample taking and DA treatment activities will be reduced. The need for international transportation of DA samples from Japan to Vienna can also be reduced.

6. Conclusion

The ENMC developed by JAEA and LANL lowered the measurement uncertainty of plutonium amount for analysis of impure scrap materials. Furthermore, JAEA confirmed following features of the ENMC.

- The ENMC can provide better measurement result for impure scrap materials than DA.
- The ENMC can get the measurement results in a shorter time than DA.

Furthermore, the ENMC has advantages which are low cost and no generation of the waste.

JAEA considers the ENMC can be used as a material accountancy tool by optimizing of measurement conditions. JAEA expects the ENMC will be used as the standard material accountancy tool for future MOX fuel fabrication plants.

The ENMC can also improve the efficiency of safeguard activity when applied in inspections because it eliminates the time delays of evaluation by DA and cuts the time spent in sampling activities. JAEA plans to demonstrate the effectiveness of the ENMC to the inspectorate in the near future, and will propose the ENMC be implemented as the inspection system.

ACKNOWLEDGEMENTS

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Assay of small fissile masses in waste by the active neutron correlation technique

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Abstract. The Joint Research Centre recently obtained a license to operate a new experimental device intended for research in the field of nuclear safeguards. One of the research projects currently being planned for the new device concerns the mass determination of small amounts of fissile materials by means of the active neutron correlation technique. The device incorporates a commercial pulsed neutron generator and a large graphite mantle surrounding the sample cavity. In this configuration a relatively high thermal neutron flux with a long lifetime is achieved inside the sample cavity. By pulsing the neutron generator, a sample may be interrogated by a pure thermal neutron flux during repeated time periods. The neutrons emitted from the fission events in the sample are partly detected in time intervals in a fashion similar to the standard passive neutron correlation technique (multiplicity counting). The frequency distributions of detected neutrons, or rather the moments hereof, can be expressed as function of the fissile mass, and sample and instrument parameters in analytical equations. The present paper describes the new experimental device, the data acquisition systems and the analysis method currently being investigated for the assay of small fissile quantities.

1. Introduction

Under the research programme in nuclear safeguards, the Joint Research Centre of the European Commission develops instrumentation and analysis methods for the non-destructive assay of nuclear materials. In relation to this, a new experimental device was designed and constructed in the laboratory of the Institute for the Protection and the Security of the Citizen (IPSC). Last year the laboratory received a full operational license from the national authorities for operating the new device including a pulsed (D-T) neutron generator, and sealed radioactive and fissile material sources. The device, called Pulsed Neutron Interrogation Test Assembly (PUNITA), is first of all intended for research in NDA methods for nuclear safeguards purposes but also in methods for detection of illicit trafficking of nuclear and non-nuclear materials. The nuclear safeguards application of PUNITA concerns the determination of the mass of fissile material in a sample independent of matrix materials and spatial source distribution. To achieve this aim, intense bursts of thermal neutrons, produced by the neutron generator and a surrounding graphite mantle, induce fission in the fissile isotopes of the sample. The fission neutrons are detected and analysed in the time domain. The analysis method consists of a further development of the standard passive neutron correlation technique. The method is particularly effective for small amounts of fissile material and is expected to be capable of assaying

235U and plutonium at unprecedented low quantities (< 1 mg). An obvious application is the assay of fissile material in waste as the method can assay small amounts even in the presence of matrix materials. Other applications could include assay of process samples in nuclear fuel cycle facilities.

2. Instrumentation

2.1. Description of the experimental set-up

The Pulsed Neutron Interrogation Test Assembly (PUNITA) is a versatile experimental tool. The instrument is designed to give the experimenter maximum flexibility with respect to detector and sample arrangements. When in the closed configuration the instrument forms a cube with a central void called the sample cavity (Figure 1). The top and bottom sides of the cube are part of a central structure, while the four vertical sides are located on movable trolleys. The trolleys move on a rail system to yield a perfect mating with the central structure. The size of the sample cavity is 50 cm by 50 cm cross-section and 80 cm height. The accelerator assembly of the (D-T) generator has a length of only 43 cm and can be placed without constraints anywhere inside the cavity. A graphite liner of 205 mm thickness is located on all six sides of the sample cavity. On the vertical sides, the graphite liner is integrated in the trolley to give access to the cavity from all sides when in the open configuration. A total of 1,350 kg of reactor grade graphite is used in the liner.

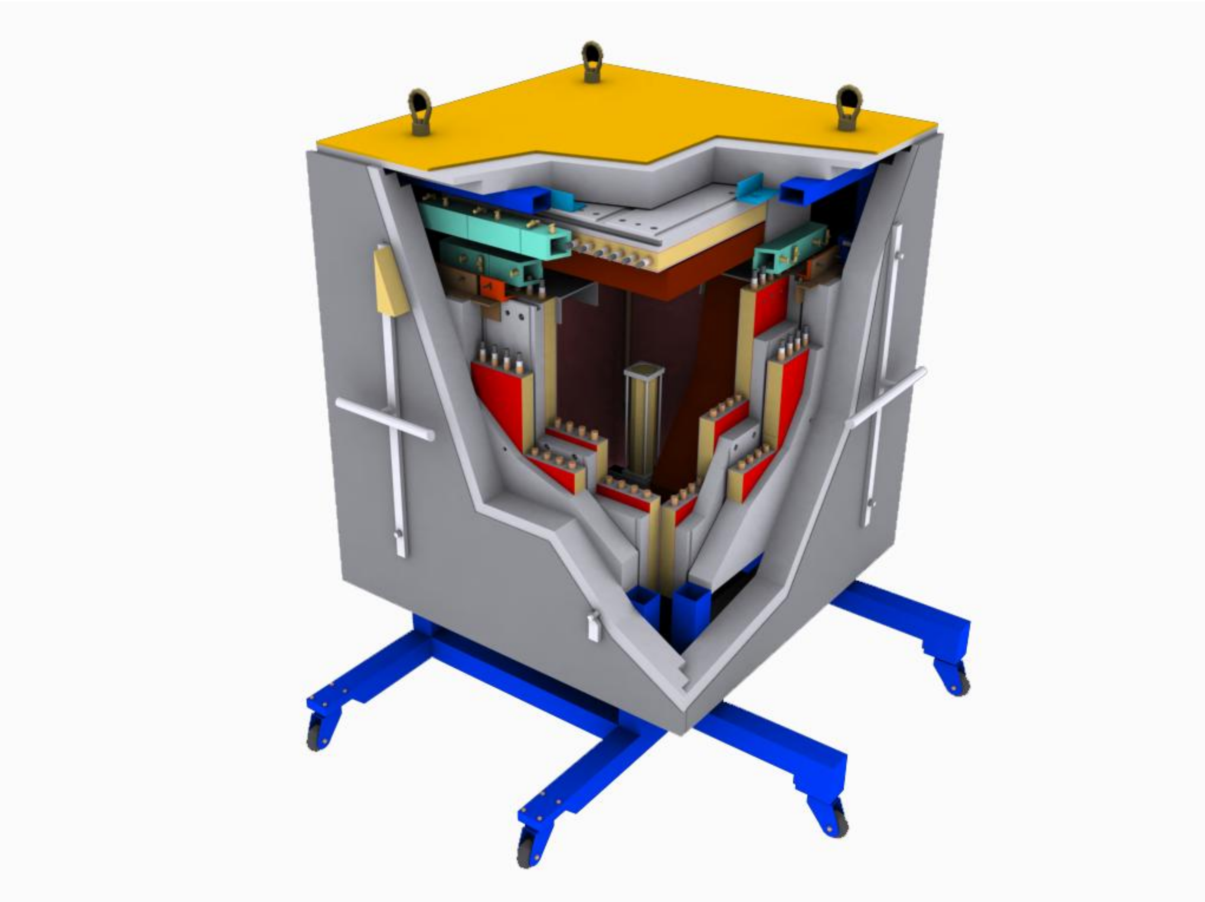

FIG. 1. Layout of PUNITA. The graphics shows the sample cavity with the accelerator assembly, the surrounding fission neutron detectors, and the source monitors embedded in the shield.

Fast neutron detector modules are integrated in each of the six sides immediately behind the graphite liner. These modules, nominated fission neutron counters, include 96 ^{3}He proportional detectors of 3040 torr and 1000 mm length. The ^{3}He detectors are embedded in a single row in a block of polyethylene with a 1 mm cadmium cladding on all surfaces. A neutron shield of 300 - 350 mm polyethylene is placed behind the fission neutron counters on all six sides. On each of the vertical sides eight ^{3}He neutron detector of 3040 torr and 500 mm length, nominated source monitors, are embedded in the polyethylene shield. Other permanently installed instrumentation include bare ^{3}He counters of pressure below 760 torr, nominated thermal flux monitors, located in the corners of the sample cavity.

The neutron generator, model A-211 from Thermo Electron Inc., is capable of emitting 14-MeV neutrons at a rate of $2{\cdot}10^8$ s^{-1}. The generator can be pulsed at rates from 10 Hz to 150 Hz. The duration of the 14-MeV neutron pulse is about 5 μs. An important feature of the generator is the pulsing of both the Penning ion source and the acceleration voltage assuring that no neutrons are emitted between pulses.

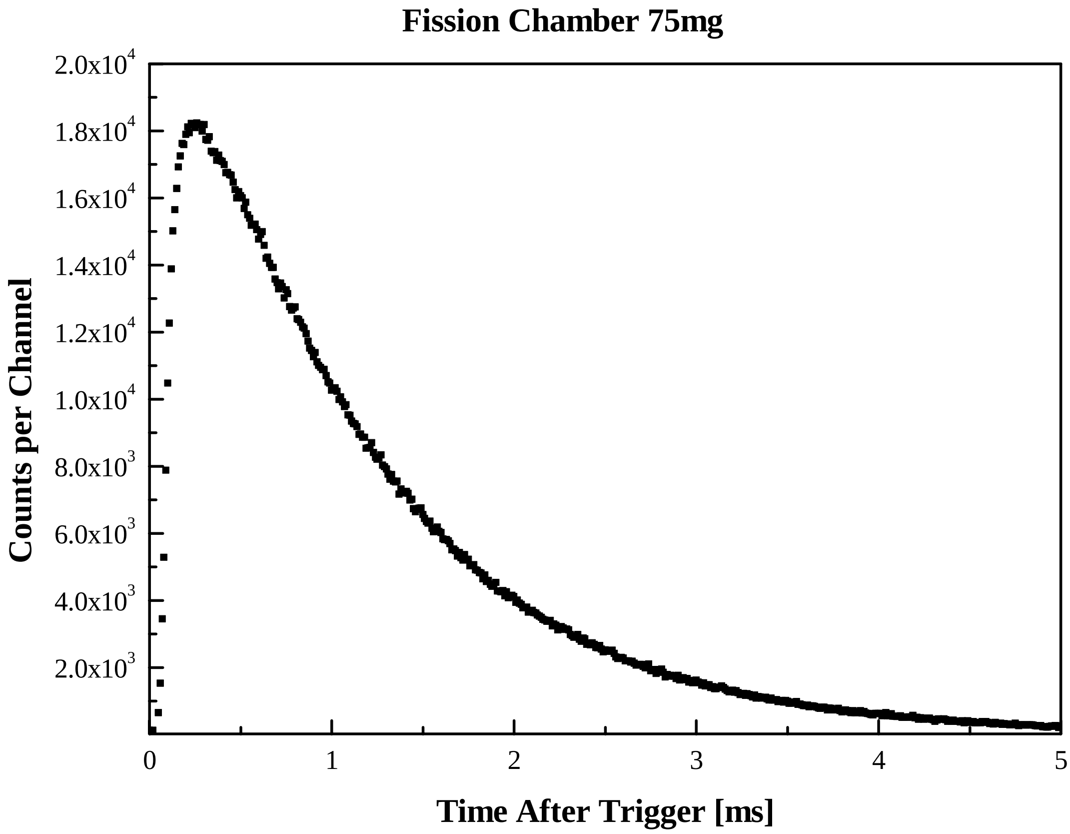

FIG. 2. Thermal neutron pulse as produced by the neutron generator and the graphite moderator. The maximum thermal flux is observed at 280 µs.

The 14-MeV neutrons emitted in each generator pulse are thermalized in the graphite liner in a period of about 300 μs following the pulse. After this time only thermal source neutrons persist in the cavity. After reaching a maximum value at about 280 μs after the fast neutron pulse, the thermal neutron flux in the sample cavity decays according to a single exponential function (Figure 2). In this time period a fissile sample in the cavity would undergo fission by thermal neutrons only. Also in this period, only the fast fission neutrons are detected in the cadmium covered fission neutron detectors. The source monitors located in the shielding are used for normalisation of the 14-MeV neutron emission from the generator. Likewise, the bare ^{3}He detectors in the sample cavity are used for normalisation of the interrogating thermal flux. The time response of the various neutron detectors mention here are recorded with Ortec MCS multi-channel scalers which are triggered synchronous to the pulsing of the neutron generator. The time of detection of the fission neutrons in the fission neutron counters is in addition recorded with the purpose-built Multi-Channel Frequency Analyser.

2.2. Description of the data acquisition system

All analysis instrumentation applied for the assay of fissile materials is based on the registration of the time of detection of neutrons. The analyzer modules and their interconnections are shown in Figure 3.

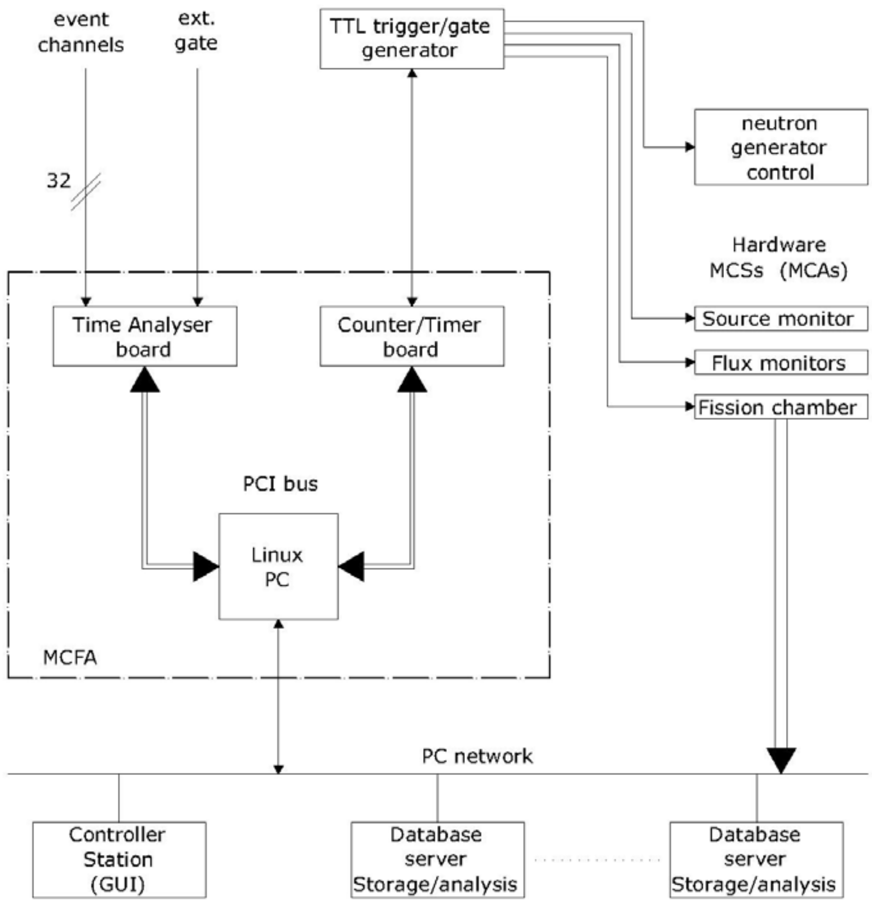

FIG. 3. Schematic of the data acquisition system on PUNITA for fissile material assay.

The MCFA (Multi-Channel Frequency Analyzer) is a purpose-built frequency analyzer (multiplicity counter) capable of handling the sub-millisecond timing requirements of the active neutron correlation technique. The MCFA has been integrated in a 2U, 19-inch rack, industrial PC. The MCFA includes two digital signal acquisition/generation boards on the PCI bus. One board is a standard 8 MHz counter/timer device. This unit generates the periodical trigger and gate signals required for the synchronization of the various data acquisition modules. The unit is operated remotely from the controller station. The other board is the Time Analyzer acquisition board. This unit has 32 individual TTL input lines which are connected to the fission neutron detectors. Whenever a neutron signal is registered on an input line, the line number and a time-tag is stored in the on-board memory. The time-tag is generated by a 50 MHz, 32 bit wide clock counter. Batches of data from the on-board memory are transmitted by DMA over the PCI bus to the computer RAM. The MCFA computer uses the Linux operating system, rather than Microsoft Windows, in order to maximize the time available for processing the signal pulse train in real time. In fact the computer runs without peripherals such as disk drives, keyboard, and video controller. All communications to and from the MCFA computer is carried out via LAN. The set-up parameters for both Time Analyzer and Counter/Timer boards are managed on the user interface on the Controller Station (networked PC) and transmitted to the MCFA computer when launching the measurement. During measurements useful data are displayed on the Controller Station monitor. These data include count rate, signal multiplet rates (e.g. Reals) and the measurement error of these quantities, as well as diagnostics and statistics on individual input lines. These data are derived from the on-line analysis of the signal pulse train. The raw data (the signal pulse train) are stored during the acquisition in a database which may be distributed on several networked PCs. At the end of the measurement, the measurement result in form of frequency distributions are also stored in the database. The analysis according to the interpretation model, as well as retrieval of old measurements, can be carried out on any of the networked PCs.

The instrumentation on PUNITA also includes a number of hardware Ortec multi-channel scaler (MCS). These instruments record the time histograms of various detectors relative to the firing of the neutron generator.

The synchronization of all acquisition modules (MCS and MCFA) with respect to the pulsing of the neutron generator is controlled by a Berkley Nucleonics Corporation TTL pulse generator. This pulse generator has eight TTL outputs with may be delayed and shaped individually with respect to the base frequency. The periodic triggering of the pulse generator can be set by front panel or by external trigger.

3. Principles of active neutron correlation analysis

The fast neutrons emitted in each generator pulse are thermalized in the graphite liner resulting in a thermal neutron flux which interrogates the unknown sample in the measurement cavity [1, 2]. The thermal neutron flux decreases according to an exponential function with a thermal neutron lifetime, $1/\Lambda$, of 1.05 milliseconds [3].

The thermal neutron lifetime in the sample is a function of the thickness of the graphite liner and the composition of the sample.

The volume averaged thermal neutron flux $\Phi(\xi)$ in the sample produces $F_S(\xi)d\xi$ fission events in the time interval $[\xi, \xi + d\xi]$ during the pulse period T_P ($0 \le \xi \le T_P$) after each source pulse. It is:

$$F_S(\xi)d\xi = \Sigma_f V\Phi(\xi)d\xi = \frac{A}{M_I}\sigma_f G_I \Phi(\xi)d\xi \qquad (1)$$

$\Sigma_f \equiv$	Macroscopic thermal fission cross-section
$V \equiv$	Volume of sample
$A \equiv$	Avogadro number
$M_I \equiv$	Molecular weight of fissile material
$\sigma_f \equiv$	Microscopic thermal fission cross-section
$G_I \equiv$	Fissile mass

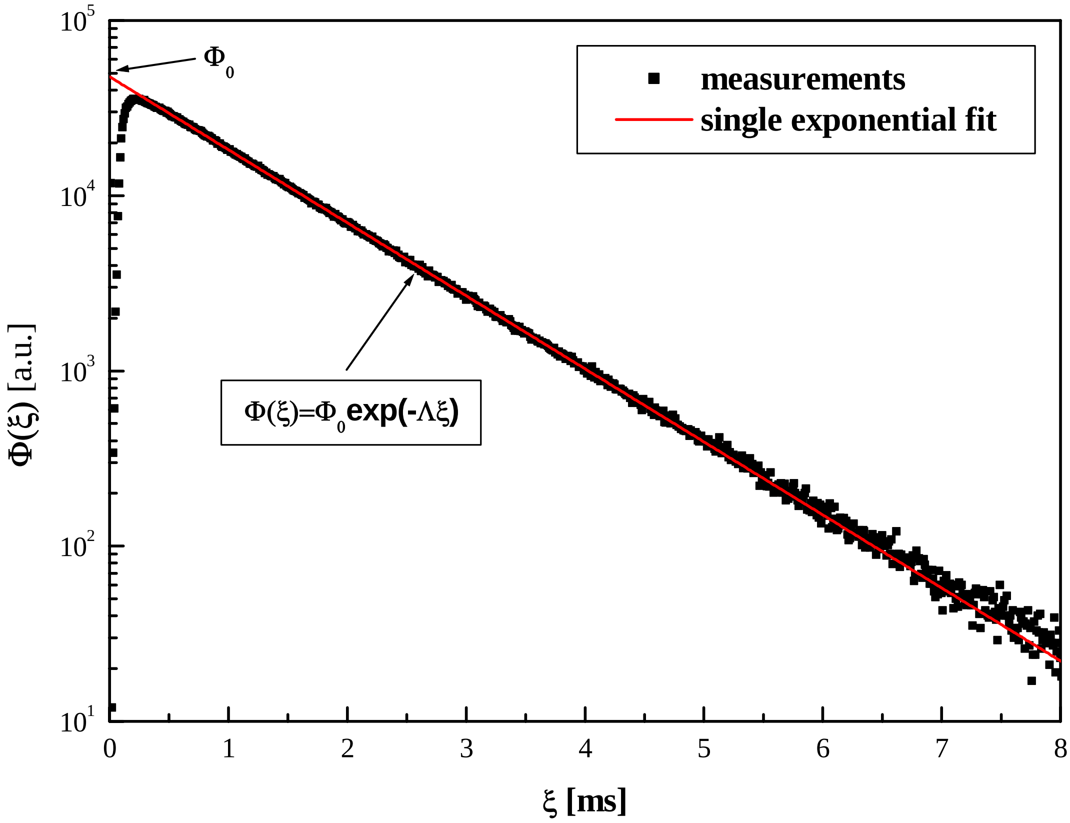

FIG. 4. The actual thermal neutron flux distribution as function of time coordinate ξ between two successive neutron generator pulses.

The induced fission events produce on the average $\nu_{I(1)}$ fast single neutrons, $\nu_{I(2)}$ fast neutron pairs, and $\nu_{I(\mu)}$ fast neutron multiplets of order μ per induced fission event. It is:

$$\nu_{I(\mu)} = \sum_{\mu=\nu}^{\infty} \binom{\nu}{\mu} P_{I,\nu} \qquad (2)$$

$P_{I,\nu} \equiv$ Probability for emission of ν fast neutrons per induced fission event [4, 5]

Some of the fast neutron multiplets of order μ escape the test sample and the surrounding graphite liner and reach the fission neutron detection assembly. Some of the fast neutrons of these multiplets are slowed down in the cadmium covered polyethylene moderator, reach thermal energy, and are detected in the incorporated ^{3}He thermal neutron detectors with the probability ε^{μ}, where μ = 1, 2, 3, … The fission neutron detection assembly, composed of a polyethylene moderator, ^{3}He detectors and an external cadmium cladding, is dimensioned such, that its generated thermal neutron population decreases according to an exponential function with the reciprocal of the decay constant 1/λ $(35\ \mu s \leq 1/\lambda \leq 60\ \mu s)$. By this means the thermal neutron lifetime in the fission neutron detector, 1/λ, is much smaller than the thermal neutron lifetime in the sample, 1/Λ.

In the case of very small fissile masses distributed in the sample $(G_I \leq 1\,g)$ the neutron multiplication and self-shielding can be neglected. It is then convenient [2, 6] to record the frequency distribution $b_x^*(i),\quad x = 0,1,2,\ldots X_{Max}$ after each neutron pulse in consecutive observation intervals:

$$\left[T_0, T_0 + \tau\right],\ \ \left[T_0 + \tau, T_0 + 2\tau\right],\ \ldots\left[T_0 + (i-1)\tau, T_0 + i\tau\right],\ i = 1,2,3\ldots I$$

The frequency is obtained from the number of events with x counts in the i-th observation interval and reads:

$$b_x^*(i) = B_x(i) / \sum_{x=0}^{X_{Max}} B_x(i) \tag{3}$$

$B_x(i) \equiv$ Total number of events with x signals observed in the interval $\left[T_0 + (i-1)\tau,\ T_0 + i\tau\right]$ accumulated from all neutron pulses during the measurement time T_M.

$T_0 \equiv$ Delay time after each generator pulse.

Each registration of the frequency distributions is started with a delay T_0 after each source pulse. This delay of $500\ \mu s \langle\ T_0\ \langle 1000\ \mu s$ is necessary to allow for:

(1) The complete decrease of the neutron population in the fission neutron counter assembly produced by the pulsed neutron source immediately after each source pulse.
(2) The slowing down to thermal energies of the fast neutrons present in the graphite liner and the matrix of the sample, which were emitted by the last source pulse.
(3) The slowing down to thermal energies of the fast neutrons, which were emitted by fission events in the test sample, having escaped the graphite liner and arrived in the fission neutron counter.

The probability $b_x(i)$, the theoretical counterpart to the frequency $b_x^*(i)$, depends on the fissile mass G_I, the thermal neutron flux $\Phi(\xi)$, the neutron detection probability ε of the fission neutron counter assembly, the two neutron lifetimes $1/\lambda$, $1/\Lambda$ and some other known instrumental and nuclear data. For the numerical data evaluation it is assumed that $b_x(i) \approx b_x^*(i)$. Then the simple mathematical expressions obtained with the factorial moments of $b_x(i)$ can be used for the interpretation of the measured frequencies. It is:

$$m_{(\mu)}(i) = \sum_{x=\mu}^{x_{Max}} \binom{x}{\mu} b_x(i) \tag{4}$$

A determination of both the spontaneous fissile mass and the fissile mass [6] requires 3 steps to obtain the frequency distribution of the thermal neutrons in the fission neutron detector assembly. These measurements concern:

(1) The frequency distribution of the background neutrons without test sample in the cavity of the pulsed neutron facility.
(2) The frequency distribution of both the background and spontaneous fission neutrons with the test sample in the facility.
(3) The frequency distribution of the background and the spontaneous and induced fission neutrons with the sample in the facility and the pulsed source operating.

This procedure has the disadvantage, that the frequency distributions of the background is measured in a separate run. This distribution depends strongly on the variation of cosmic radiation with time and becomes more important with elements in the matrix of the test sample, having a high mass number. This shifts the detectable fissile mass limit $G_{I,Min}$ to higher values. A determination of very small fissile masses (less than 10 mg) in the sample should be performed in one step. Then the spontaneous fissile mass G_S must be determined with the passive neutron correlation technique [7-10]. A one step fissile mass determination is possible with the formulas derived in [6]. It is:

$$f_1\left\{m_{(1)}(i)\right\} = m_{(1)}(i) = A_1 + B_1 e^{-\Lambda i \tau} \tag{5}$$

$$f_2\left\{m_{(1)}(i), m_{(2)}(i)\right\} = A_2 + B_2 e^{-\Lambda i \tau} \tag{6}$$

or in general

$$f_\mu\left\{m_{(1)}(i), m_{(2)}(i) \ldots m_{(\mu)}(i)\right\} = A_\mu + B_\mu e^{-\Lambda i \tau} \quad i = 1,2,3\ldots \tag{7}$$

with

$$B_\mu = \Phi(T_0) G_I \varepsilon^\mu \frac{A}{M_I} \sigma_f \nu_{I,(\mu)} \tau \varpi_\mu(\lambda, \Lambda, \tau) \tag{8}$$

and

$$\varpi_1(\lambda, \Lambda, \tau) = \frac{1}{\Lambda\tau} \frac{\lambda}{\lambda - \Lambda} e^{\Lambda\tau}(1 - e^{-\Lambda\tau}) \tag{9}$$

$$\varpi_2(\lambda, \Lambda, \tau) = \frac{\lambda}{\lambda - \Lambda} e^{\Lambda\tau} \left\{ \frac{1}{\Lambda\tau}(1 - e^{-\Lambda\tau}) - \frac{1}{\lambda\tau}(1 - e^{-\lambda\tau}) \right\} \tag{10}$$

$$\Phi(T_0) = \Phi_0 e^{-\Lambda T_0} \tag{11}$$

The function $f_\mu \left\{ m_{(1)}(i), m_{(2)}(i) \ldots m_{(\mu)}(i) \right\}$ is composed of a sum of products of factorial moments of the probability distribution of order 1 to μ. It is proportional to a background term A_μ and a term B_μ, weighted with an exponential function having the same argument $-\Lambda i \tau$ for $\mu = 1, 2, \ldots$, according to the Eqs. (5-7). The background A_μ is composed of correlated neutron signals of order μ from background due to cosmic radiation, nearby sources and spontaneous fission neutron emitters present in the sample. The parameters A_μ, B_μ and Λ are determined by least square fits on the function f_μ. The amplitude B_μ of the exponential function is proportional to the thermal neutron flux $\Phi(T_0)$ at time $\xi = T_0$ (Figure 4), to the fissile mass G_I, to the $\mu - th$ power of the neutron detection probability ε of the fission neutron detector assembly and to known parameters of the sample and the pulsed neutron interrogation facility. The value of the thermal neutron flux $\Phi(T_0)$ is eliminated by measuring the test sample relative to a standard with a known mass $G_{I,R}$. It is:

$$\frac{B_\mu}{B_{\mu,R}} = \frac{\Phi(T_0) G_I \varepsilon^\mu \dfrac{1}{M_I} \sigma_f \nu_{I,(\mu)} \varpi_\mu(\lambda, \Lambda, \tau)}{\Phi_R(T_0) G_{I,R} \varepsilon_R^\mu \dfrac{1}{M_{I,R}} \sigma_{f,R} \nu_{I,R,(\mu)} \varpi_\mu(\lambda, \Lambda_R, \tau)} \tag{12}$$

The flux ratio $\Phi(T_0)/\Phi_R(T_0)$ is replaced by the counting rate ratio $C_0/C_{0,R}$ of the source monitors measured during the two runs, the first with the test , the second with the reference sample R. Eq. (12) is further simplified if both the test and the reference sample have the same neutron physical data. It is:

$$\frac{B_\mu}{B_{\mu R}} = \frac{C_0 G_I \varepsilon^\mu \varpi_\mu(\lambda,\Lambda,\tau)}{C_{0,R} G_{I,R} \varepsilon_R^\mu \varpi_\mu(\lambda,\Lambda_R,\tau)} \tag{13}$$

The thermal neutron lifetime $1/\lambda$ in the fission neutron detector assembly can be made rather insensitive to the matrix of the test sample. This is not the case for the thermal neutron lifetime $1/\Lambda$. This neutron lifetime can be different for the test and the reference sample.

The fissile mass ratio $\frac{G_I}{G_{I,R}}$ is found, by forming the ratio $\left[\left[B_2/B_{2,R}\right]/\left[B_1/B_{1,R}\right]^2\right]$. It is:

$$\frac{G_I}{G_{I,R}} = \frac{C_{0,R}}{C_0}\frac{B_{2,R}}{B_2}\left[\frac{B_1}{B_{1,R}}\right]^2 \frac{\varpi_2(\lambda,\Lambda,\tau)}{\varpi_2(\lambda,\Lambda_R,\tau)}\left[\frac{\varpi_1(\lambda,\Lambda_R,\tau)}{\varpi_1(\lambda,\Lambda,\tau)}\right]^2 \tag{14}$$

The ratio of the neutron detection probabilities $\varepsilon/\varepsilon_R$ follows from the ratio $\left[\left[B_2/B_{2,R}\right]/\left[B_1/B_{1,R}\right]\right]$. It is:

$$\frac{\varepsilon}{\varepsilon_R} = \frac{B_{1,R}}{B_1}\frac{B_2}{B_{2,R}}\frac{\varpi_1(\lambda,\Lambda,\tau)}{\varpi_1(\lambda,\Lambda_R,\tau)}\frac{\varpi_2(\lambda,\Lambda_R,\tau)}{\varpi_2(\lambda,\Lambda,\tau)} \tag{15}$$

The application of this measurement and interpretation procedure has the advantage, that the measurement time for a fissile mass assay is considerable shortened, the thermal neutron lifetime in the test sample is measured more accurately and the background data are obtained contemporarily with the test sample data.

ACKNOWLEDGEMENTS

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Analysis of time correlation measurements with the Active Well Coincidence Counter

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Abstract. We present the use of the Monte Carlo N-Particle–Politechnic of Milan (MCNP-PoliMi) code for the study of active well coincidence counters. The code simulates the operation of the shift register and can be used to perform time interval analysis. The code is described, and preliminary results for uranium metal samples of varying mass are discussed.

INTRODUCTION

Active well coincidence counters (AWCCs) are widely used for nondestructive assay applications in nuclear safeguards and nuclear waste characterization. The method is based on the detection of correlated neutrons from fission by He-3 detectors embedded in a polyethylene moderator. In the assay of uranium, an active measurement must be performed to induce fission in the material, and typically an Am/Li neutron source is used as the active source.

Monte Carlo studies of the measurement setup are useful in the design, optimization, and analysis of the entire measurement system. The simulation must take into account many factors, for example the Am/Li neutron spectrum, the multiplicity of neutron emissions in induced fission events, and the detection of thermalized neutrons by the He-3 counters. In this study, we address these issues and present a detailed analysis of the measurement system that includes parameters such as the length of fission chains generated in the fissile material by the source neutrons, the time of neutron detection in the He-3 counters, and the generation number of the detected neutrons. The simulations are performed with the Monte Carlo N-Particle–Politecnico de Milan (MCNP-PoliMi) code [1].

The simulation results are compared with measurements performed on uranium oxide standards with an AWCC that is in use at the Y-12 National Security Complex. Figure 1 shows the geometry of the MCNP-PoliMi simulation for the AWCC.

In addition to the simulation of traditional multiplicity parameters given by the shift register (singles, doubles, and triples), MCNP-PoliMi allows the user to simulate the entire distribution of time correlations between detectors and detector autocorrelations. We show that this approach, also known as time interval analysis and first proposed by Bruggeman and colleagues in 1996 [2], has the potential to lead to a more robust and complete analysis compared to the measurement of multiplicity alone.

MONTE CARLO OUTPUT

The MCNP-PoliMi code output records all capture events occurring in the He-3 detectors, together with the time at which these capture events occur. To obtain this data, it is sufficient to specify the cell numbers of the He-3 detectors in the MCNP-PoliMi input. The output data are then post-processed with a specifically developed post-processing code.

An excerpt of the MCNP-PoliMi data output file is shown in Table 1. Each line in the output file corresponds to a capture event in the He-3 detectors. The time (following a source event) at which the capture occurs is given in the column labeled “Time” in the table. The generation number of the neutron (number of fissions in the chain that produced the neutron) is given in the column labeled “Generation number” in the table. The column labeled “Number of scatterings” gives the total number of scatterings in the neutron’s history preceding its capture. The column labeled “Energy” gives the energy of the neutron before its capture by the He-3 detector. The data output of the code is post-processed using a specifically developed code, which is described in the following section.

Table 1. Excerpt from MCNP-PoliMi data output file.

History number	Particle number	Projectile type*	Interaction type^Δ	Target nucleus•	Cell number of collision event	Time (shakes)	Generation number	Number of scatterings	Energy (MeV)
1	6	1	0	2003	1	32401.6	3	221	8.15E-09
3	1	1	0	2003	1	18052.97	0	156	7.61E-08
7	1	1	0	2003	1	11250.02	0	107	3.34E-08
9	4	1	0	2003	1	1894.416	1	17	5.72E-06
10	1	1	0	2003	1	5845.552	0	62	9.25E-08
16	3	1	0	2003	1	4388.813	1	73	4.58E-08
18	7	1	0	2003	1	19023.05	2	153	8.27E-08
18	4	1	0	2003	1	8990.743	2	81	4.27E-08
32	1	1	0	2003	1	7732.952	0	67	1.31E-08
35	1	1	0	2003	1	6319.3	0	49	1.53E-08
...									

* 1 = neutron
Δ 0 = absorption
• 2003=helium-3

MONTE CARLO POST-PROCESSING

The post-processing code simulates the operation of the shift register typically used with AWCCs. In this instrument, the pulse train obtained by the detection events is processed to obtain a distribution of multiplets. The operation of the shift register is shown schematically in Figure 1, where a pulse train is shown as a function of time. The first pulse in the train (shown in red) opens the “time window,” but only after a “pre-delay” time during which the counter is dead. After the time window is opened, the number of pulses occurring within the time frame is recorded. In Figure 1, the four pulses occurring within the time window would be recorded by the shift register. This procedure is repeated for each pulse in the pulse train and a histogram of multiplicities is acquired. Typical values for the pre-delay and time window are 4 and 64 μsec, respectively.

It should be noted that the simulation does not take into account accidentals because each history is initiated by a single source event and evolves independently of any other history until all the particles pertaining to it are tracked.

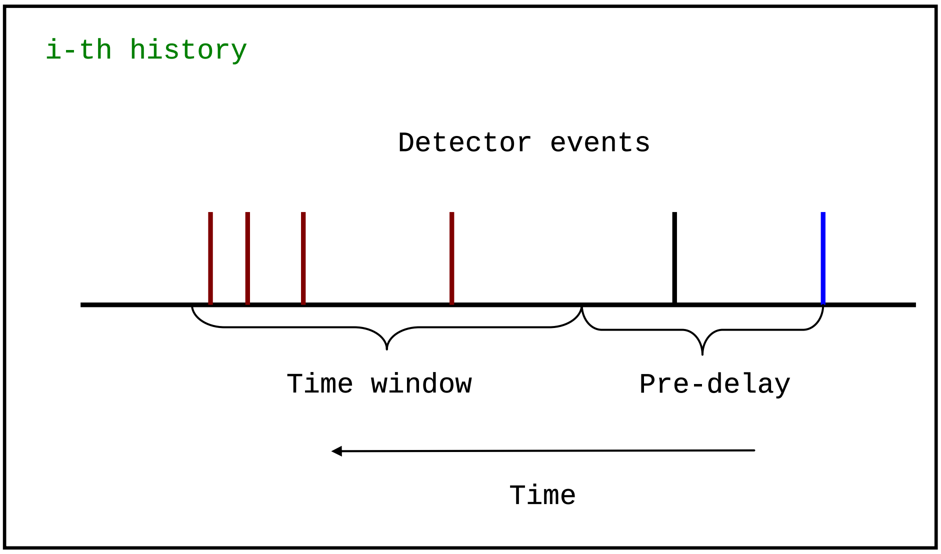

Figure 1. Illustration of pulse train for multiplet simulation.

ACTIVE WELL COINCIDENCE COUNTER SIMULATIONS: GEOMETRY

Simulations were performed for an AWCC that is in use at the Y-12 National Security Complex and was manufactured by JOMAR (model JOMAR 51). It consists of 42 He-3 tubes arranged in two concentric rings embedded in a polyethylene moderator. The radius of the inner ring is 30.8 cm and that of the outer is 38.1 cm. The sample cavity has a diameter of 22.9 cm. The tubes are operated at a pressure of 4 atm and their active length is 55.8 cm. Two Am/Li neutron sources are placed above and below the sample cavity. Figure 2 shows two views of the geometry of the MCNP simulation.

The first case modeled consisted of neutrons from the Am/Li sources and no sample in the AWCC. Four cases were then modeled for cylindrical uranium metal samples having radii 5.08 cm and varying heights. The sample height and mass are given in Table 2. All samples had composition 92 wt% U-235 and 8 wt% U-238. In all cases 10e6 neutron histories from the Am/Li sources were run.

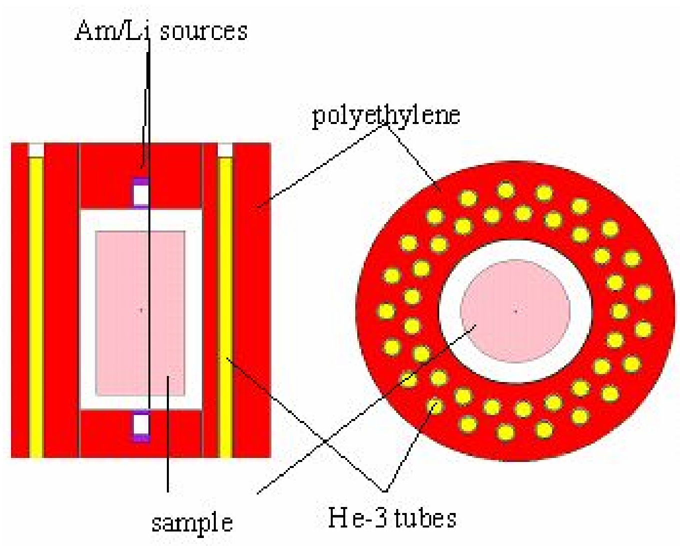

Figure 2. Geometry of Monte Carlo N-Particle simulation.

Table 2. Sample characteristics.

Name	Mass (g)	Height (cm)
u0	0	0
u1	3.06457E+03	2
u2	4.59685E+03	3
u3	6.12913E+03	4
u4	7.66142E+03	5

ACTIVE WELL COINCIDENCE COUNTER SIMULATIONS: RESULTS AND DISCUSSION

The resulting multiplicity distributions for the four cases are shown in Figure 3. As can be seen, the tail of the multiplicity distributions increases with sample mass. This is expected because of fission chain multiplication effects in the fissile material. In the case with no uranium sample, u0, the multiplicity is always zero, because only one neutron at a time is emitted by the Am/Li source. Figure 4 shows the generation numbers of the neutrons captured by the He-3 detectors. The figure shows that most of the neutrons are generation zero neutrons (i.e., neutrons from the Am/Li source). In the case without uranium sample, all neutrons detected are generation zero neutrons. The figure also shows that as the mass of the fissile material increases, the generation numbers also increase.

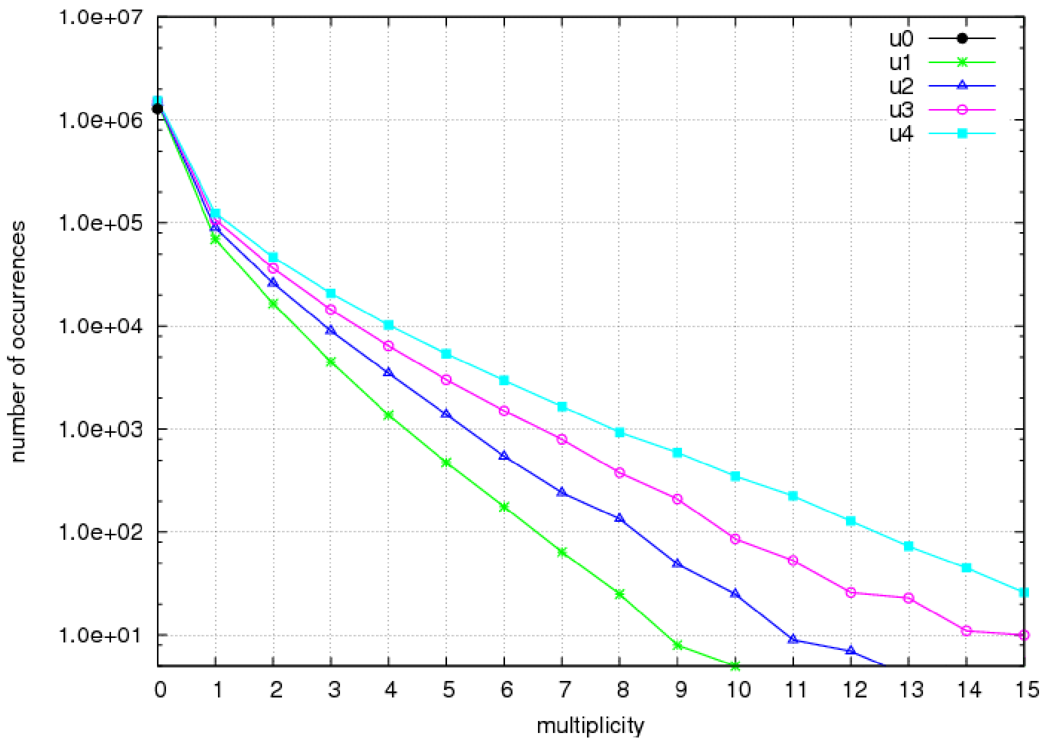

Figure 3. Shift register multiplicity for four sample masses.

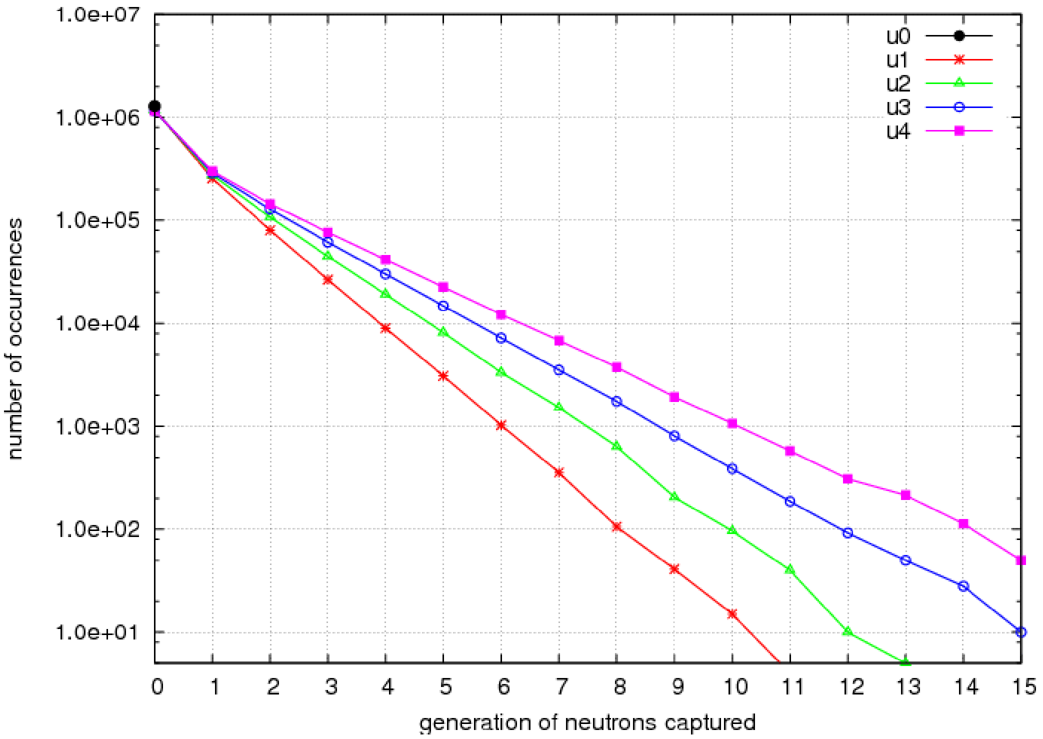

Figure 4. Generations of neutrons captured in the He-3 detectors.

Figure 5 shows the number of scatterings that the neutrons undergo before being captured in the He-3 detectors. The curves for the four samples have not been normalized. The shape is similar for all cases, with a peak at approximately 18 scattering events [3].

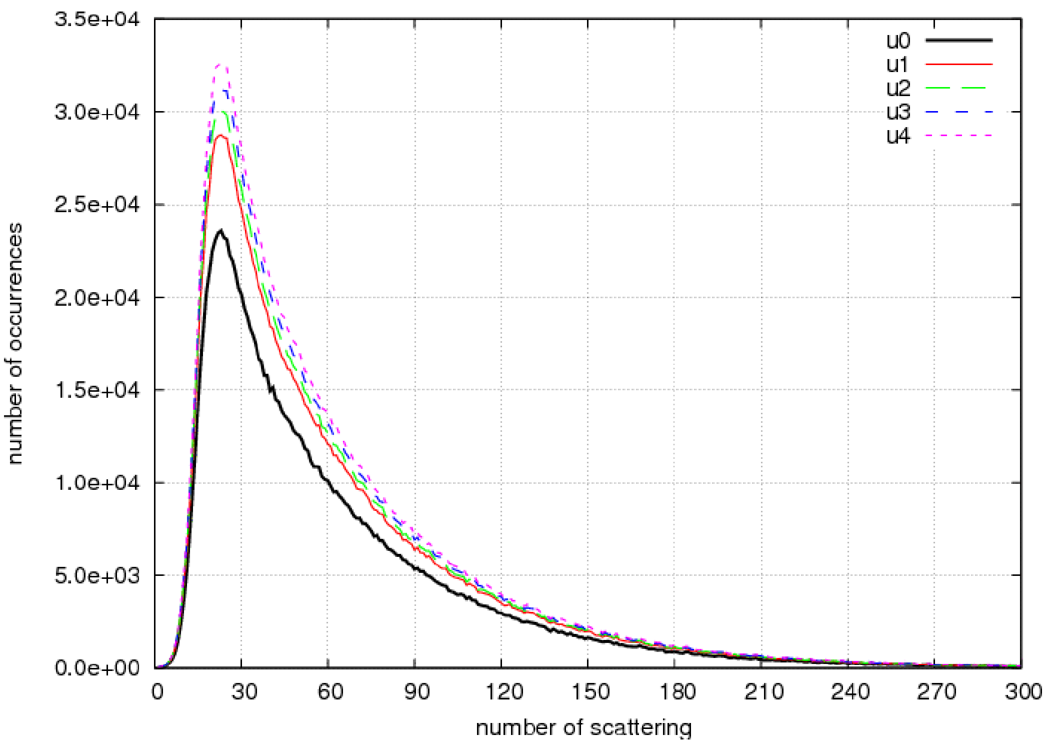

Figure 5. Number of neutron scatterings before capture in the He-3 detectors.

Figure 6 shows the neutron energy distributions before capture. Again, these distributions are similar for all four fissile samples, showing that the neutron slowing down process does not depend on the sample mass.

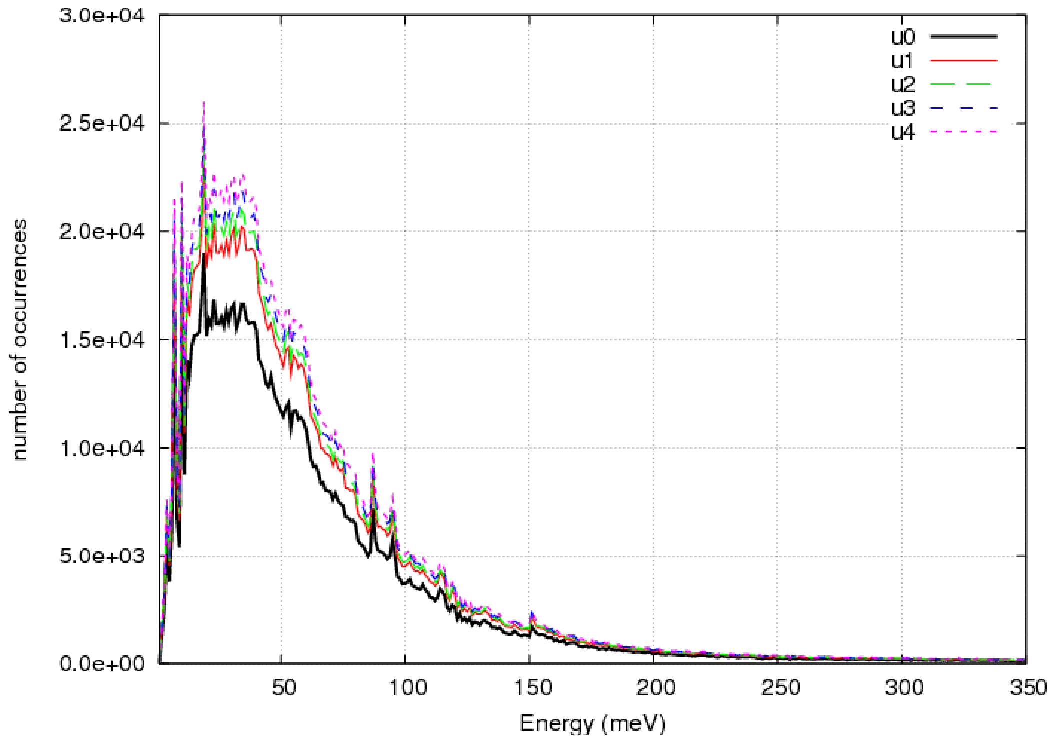

Figure 6. Energy distribution of neutrons before capture in the He-3 detectors.

Figure 7 shows the time interval analysis for triplets of neutrons. The number of triplets is shown as a function of the time interval between the first and second pulses (t_1) and the time interval between the second and third pulses (t_2).

CONCLUSIONS

This paper discussed the use of the MCNP-PoliMi code and a new post-processing code to simulate the measurements performed with an active well coincidence counter. The results presented show how the information given in the data output of the code can be used to gain an improved understanding of the physical processes that occur in the active well. In particular, the results show that it is possible to simulate the operation of the shift register, to evaluate the energy degradation of the neutrons by analyzing the number of scatterings and their energy before detection, and to determine the efficiency of the counter, both to source neutrons and to induced fission neutrons.

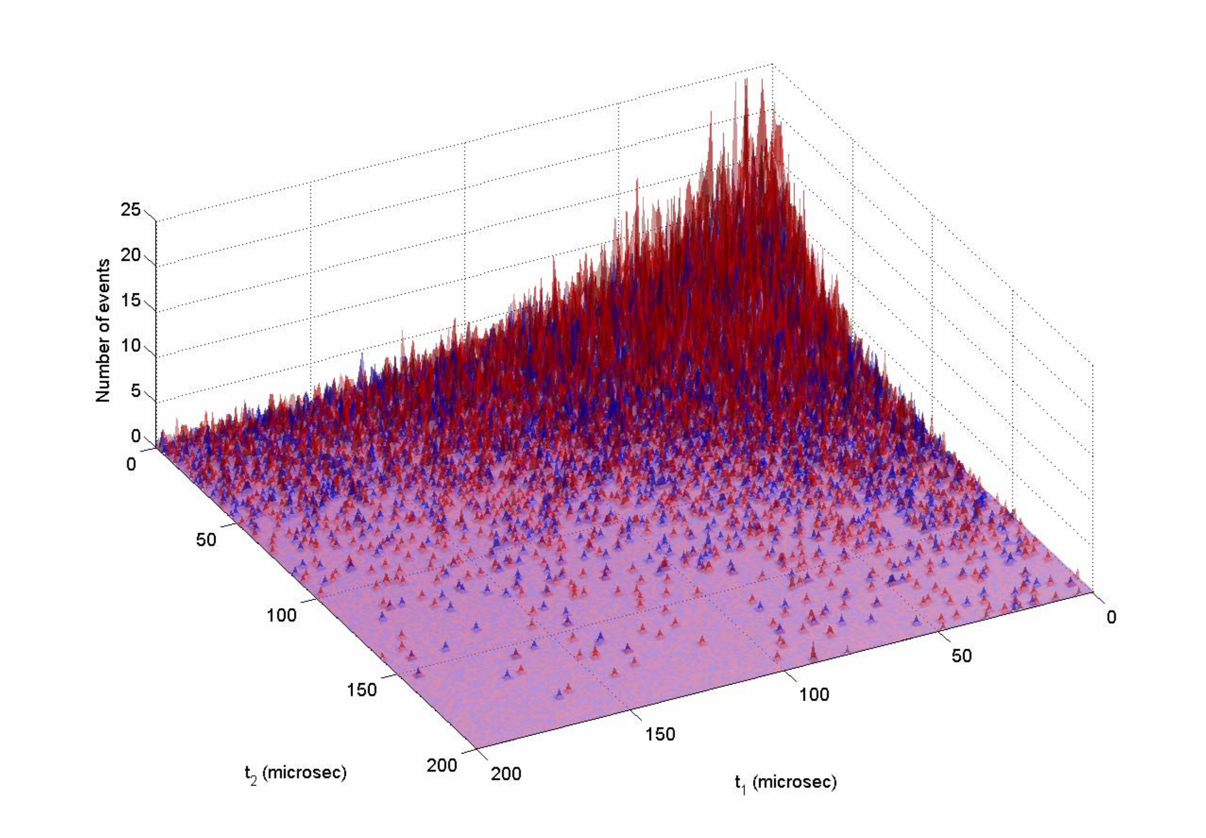

Figure 7. Time distribution of pairs of neutrons captured in the He-3 detectors.

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Monte Carlo analysis of the statistics of neutron detection by organic scintillators

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Abstract. We describe an event-by-event analysis of neutron detection in organic scintillators (plastics and liquids). Monte Carlo simulations and analytical descriptions are discussed. The analysis of the statistics of neutron collisions is important to understand the mechanism of neutron detection, and to perform subsequent unfolding procedures aimed at determining the incident neutron spectrum.

Introduction

Several measurement systems used in the area of nuclear nonproliferation employ liquid and plastic scintillators to detect neutrons and gamma rays from assemblies of fissile materials. The analysis of the neutron response functions of these detectors is essential for the development of unfolding techniques that can be used in neutron spectroscopy and neutron source identification methods [1]. The neutron response in these types of detectors is dominated by scattering on hydrogen, which is the mechanism that generates most of the scintillation light. Neutron scattering on carbon, the other main constituent of the scintillator, produces significantly less light. However, scattering on carbon affects the detector response because the neutron can lose energy in the scattering collision and, therefore, deposit less energy in subsequent interactions with hydrogen nuclei.

In this paper we present a detector response investigation that is based on an event-by-event statistical analysis of neutron histories in the detector. Quantities such as the number of collisions occurring in a scintillator are not measurable: what is measured is the combined effect of all collisions that contribute to a detector pulse. However, the *statistics* of the number of collisions is important when we consider neutron spectrum unfolding procedures. In fact, two effects, namely (i) nonlinearity in the light production from hydrogen scattering and (ii) small light output from carbon scattering, make the light output dependent not only on the *total* neutron energy deposited in the detector, but also on the *history* of the neutron energy deposition.

This paper describes the Monte Carlo simulation of neutron interactions with the detector material for varying incident neutron energies and varying detector sizes. The simulations are performed with the code MCNP-PoliMi, which allows event-resolved predictions of the interactions of neutrons with the detector material. A subsequent post-processing of the simulation results allows one to determine the number of elastic collisions that the neutrons undergo with hydrogen and carbon, together with the amount of energy that is deposited as a function of the number of collisions. The light output generated by the hydrogen and carbon recoils is also modeled, and the total detector efficiency is determined as a function of the incident neutron energy.

Monte Carlo Simulations

The simulations were performed using the code MCNP-PoliMi [2]. The detector modeled consists of a cube having the dimensions 10 × 10 × 10 cm and a H:C composition in the ratio 0.548:0.452. This composition corresponds to that of the liquid scintillator BC-501A manufactured by Saint-Gobain.

Separate simulations were performed for each incident neutron energy interval, in the range 0 to 5 MeV in 0.1 MeV intervals, with neutron energies distributed uniformly in the 0.1 MeV intervals. The neutrons were incident perpendicularly on one of the faces of the detector cube, and uniformly distributed within the face.

The Monte Carlo output files record every neutron interaction that occurs in the detector volume, including interaction type, energy deposited, and collision nucleus. The energy deposited is then converted into light output, taking into account the identity of the collision nucleus (hydrogen or carbon). In the case of hydrogen, the relationship between energy deposited (T in MeV) and light output (L in MeVee) is

$$L = aT^2 + bT \equiv L(T) \tag{1}$$

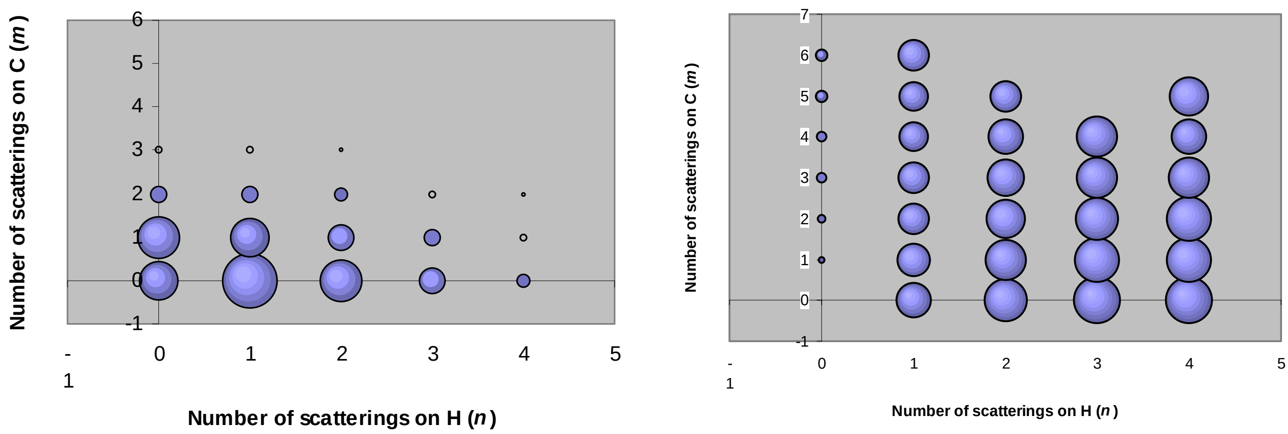
with a = 0.035 MeVee/MeV^2 and b = 0.141 MeVee/MeV for liquid scintillators. In the case of scattering on carbon, the light output is very small. For this study we used the following relationship:

$$L(T) = cT \ , \tag{2}$$

with c=0.02 MeVee/MeV.

Monte Carlo Results: Probability Distributions and Average Pulse Heights

Figure 1a shows the relative probability that neutron histories will include *n* scatterings on hydrogen and *m* scatterings on carbon for incident neutrons having energy uniformly distributed in the interval 1 to 1.1 MeV, on a detector having dimensions 10 × 10 × 10 cm. In the figure, the radius of the sphere is proportional to this relative probability. Figure 1a shows that the most probable event at this neutron energy and detector size is single scattering on hydrogen. Histories with only two scatterings on hydrogen (n = 2, m = 0) and only one scattering on carbon (n = 0 and m = 1) are approximately equally probable. Slightly less probable are histories with no interactions (n = 0 and m = 0) and histories with one scattering on carbon and one on hydrogen (n = 1 and m = 1).



(a) (b)

FIG. 1a. Relative probability of neutron histories as a function of the number of scatterings on hydrogen (n) and on carbon (m) (1- to 1.1-MeV incident neutrons). FIG. 1b. Average pulse heights as a function of the number of scatterings on hydrogen (n) and on carbon (m) (1- to 1.1-MeV incident neutrons).

Figure 1b shows the average pulse height generated by neutron histories with *n* scatterings on hydrogen and *m* scatterings on carbon. In the figure, the radius of the sphere is proportional to an average pulse height for *n*, *m* scatterings on hydrogen and carbon, respectively. As expected, the histories consisting solely of scatterings on carbon do not generate much light in the detector (n = 0).

Single scatterings on hydrogen (n = 1, m = 0) generate less light than multiple scatterings on hydrogen (n = 2 to 4, m = 0).

Monte Carlo Results: Pulse Height Distributions

The Monte Carlo analysis allows us to calculate not only the average pulse height generated in the detector as a function of the *n* and *m* scatterings on hydrogen and carbon, but also the entire pulse height distribution. One such distribution is shown in Fig. 2 for 1-MeV incident neutrons. Figure 2a shows the total pulse height distribution, and Fig. 2b shows the contributions to the total distribution from histories comprising *n* and *m* scatterings on hydrogen and carbon, respectively, with n = 1, 2, and 3 and m = 0 and 1. Fig. 2b shows that multiple scatterings on hydrogen contribute to increasing the detector response at higher pulse heights, whereas scatterings on carbon contribute to an increase in the detector response at lower pulse heights. The distribution labeled “Others” represents the cumulative contribution of all other combinations of *n* and *m* scatterings.

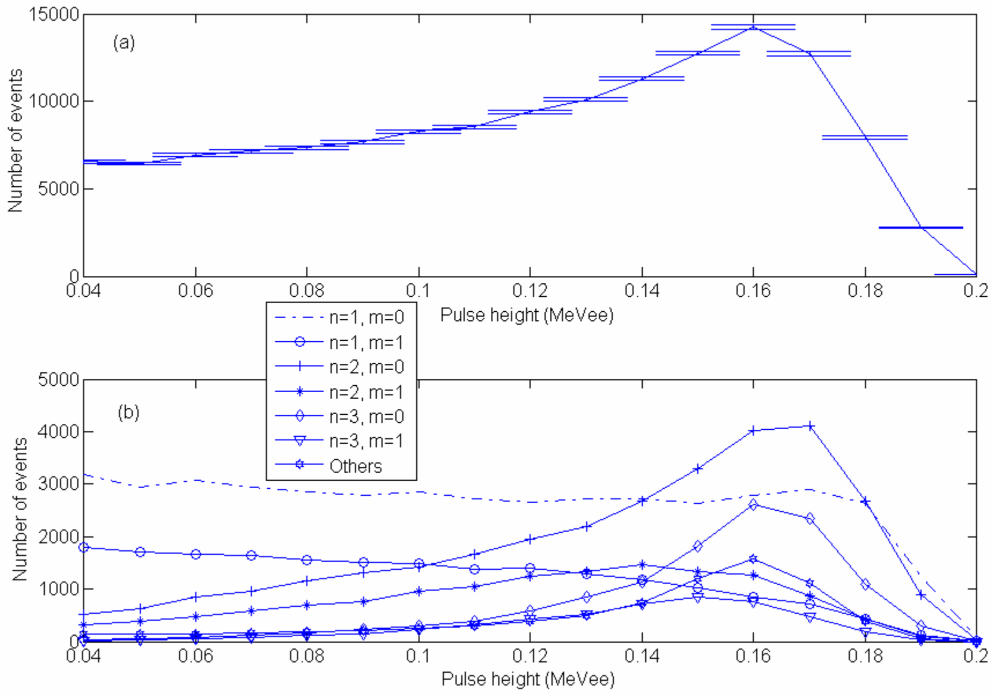


FIG. 2. Pulse height distributions for 1- to 1.1-MeV incident neutrons. (a) Total pulse height distribution is shown with error bars that represent statistical error. (b) Total is subdivided according to the total number of scatterings on hydrogen (n = 1, 2, and 3) and carbon (m = 0 and 1) in the neutron history. Error bars are not shown for clarity.

Detector Size

Figure 3 shows the pulse height distribution for 1- to 1.1-MeV incident neutrons generated in two different detector sizes: a 10 × 10 × 10-cm detector (as in Fig. 2) and a 20 × 20 × 20-cm detector. In the larger detector the peak at high pulse heights is more pronounced because of the increased probability of multiple scatterings of neutrons in collisions with hydrogen nuclei.

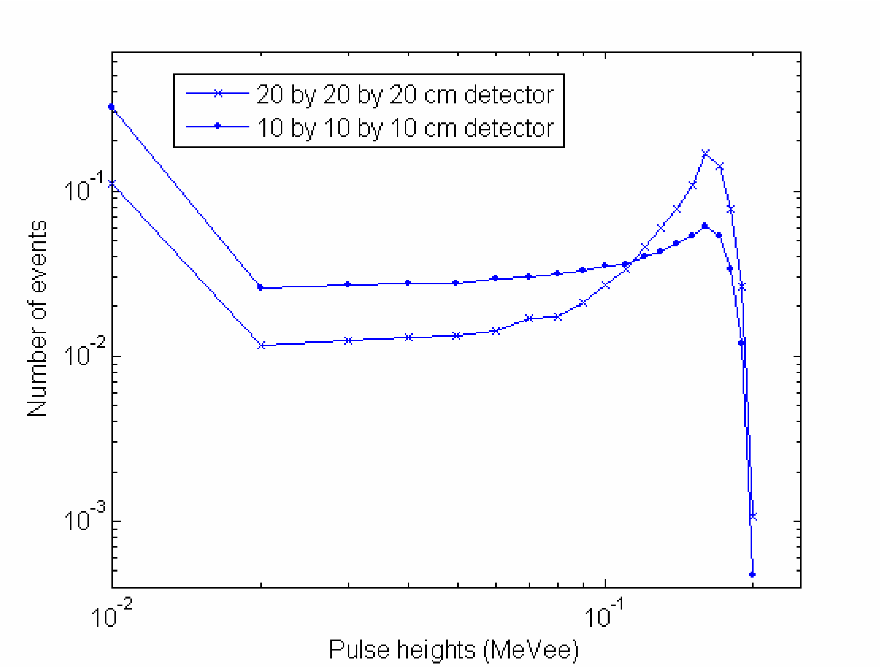


FIG. 3. Pulse height distributions for 1- to 1.1-MeV incident neutrons and two detector sizes.

Continuous Spectrum: Cf-252

In the case of a continuous neutron energy spectrum, e.g. Cf-252, the resulting pulse height distribution is the sum of many distributions, such as those shown in Fig. 2b. As a result, the cumulative distribution will typically include a broad continuum that extends up to the maximum light output generated by the neutrons having the highest energy, and an accumulation of many pulses at low pulse heights. In an experiment, the detector threshold setting in the data acquisition system will serve to eliminate the contributions at the low pulse heights.

Figure 4a shows the pulse height distribution for Cf-252 neutrons impinging on a liquid scintillator. Note that the scale in the figure is now log-log. Figure 4b shows the contributions to the total distribution from histories comprising *n* and *m* scatterings on hydrogen and carbon, respectively, with *n* = 0 to 3 and *m* = 0 to 3. Note that the histories comprising only scatterings on carbon (*n* = 0) contribute mainly to the low pulse heights.

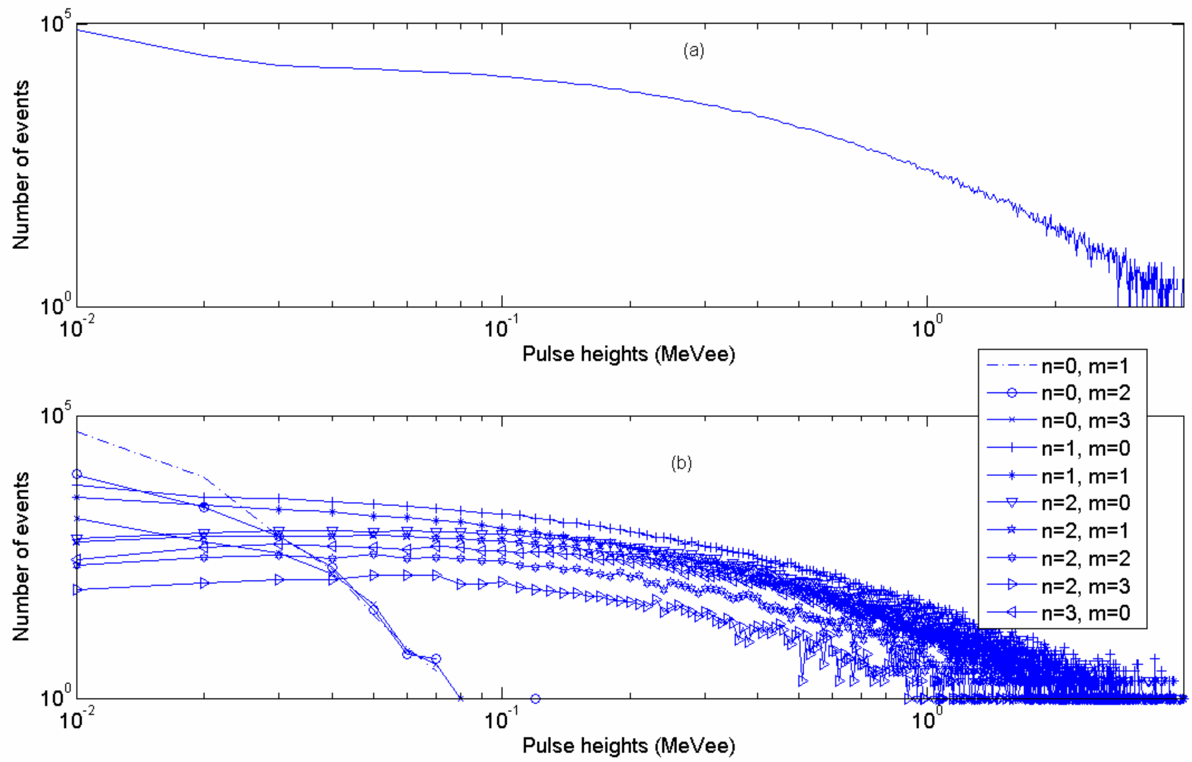


FIG. 4. Pulse height distributions for Cf-252 neutrons. (a) Total pulse height distribution. (b) The total is subdivided according to the total number of scatterings on hydrogen (n = 0, 1, 2, and 3) and carbon (m = 0, 1, 2, and 3) in the neutron history. Error bars are not shown for clarity.

Figure 5 shows a comparison between experimental data [3] and simulated data. The experimental data were acquired with the detector threshold set at approximately 0.12 MeVee. As can be seen, there is a very good agreement between the simulated and measured data: the average relative error was approximately 6% in the pulse height range 0.12 to 1.12 MeVee. The comparison shown in Fig. 5 serves as a validation for the Monte Carlo approach.

In addition to providing an estimate of the total pulse height distribution, Monte Carlo can be used to simulate the individual components, such as those shown in Fig. 4b. This ability is essential in the analysis of the response of existing detectors and in the development of new detector types.

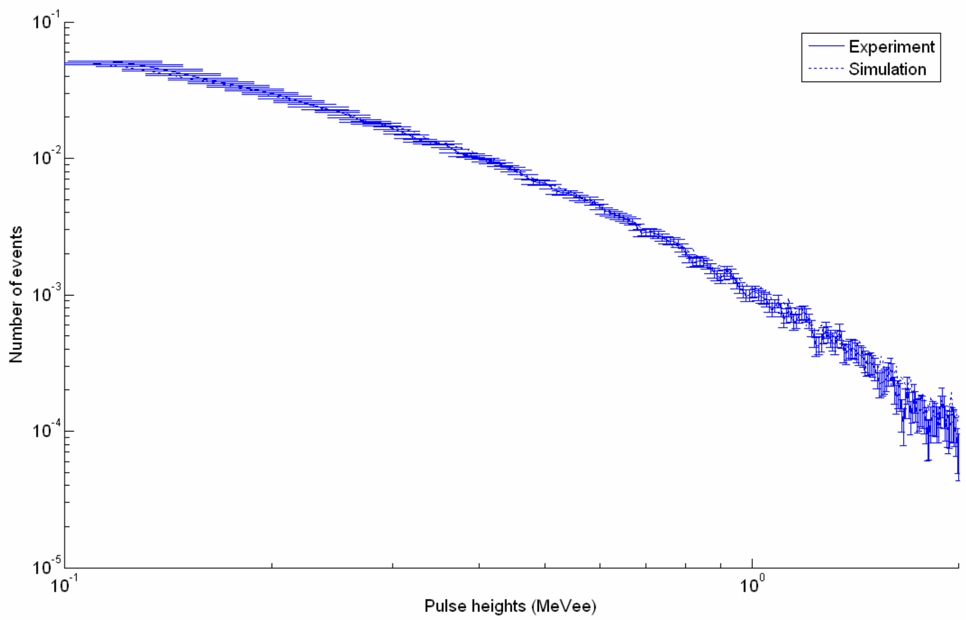


FIG. 5. Pulse height distributions for Cf-252 neutrons. Experimental data [3] shown with error bars, simulation shown with dotted line.

Analytical Approach and Discussion

The distribution of the energy transferred (*T*) by a neutron having incoming energy E_0, in a single collision with hydrogen is equal to

$$p(T,E_0)dT = \frac{dT}{E_0}, \tag{3}$$

where, as expected, $p(T,E_0)$ is independent of the amount of energy transferred, T. It follows that the distribution of light generated in the detector by a neutron having incident energy E_0 is

$$f_1(L,E_0) = \frac{1}{E_0\sqrt{b^2+4aL}}, \tag{4}$$

with L lying between 0 and $L_{\max} = aE_0^2 + bE_0$. For a given light intensity L in (4), the transferred energy is given as

$$T(L) = \frac{\sqrt{b^2+4aL}-b}{2a}. \tag{5}$$

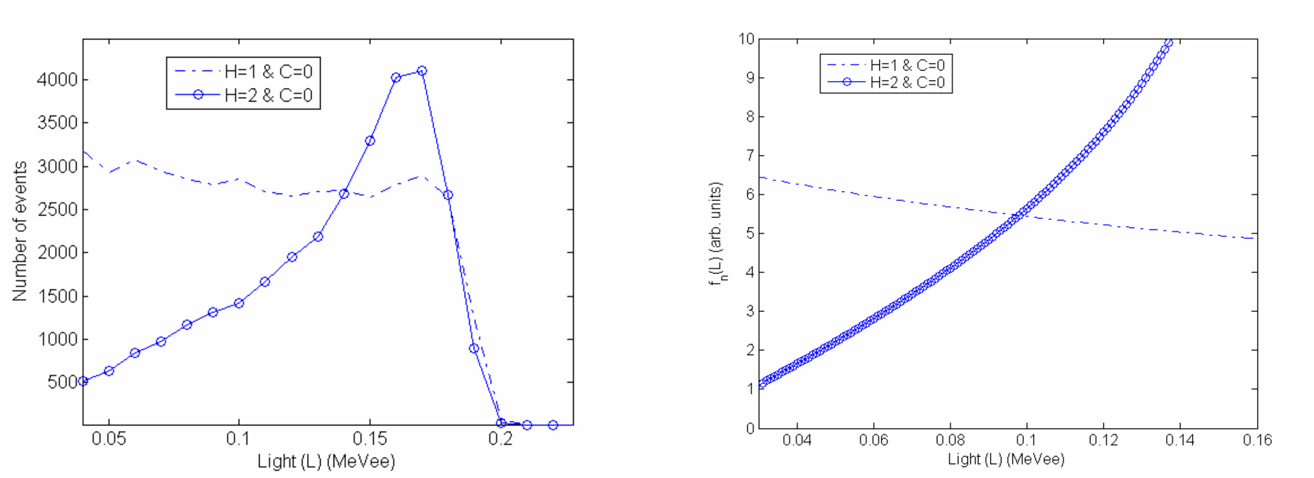
That is, f_1 is relatively flat since *a*<<*b*, and it decreases slowly for increasing light intensities, as shown in Figure 6 (b).

From here the distribution of the light generated by neutrons colliding more than one time with hydrogen can be calculated by convolution-type integrals. For the distribution after two collisions, $f_2(L,E_0)$, one has:

$$f_2(L,E_0)=\int_0^L f_1(L-l,E_0-T(l))\,f_1(l,E_0)\,dl = \\ =\frac{1}{E_0}\int_0^L \frac{1}{\left(E_0-T(l)\right)\sqrt{b^2+4a(L-l)}}\frac{1}{\sqrt{b^2+4al}}dl, \tag{6}$$

where $T(l)$ is given by Eq. (5).

The integral in Eq. (6) can be calculated numerically and is shown in Figure 6b. As can be seen, there is a general agreement in the shape of f_1 and f_2 between the Monte Carlo simulations (Fig. 6a) and the analytical solution (Fig. 6b).



(a) (b)

FIG. 6a. Monte Carlo simulation of light output for 1-MeV incident neutrons in the case of one scattering on hydrogen and two scatterings on hydrogen. FIG. 6b. Analytical solution of the light output for the case of one scattering on hydrogen and two scatterings on hydrogen (1-MeV incident neutrons).

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Short Notice Random Inspection (SNRI) regime at a low enriched uranium fuel fabrication plant in Spain

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Abstract. The IAEA safeguards criteria require that, in addition to an annual verification of the facility operator's physical inventory taking, a regime of randomised inspections be implemented at fuel fabrication plants, in order to achieve 100% verification coverage for domestic transfers of nuclear material. The advance notification period for such randomised inspections depends on the time that such material is available for verification, commonly referred to as 'residence time'. In the case of this facility, the residence time is almost zero; therefore the short notice random inspection (SNRI) regime being developed is based on a very short inspection notification time that would match the very short residence time, in order to achieve the safeguards goals. Another challenge addressed here is related to the demand of some States to have their representative attend all inspections with the IAEA. The paper addresses the implementation issues and activities to be performed by the IAEA, together with a regional inspection authority.

1. Introduction

The IAEA implements safeguards at several low enriched uranium (LEU) fuel fabrication plants (FFPs) in different areas of the world. The throughput of these plants varies from 100 tons to 500 tons of uranium per year. The main products of the plants are fuel assemblies for light water power reactors (LWRs). Subsidiary products could be fuel rods and pellets. The feed to these LEU plants is either uranium dioxide (UO_2) powder or uranium hexafluoride (UF_6). which requires a related conversion plant, normally on the site of the LEU fabrication plant.

Fuel assemblies are used in the cores of LWRs to produce heat energy which is transformed into electrical energy. Plutonium is produced in irradiated LEU fuel assemblies, increasing the proliferation sensitivity of the fuel. However, recent experience shows that the LEU material itself is a potential proliferation risk if further enriched to weapons grade. This is one of the main reasons for the IAEA's decision to strengthen the safeguards measures at such facilities. Two major obligatory requirements have recently been introduced by the IAEA: the implementation of short notice random inspections (SNRIs), and the use of advanced non-destrictive assay (NDA) equipment - specifically, the uranium neutron coincidence collar to verify fresh fuel assemblies. A short implementation period has been granted up to 2006, to permit adequate time to reach agreement on a procedure with the State/regional authority and operator. In October 2005, the IAEA, the European Commission (EC), the Spanish State Authority and the operator of the LEU fuel fabrication plant at Juzbado, Spain had reached the first agreement for implementation of the SNRI at the Juzbado plant. The agreement is based on performing SNRIs during a two-day period, as triggered either by the IAEA or the EC. It was agreed that the IAEA inspectors would be at the facility alone for the first day, performing a certain number of activities, while the EC inspectors would join the IAEA team the next day for additional activities, and vice versa.

Further details of implementation were under final discussion. Following the IAEA Technical Review Committee (TRC) review, the IAEA Deputy Director General of Safeguards approved this SNRI procedure for a trial period of one year. It was planned to start the implementation of this agreement as of 1 January 2006. Unfortunately, the agreement was not implemented for several reasons which are beyond the scope of discussion in this paper.

Two circumstances had to be addressed in the development of the safeguards procedures for this plant. Firstly, the operator routinely packs the finished fuel assemblies produced into closed containers ready for shipment, without unnecessary delay. This results in the absence of residence time (normally a few days) that would allow verification by the IAEA. Secondly, there is a requirement of the EC and the Spanish Authority that the EC participate in all IAEA inspections, including SNRIs. Therefore the procedure, which is discussed in this paper, takes these two requirements into consideration and is able to satisfy the safeguards requirements of all the parties involved. Note that there is no conversion plant at Juzbado and the only feed is the UO_2 powder received from abroad.

2. Short Notice Random Inspections at Juzbado

2.1. Definition

Short notice random inspections (SNRIs) are defined as routine inspections performed at random intervals with a short advance notification to the State/regional authority and the operator.

2.2. Objectives

The main goal of a SNRI is to fully cover the verification of nuclear material involved in the transfers (receipts and shipments) of a facility. A second goal is the verification of nuclear material at the strategic points (e.g. pellets sampling from process lines). The verification of the nuclear material in transfer and at the strategic points would strongly contribute in the evaluation of the material unaccounted for (MUF) and of the shipper/receiver difference (SRD) of the plant. Another important goal is the confirmation of the declared operation of the facility. In addition, SNRIs would confirm the absence of borrowing of fuel assemblies, particularly between the LEU fuel fabrication plant and the domestic LWRs.

2.3. SNRI elements

2.3.1. Declaration

At the beginning of each year the facility operator will provide a declaration of the planned powder receipts and fuel assembly production per week for the entire year. At the end of each month, the IAEA will receive (via encrypted e-mail) an updating of this declaration Any change to the declaration should be notified as soon as it is known to the operator. The declaration should include, at least, the number of items (powder drums, fuel assemblies), the quantity of nuclear material involved, the dates of receipt/production/shipment and the shipper /receiving facility/State.

2.3.2. Inspection frequency

Each year three to five (average four) SNRIs should be carried out. This number is expected to be reduced at the start of implementation of an integrated safeguards regime. Each SNRI is planned for two consecutive days. They will be selected from all working days of the plant (about 250 days per year). It is expected that the EC inspectors will participate only on the second day, following the commencement of the SNRI triggered by the IAEA.

2.3.3. Notification

IAEA inspectors will be at the gate of the facility from 09:00 to 09:30 and during this window the formal notification fax will be sent to the EC/State authority/operator. This will be followed by a telephone communication to confirm receipt. The inspectors should have access to the facility within two hours from the time of notification (expected access before 11:00). The notification window, in principle, could be adjusted (e.g. to afternoon) upon agreement of the parties concerned.

2.3.4. Residence (retention) period

The residence period is defined as that from the day of material receipt (powder) or product produced (fuel assemblies) until the time when the material is no longer accessible for verification (e.g. consumption of powder in the process or placing the fuel assemblies in closed containers). In this particular plant (Juzbado), the operator was not able to guarantee any residence period, particularly for the fuel assemblies. This is the main reason that it was accepted to perform the SNRIs with the presence of inspectors at the facility gate at the moment of the notification, with a maximum delay of access to the facility of two hours.

2.3.5. Verification

Upon the IAEA inspectors access to the plant, the operator will provide a detailed hard copy of the itemized list of all packed and not-yet-packed finished fuel assemblies. The entire nuclear material inventory list will also be provided. Additional information of the SRD of powder receipts/shipments and receipts of any scrap or other nuclear material will also be provided.

First day activities:

— Verification of the inventory of accessible fuel assemblies, by item counting and identification and random verification by NDA/HM-5 for gross defects and NDA/UNCL for partial defects. Normally two assemblies will be verified by UNCL. In general, no request will be made by IAEA inspectors to open closed containers with fuel assemblies ready for shipment (as this material is already covered by the date selection of the SNRI).
— Pellet sampling at the fuel rod loading station. This activity will be performed in conjunction with access to the process area (with observation of the number of operating process lines).
— Random selection of feed powder drums received since the last inspection as needed for the evaluation of SRD (if applicable).
— Random verification of scrap prepared for shipment (if applicable).

Second day activities:

— Examination of nuclear material accountancy records (book examination).
— Verification of the randomly selected powder drums, including sampling for destructive analysis (DA) as needed for SRD evaluation.
— Other inventory verification activities (e.g. powder, pellets, rods, waste) depending on time availability (optional).

3. Case of force majeure

The operator should inform the IAEA of any known instance that may limit the access and/or the activities of the SNRIs. This notification should be received by the IAEA as early as possible but prior to the opening of the notification window of the SNRI. The cases of possible force majeure are defined in the SNRI procedure. Examples of force majeure are malfunction of handling equipment, emergency shutdown of one operating process lines, safety inspection and natural calamity.

4. Conclusion

It is possible to implement a SNRI regime at a LEU fuel fabrication plant that has no residence time to freeze its feed receipts or produced fuel assemblies and is under regional and IAEA safeguards.. Sincere commitment of all parties that have previously agreed to such a safeguards procedure is needed to start the implementation. Delay in starting the implementation of a SNRI would result in failure to attain the IAEA safeguards goals.

Implementation of SNRI and borrowing inspections at LEU FFPs in Japan

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Abstract. The SNRI (Short Notice Random Inspection) and Borrowing inspections are in place at all four Japanese LEU FFPs (Low Enriched Uranium Fuel Fabrication Plants). Routine implementation of SNRI started in 2000 and ninety-nine SNRIs have been conducted up until Aug. 2006. The most important features of SNRI include 100% verification coverage of material flow into and out of the plant by applying random sampling theory and unpredictability by giving no prior notice of inspection. This is in sharp contrast to the conventional interim inspection regime where prior notification to the state/operator was given and no statistical random selection of inspection date was implemented, thus the detection probability could become nearly zero unless a large percentage of material were by coincidence available at the time of inspection that was practically impossible for the facilities with relatively large throughput like Japanese LEU FFPs. However for implementing SNRI there were number of technical and administrative obstacles to overcome, including development of on-line data transmission system, permanent storage of inspection equipment at the facility including neutron source and sorting-out of the concepts, terminologies and reflection of them to the mutually agreeable procedures. For the borrowing inspections thanks to successful development of the way for speedy estimation of in-process inventories and in-depth consultations between IAEA and state/operators including a number of field studies at the plants, the standard borrowing inspection routines have been established and the actual inspections have been implemented without major problems.

1. Overview of Japanese LEU Fabrication Plants

There are four LEU FFPs in Japan, two plants, Global Nuclear Fuel- Japan (GNF-J) and Nuclear Fuel Industries –Tokai Works (NFI-T) produce BWR fuel and another two, Mitsubishi Nuclear Fuel (MNF) and Nuclear Fuel Industries –Kumatori Works (NFI-K) fabricate PWR fuel. Only MNF converts UF6 to UO2 with receiving enriched UF6 from overseas enrichment plants as well as from the domestic enricher (JNFL Rokkasho-Mura Enrichment plant). MNF mostly uses converted UO2 powder for it's own fuel production however it also converts certain amount of UF6 to UO2 for other LEU fabricators in accordance with the commercial contracts. Other three plants mostly import UO2 powder from overseas supplies besides acquiring relatively small amount from MNF.

2. Summary of Safeguards Scheme Applied to LEU Plants

Conventional comprehensive Safeguards scheme had been applied to Japan up until 1999 whereby five scheduled interim and a PIV inspections a year were conducted for each LEU plant. The borrowing inspection, DIV (Design Information Verification) and environmental sampling were considered possible but they were not well formulated, needless to say there were no documented agreed standard procedures. In the mid 1990s the international community called upon the need for implementing strengthened Safeguards measures that were felt necessary triggered by the revelation of weakness of conventional safeguards scheme including Iraqi clandestine nuclear development program. In response to these voices the Japanese LEU plants collaborated with IAEA's efforts for developing and introducing more effective SG measures including SNRI, routine application of the borrowing inspection and DIV. On top of the above the operators have fulfilled the requirements in accordance with the additional protocol including complementary access measures, submission of additional information. Since the Japanese LEU facilities were entered into the Integrated Safeguards (IS) Scheme beginning 2005, SG measures based on agreed IS approach have been in place: currently they are summarized to be consisting of the following six measures, 1) SNRI 2) PIV 3) Borrowing Inspection 4) DIV 5) Additional Information and 6) Complementary Access.

3. SNRI

3.1. Chronology of SNRI Implementation

Installation of the mailbox computer at JNF (now renamed as GNF-J) facility dates back to October 1996[1)] followed by several tests and rehearsals and the first actual conduct of SNRI scheme inspection took place in January 1998. Having evaluated the effectiveness and practicality of the SNRI procedures developed through the trials/rehearsals carried out at JNF, the SNRI scheme was accepted by the Japanese government and other LEU FFPs to apply, it was formally expanded to cover all FFPs in 2000.

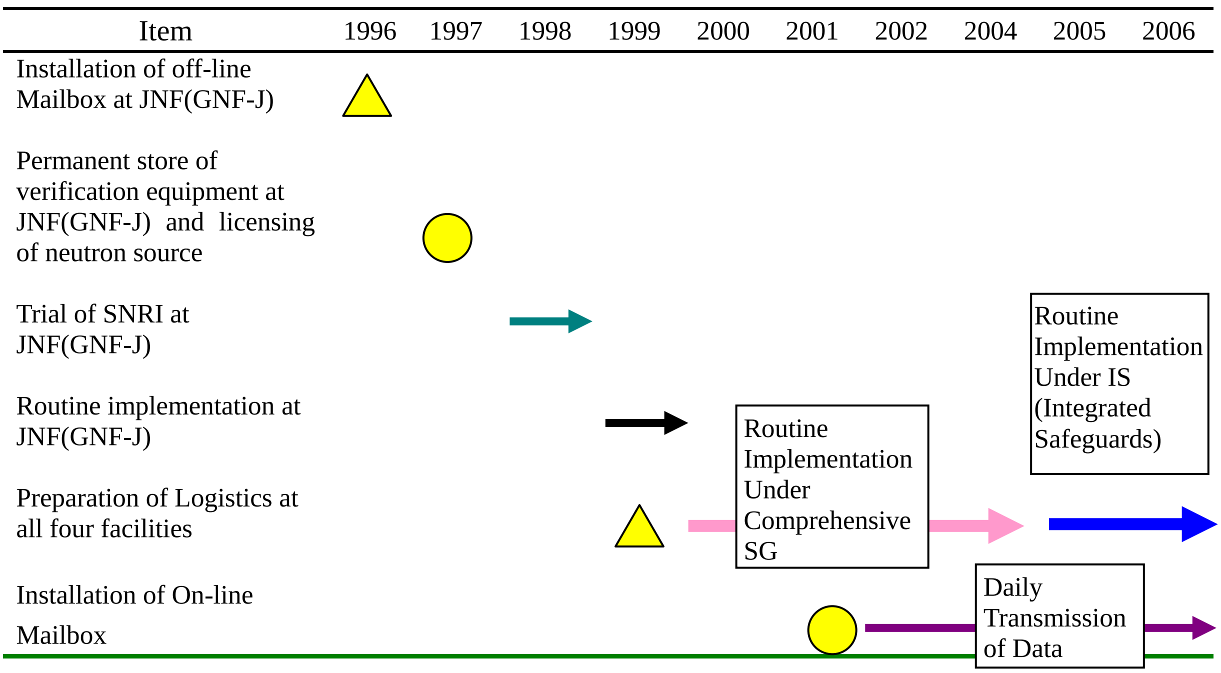

Fig. 1. Chronology of SNRI Implementation.

3.2. Overview of SNRI

Along with the physical inventory verification, the verification of the material flow is an indispensable element of SG approaches. In the case of traditional inspections where the inspection schedule was arranged and established between IAEA/state/operator weeks in advance and the interval between two interim inspections was almost fixed constant, the main activities to carry out at the interim inspection were the interim inventory verification and not for flow verification. The inspection was scheduled virtually independently with the receipt or shipment schedules of the facility. Consequently the verification coverage of the flow was actually very low. On the contrary SNRI makes 100% coverage of verification by applying a randomized statistical sampling technique to the population of annual material flows.

Table 1 summarizes major differences between conventional (or traditional) inspection and the SNRI. As mentioned earlier the logistics such as the measuring equipment could be arranged and transported to the facility just in advance of the inspection in the case of conventional inspection. This practice cannot work in the case of SNRI from the nature of unpredictability and the permanent storage at the facility became necessary. This includes also the need for the facility to obtain necessary licensing for storing Am-Li neutron source. It is without saying that on top of the above the operator and the inspectorates had to make a number of modifications of the computer softwares and procedural arrangements to cope with SNRI.

Table 1. Comparison of SNRI against Conventional Inspection.

Item	Conventional	SNRI
Advance Notice of Inspection	**> 1 week**	Short notice: Verification starts **within 2hrs after notice**
Selection of Inspection Date	Negotiated in advance between IAEA/State/Operator: thus **No need to have stand-by inspectors**	**All operational dates** are Potential SNRI date: IAEA initiates. No refusal allowed. **State inspectors are on stand-by on every operational day**
Frequency	5/Year	5/Year now 2/year at IS mode
Verification Coverage	**20%** of Receipts/Shipments	**100%** of Receipts/Shipments
Verification Practice	**Receipt**: Coincidentally existing items at inspection day. **Assembly**:100% of unpacked Items (inventory verification) **Pellet sampling** for evaluation of operators measurement system	**Receipt**: 100% **Assembly**:100% of newly produced items **Pellet sampling** for evaluation of operators measurement system
Measurement Equipment	Transportation by the day before inspection	**Permanent storage** at the facility including neutron source
Preparation of Item Listings and Logistics	On the days before inspection	**Right after Notification in parallel with inspection**
Mailbox	None	**Daily posting** and transmission

3.3. Terminology

The fundamental concepts of SNRI are well represented by discussing the terminology of three technical terms, i.e., the Birth (B), the Death (D) and the Residence Time (RT or Lifetime). In the case of an assembly the Birth is defined in general, as the event when it becomes verifiable by assembling operation, however in the case of two facilities it is defined as the QC release date that usually a couple of days behind the assembling. The D takes place when it becomes unverifiable by packing operation into shipping crate. In the case of received powder or UF6, the B in general coincides with the day or the next day of customs clearance for foreign imports or the completion of unloading from the transport vehicle for domestic receipts. The D is defined in general, as the time when the item becomes unverifiable by moving to the process station.

Actual definitions of B and D vary from a facility to facility, thus facility specific definitions need to be agreed upon between the inspectorates and the operator and they must clearly be described in the procedures for each facility.

Table 2. Concepts and Terminology of SNRI.

Item	Definition	UO2 Powder/UF6	Assembly
Birth(B)	Material to become Verifiable by a processing or movement	Receipt (the day or the next day of unloading or customs clearance)	Completion of Assembling or QC Release
Death(D)	Verifiable material to become unverifiable by moving to process	Release to process	Packing to Shipping Crates
Lifetime or Residence Time(RT)	Verifiable Period (Death date – Birth Date)	Typically several days	Typically several weeks

3.4. Data Transmission

A great advantage of the Japanese SNRI approach is the routine operation of the so-called on-line “mailbox” system. As the structure of the system is illustrated in Fig.2 each facility is equipped with a PC connected on-line to JSGO (Japanese Safeguards Office: State Authority concerning Safeguards Matters). The operator collects the data from his own host computer system, converts them to fit to the agreed data structure, and then posts to the remote PC terminal. The software on the PC encrypts them after QC check and transmits to JSGO server. The posting is done at the latest by 15:00 for every working day for the transactions of the previous day. The IAEA server collects the data from the JSGO server. By accessing to the servers the inspectorates can have up-to-date knowledge on the material flows and relevant inventories of the receipts and the products for inspection planning and other purposes.

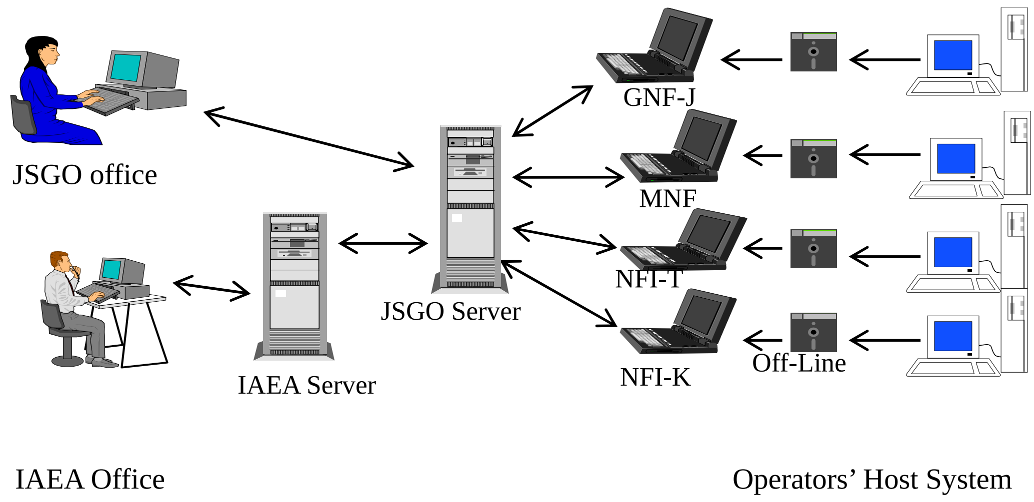

Fig. 2. Mailbox and Data Transmission.

3.5. B, D Data Statistics

In light of giving the inspectorates maximum freedom of choosing the most appropriate date for SNRI, it is important to keep the flow items verifiable as long as possible. On the other hand the operator needs to be as much flexible as possible for production scheduling purposes. For the assemblies it was mutually agreed to set the Minimum Residence Time (MRT) to seven working days. The MRT is defined, as the minimum lifetime during that period an item must be kept as verifiable. The fuel assemblies are obliged to undergo 100% of internal quality inspection followed by the inspections by the customer and the governmental body (METI: Ministry of Economy, Trade and Industry) that are made on a sampling basis on a couple of days a month. Only after completing the final METI inspection the assemblies can go to the packing process. Therefore they stay verifiable for a considerably longer period. The statistics of GNF-J (Fig.3) show the actual minimum residence time was 20 days and the average was 73 days. While the detailed statistics vary from a facility to facility, the seven days MRT imposes no real constraints to the plant operations of any facility.

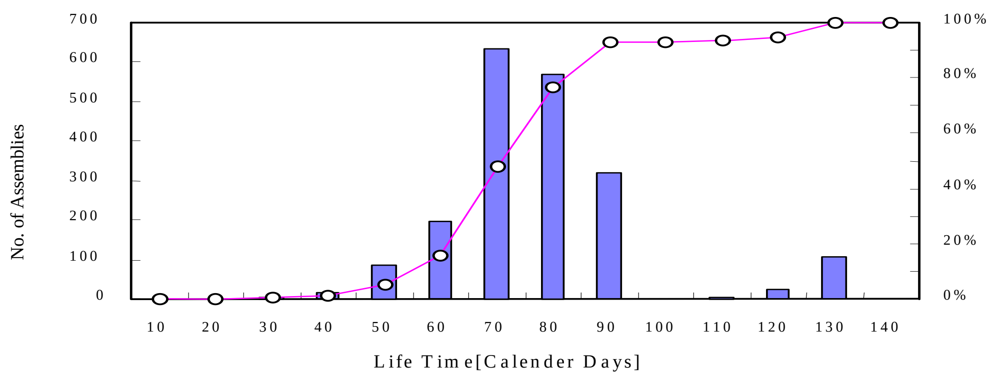

Fig. 3. Statistics of Assembly Residence Time of GNF-J.

On the contrary, for the received material, UO2 powders in particular, there are no customer or governmental inspections to undergo except for the customs clearance to the imported items. Therefore they are mainly dictated by the plants production scheduling that is aiming at reducing cost by lowering inventories among other efficiency improving measures. Fig.4 shows the actual residence time statistics for GNF-J, the average was about 8 days and more than 2% of powder falls with 0 residence time.

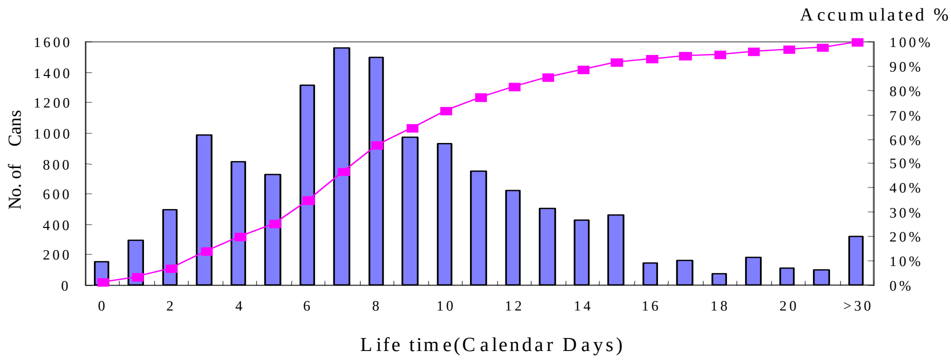

Fig. 4. Statistics of Powder Residence Time of GNF-J.

To give the opportunity to verify such short life material the operator notifies the inspectorates well in advance the relevant information including the day of birth, number of items, the quantities involved. Since the first implementation of SNRI the residence time for the powder has not been made crystal clear because of somewhat confusing interpretation of Birth. Now however it was agreed by the operator to collaborate even more to the inspectorates to have the MRT of 2 working days (WDs).
Fig.5 illustrates the timeline of the powder receipt. Regardless of the shipper the B is now defined to be the next working day of the receipt that is D2 and the material is guaranteed to stay verifiable up until the end of the agreed window period (9:30-10:00 AM) on D4. Should the SNRI take place on D4 100% of the population that was born on D2 would be verifiable. Since the definition, the MRT is Earliest Death (D4) – Birth (B2)=2 WDs. However the inspectorates are guaranteed to have the opportunity to verify a hundred percent of population of the receipt for three WDs from D2 to D4. The operator on the other hand has to endure the burden to keep material verifiable without sending to the processes for the material arrived, for example, on Wednesday up until 10:00 AM of the following Monday even if they are required urgently for the production purposes. In other words the operator must plan the production schedule in such a way that they would not use the receipt of arrival on Wednesday until Tuesday next week for six calendar days.

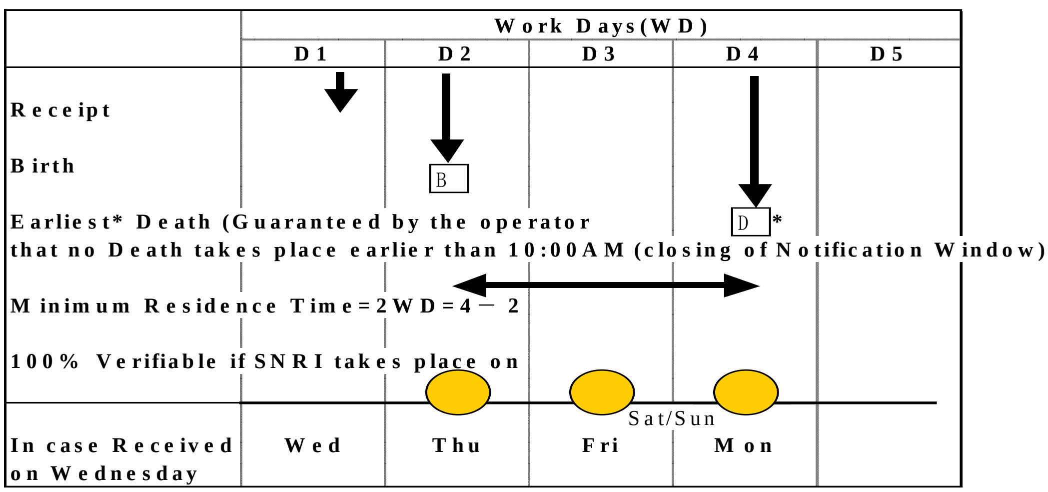

Fig. 5. Timeline of Powder Receipt at GNF-J.

With the daily entry of the transaction and the remote transmission the inspectorates are now able to maintain the continuity of knowledge of the receipt and the products by just accessing the server. Joined with the advance notification it is expected the inspectorates utilize them for even more efficient and effective safeguards activities. Fig. 6 and Fig. 7 are just examples of the statistics that can be obtained from the mailbox declarations.

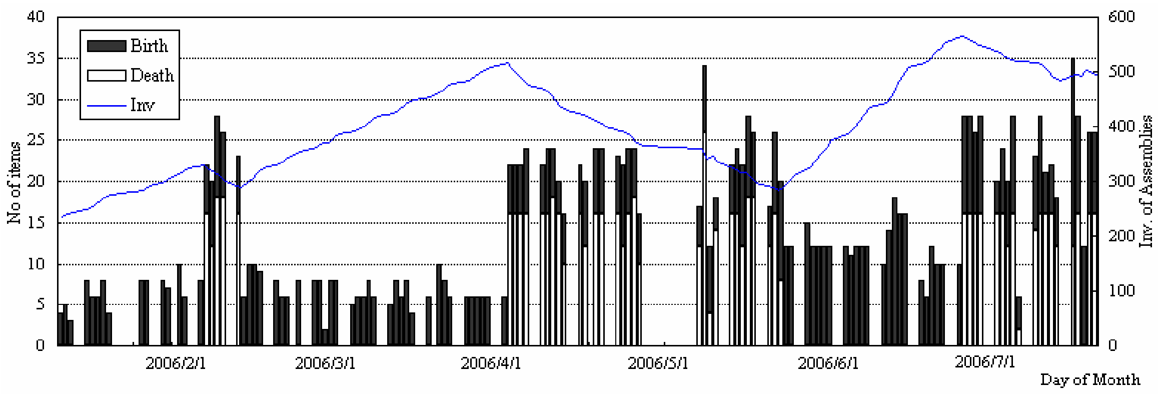

Fig. 6. Daily Birth, Death and Inventory statistics derived from mailbox entries (Assembly).

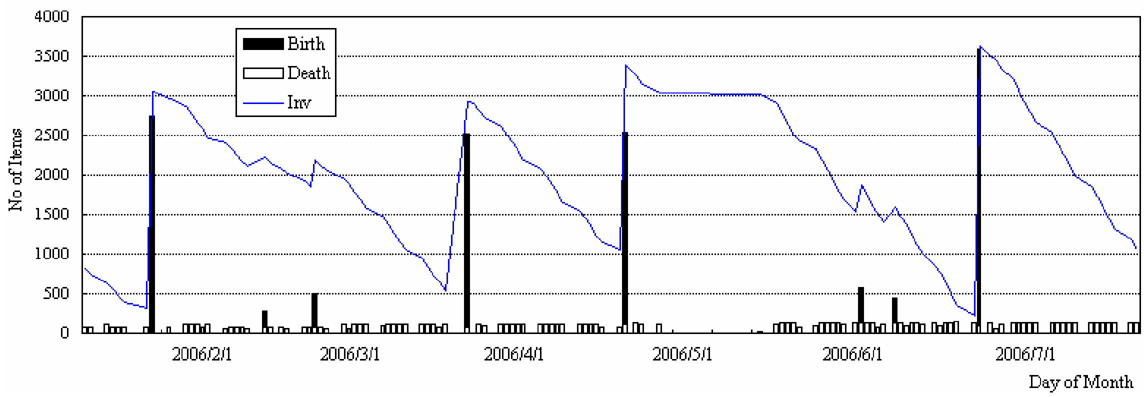

Fig. 7. Daily Birth, Death and Inventory statistics derived from mailbox entries (Powder).

3.6. SNRI Timeline

As shown in Fig.8 the IAEA notifies the operator through JSGO during the notification window that an SNRI is to be carried out indicating the facility, activities to be carried out and names of inspectors. With the exception of NFI-K, the IAEA inspectors arrive at the facility at minutes after 10AM, joined by national inspectors (JSGO/NMCC) and start NDA equipment setup/calibration. The operator prepares necessary data including the general ledger; location maps and itemized listings of the subjected population and submits the inspection team with electronically readable media. Accordingly the team generates a sampling plan and carry out verification activities such as item counting (IC), identification (ID), weighing, DA sample taking and NDA on top of book auditing. For the evaluation purpose of the operator’s measurement system pellet samples are also withdrawn from the station near to loading processes. Thanks to proximity of the location to NMCC Tokai office the inspection team can start full inspection work pretty early for MNF and NFI-T and the inspection is usually completed in a day with some extra overtime work. In the case of GNF-J which is located about 200Km south of Tokai the state inspectors from Tokai only arrive in the early part of afternoon Consequently the time allowed for actual verification is very limited to the extent that assembly verification, pellet sample taking and IC, ID, sealing of the sampled powder containers in addition to book audit activities can barely be finished in the first day. For the transportation of enriched uranium powder the NPC container that normally contains 27 items with total weight being around 0.5ton UO2 is used. Since withdrawal and movement to the measurement station from the store of the NPCs that are randomly chosen for verification needs a considerable time period. The store is densely packed and no sufficient space available for reshuffling for taking the targeted NPCs out from the deep position of the store. Therefore usually the follow-up verification day is to be agreed upon between the operator and the inspectorates. On the follow-up day the items in the selected NPCs are verified according to standard procedures. For NFI-K that is located more than 500Km away from Tokyo the state inspector from Tokyo can only get on site around 2PM even by flying to the airport luckily situated near the site. The first day activities are limited to sample planning, IC, ID and sealing of the sampled items. Most of the verification activities including book audit are carried out on the second day.

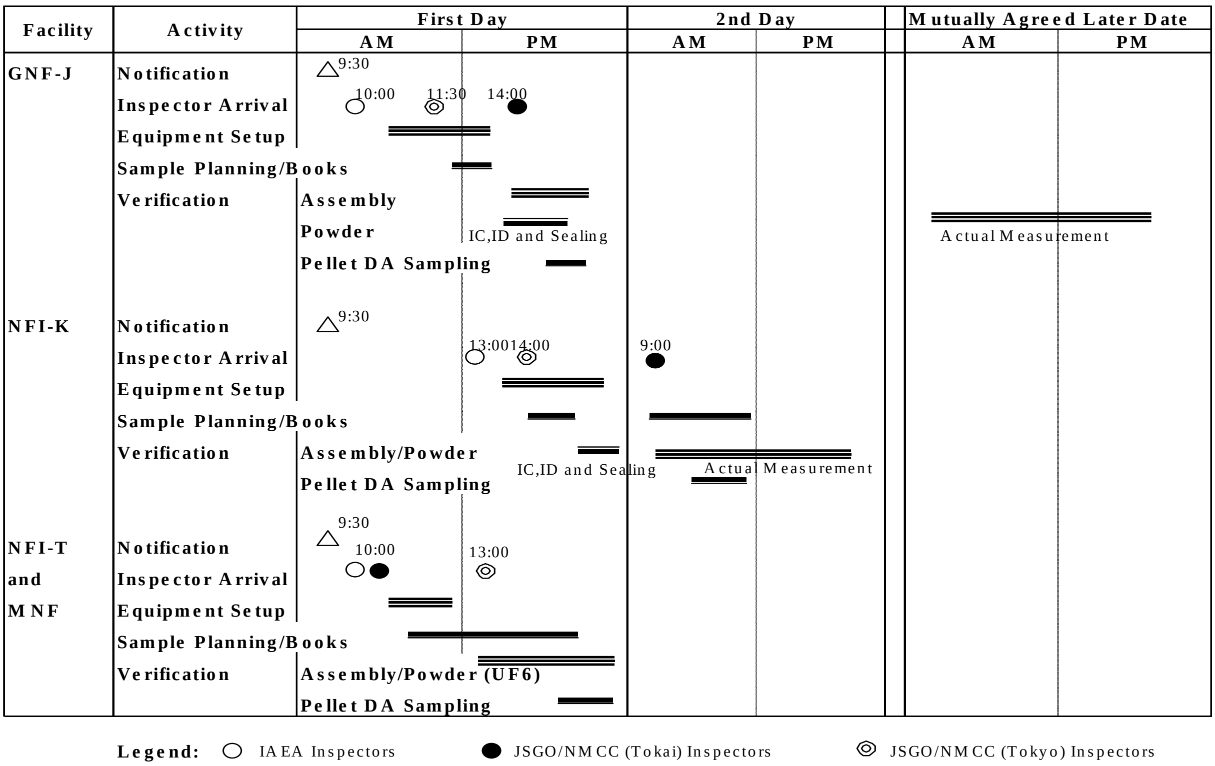

Fig. 8. Typical SNRI Timeline.

4. Borrowing Inspection

The borrowing inspection is to confirm no hypothetical borrowing scenario between LEU FFPs occurred. For this purpose the simultaneous verification of a randomly chosen borrowable stratum of a randomly chosen FFP has been in practice with the PIV of an LEU FFP. In other words at the time of the PIV at one of LEU FFPs, one of other facilities will be randomly selected for a borrowing inspection on short notice. One of the most challenging activities in the borrowing is the way to establish the physical inventory without stopping the processes. While the operator generates the inventory listings for storages that are supposed to be relatively static, it is not possible for the materials right moving through the processes. For this purpose the IWES (In-process Walk-through Examination System) has been developed in full cooperation of IAEA/JSGO/NMCC/Operators and successfully in use since 2004. According to the pre-programmed route the inspectors together with operator escorts walk through the processes and make speedy estimation of the inventories in the processes. (For the details refer to Ref.3).

5. Conclusion

Both SNRI and borrowing inspections have been developed and conducted in such a way as to improve effectiveness and efficiency of IAEA Safeguards. The parties, IAEA/JSGO/NMCC and the operators worked hard to developing and implementing these new innovative SG approaches, in a very cooperative manner, the operators recognizing the need of improving the IAEA SG, and the inspectorates understanding the spirit of SG agreement [4) in particular, "avoiding undue interference in the operation of facilities" and "concentration of verification procedures to direct use material and minimization of verification procedures to indirect use material". Now that the IS regime has been introduced to the Japanese FFPs since 2005 that the frequency of SNRI and borrowing inspection is drastically reduced (from average 5 to 2 annum for SNRI and 0.5/year*facility for borrowing inspection), one of the operators' biggest challenges is the way to maintain the quality of inspection support and keeping motivations to the highest level without having frequent inspections. Potential solutions will be timely consultations for improving practices including solving problems and active involvement of inspectorates and operators for the refinement and periodic revisits of relevant practices.

ACKNOWLEDGEMENTS

The authors are grateful to inspectors for their terrific endeavors to the development of the SNRI and borrowing inspection regime. The former IAEA staff, Messrs. Alston and Huenefeld were the driving force for introducing SNRI. Without them the SNRI concept would never been materialized in the real field. P. Durst was the inventor of the "walk-through" concept for borrowing inspection. IAEA inspectors, Messrs. Tsvetkov, Krpo, Burmester and NMCC inspector Namekawa have given great contributions for developing and actual implementation of SNRI and borrowing inspections.

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The transition to integrated safeguards: The Canadian experience

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Abstract. Canada attained the broader safeguards conclusion in September, 2005. This paper describes the Canadian experience in our quest to attain the conclusion and in interacting with the IAEA and with Canadian industry on the development of a State-level approach for Canada to be implemented after the conclusion had been attained. The paper will describe the process which featured an open and transparent dialogue between the IAEA, the SSAC and Canadian operators - a key element in all stages of development. The paper will also outline Canada's preparations for signing and implementing the Additional Protocol, our input into the conceptualization of an IS approach for Canada, and our approach to attaining the conclusion. Finally, it will identify some of the challenges that remain to be addressed in transitioning to the full implementation of an IS State- level Approach.

1. Introduction

For Canada, the journey along the road to integrated safeguards has been primarily influenced by two factors: (i) the presence of an extensive, well-developed nuclear industry dedicated to peaceful uses and (ii) a strong and consistent commitment to nuclear non-proliferation. The first factor introduced the reality that safeguards implementation in Canada pursuant to the Canada/IAEA safeguards agreement was likely to consume a considerable portion of the Agency's safeguards resources. The second factor greatly influenced Canada's decision to support and help conceptualize and develop new measures for strengthening the international verification system administered by the IAEA and to expeditiously adhere to the enhanced verification norm created by them. Thus, for Canada, as for many other States, the goal is to have a credible and effective safeguards system that is implemented as efficiently as possible. In this context, the attainment of the broader safeguards conclusion was important in order to ensure that the IAEA could provide the basis for the international community to have the highest level of confidence possible, pursuant to the safeguards system, in the peaceful use of nuclear energy in Canada. At the same time, the broader safeguards conclusion would provide the basis for maximizing the efficiency of safeguards implementation in Canada through the implementation of a State-level integrated safeguards approach.

This important milestone (the broader safeguards conclusion) was achieved in September, 2005. Currently we are working with the Agency on moving towards the implementation of a Canada-specific State-level integrated safeguards approach. The remainder of this paper will examine how we were able to achieve this milestone and identify some of the challenges that will need to be addressed in the future.

2. The Canadian Fuel Cycle

Canada has a significant nuclear fuel cycle that has developed over the last 60 years. The seeds of Canada's nuclear program were sown during World War II when Canada participated in the Allied war effort. Since that time, Canada has pursued the peaceful uses of nuclear energy through the development of an extensive nuclear industry which serves both domestic and export markets. Canada's nuclear power program is based upon natural uranium fuelled, heavy water moderated reactors, commonly referred to as CANDU reactors (CANadian Deuterium Uranium). The nuclear fuel cycle includes: uranium mining and milling; refining and conversion; fuel fabrication; nuclear power reactors; research reactors; spent fuel storage; and research and development activities (see Table 1).

Table 1. Main Elements of the Canadian Nuclear Fuel Cycle.

Type of Facility	**Number of Facilities**	**Location**
Mines & Mills	Numerous	Saskatchewan
Uranium Refineries	1	Ontario – Blind River (U3O8 to UO3)
Conversion Plants	1	Ontario – Port Hope (UO3 to UF6 and UO2)
Fuel Fabrication	3	Ontario – Toronto, Peterborough, Port Hope
Power Reactors	22	Ontario – Pickering (8), Bruce (8), Darlington (4)
		Quebec – Gentilly (1)
		New Brunswick – Point Lepreau (1)
Large Research Establishments	1	Ontario – Chalk River

In addition, to the facilities noted above, there are a number of research reactors and sub-critical assemblies located across the country; 3 shutdown reactors; and several spent fuel dry storage facilities. Accordingly, facilities and locations relevant to safeguards implementation span the entire country with a concentration of power production and research activities in eastern Canada and mining activity in northern Saskatchewan.

3. The Canadian Approach

Fundamental to the attainment of the broader safeguards conclusion is adherence to and compliance with the instruments that provide the basis for such a conclusion. The comprehensive safeguards agreement between Canada and the IAEA pursuant to our NPT obligations entered into force in February, 1972. In September, 1998, Canada signed the Additional Protocol (AP) to the comprehensive safeguards agreement. The AP introduced new elements to the traditional safeguards approach. Accordingly, it was essential to ensure that we were in a position to meet these new requirements.

At the same time, Canada was actively engaged in efforts to conceptualize and develop State-level integrated safeguards approaches that could be implemented after the broader safeguards conclusion was attained. Obviously, particular emphasis was placed on the Canadian context..

3.1. The Additional Protocol Preparations and Implementation

One of the most important considerations relevant to ensuring that the requirements of the Additional Protocol are met is the establishment of an appropriate formal legal framework. In Canada's case, this was accomplished through the promulgation of the Nuclear Safety and Control Act (NSCA) and its associated Regulations which entered into force in May, 2000. The NSCA sets out broad powers for the Canadian Nuclear Safety Commission (CNSC) to regulate the use of nuclear energy and materials to protect health, safety, security and the environment and to respect Canada's international commitments on the peaceful use of nuclear energy. The requirements of the safeguards agreement and the AP are incorporated into safeguards license conditions which are part of the relevant licenses issued by the CNSC. These conditions delineate obligations on the licensee regarding the provision and updating of information and records, access by IAEA staff, the provision of services to IAEA staff, the provision of assistance in sampling and measurement, and the installation and maintenance of safeguards equipment.

A second component of the legal framework is the Subsidiary Arrangements (SA) with the IAEA. While not required by the AP, Canada considered it necessary, in light of the extensive and complex nature of its nuclear activities, to negotiate specific arrangements related to the AP such as all forms of complementary access and information reporting. These SAs became the basis for model AP SAs now in use by the IAEA.

Canada, through experience, was aware that the early and continued involvement of the Canadian nuclear industry in the development and implementation of safeguards measures is of critical importance. Canadian industry outreach on the strengthened safeguards regime began in 1994. This outreach featured discussions on the framework of strengthened safeguards, the impact on industry and the modalities of implementation of the required measures. These industry consultations proved to be invaluable as they provided Canadian industry with a forum to express their concerns and to provide practical input regarding the feasibility and effectiveness of proposed measures. Participation in field trials to test various elements of strengthened safeguards proposals, such as enhanced access to information and locations, was another successful vehicle for engaging industry and for facilitating an understanding of the objectives of the new measures.

After the IAEA Board of Governors approved the *Model Protocol Additional to the Agreement(s) Between State(s) and the International Atomic Energy Agency for the Application of Safeguards (INFCIRC/540)*, the focus of the industry outreach changed. Emphasis was placed on the issues associated with implementation of the Additional Protocol, such as the production of Canada’s initial declarations, the establishment of procedures for complementary and managed access and the reporting requirements. This outreach program included a meeting between industry representatives, the IAEA and the CNSC in April, 2000, at which the IAEA DDG-Safeguards and his senior staff directly engaged the industry on the importance of the AP. Additional meetings of this nature were held, the most recent being in June, 2004.

Throughout 2000 and early 2001, the CNSC prepared the procedures and mechanisms for the collection of information for Canada’s declarations under the Additional Protocol. A dedicated project team was formed to achieve this task. Industry outreach at the facility level, as well as the establishment of data handling procedures and processes, were elements of the project. The collection of information was assisted through IAEA guidance provided in the guidelines and format document for the preparation and submission of declarations pursuant to Articles 2 & 3 of the AP. Additionally, the Protocol Reporter software was also distributed to most of the affected locations in Canada and workshops were done with industry to explain the use of this new tool. The project team also undertook extensive discussions with the IAEA on all aspects of the protocol requirements. This included providing the IAEA with an advance draft of the initial declarations for discussion. This project concluded in March, 2001, when Canada submitted a set of wholly electronic declarations under Articles 3.a and 3.d of the Additional Protocol. The official declaration was hand-delivered by Canadian governmental officials in Vienna to the IAEA on a single CD-ROM containing both the textual descriptions and the site maps. In order to test electronic transmission channels for future use, the declarations (and accompanying documents and images) were also remotely transmitted unofficially to the IAEA.

The establishment of reliable and consistent procedures for complementary and managed access, for addressing questions and inconsistencies and for providing the annual updates pursuant to the declarations is another important element of the implementation of the AP.

Access procedures were required to address the entire nuclear program, including mines in remote locations and shutdown facilities. In consultation with both the IAEA and the industry, the CNSC established the necessary procedures and recorded them in a separate, internal manual which is reviewed annually. An important feature in the procedures is the requirement for operators at the locations subject to complementary access to provide a post-complementary access report to the CNSC. Regarding managed access, the CNSC strategy was to indicate in advance the areas or situations requiring this treatment. This was done by highlighting these areas of managed access or other access considerations within the text of the declarations for individual sites or locations in the AP declaration. The identification of site contacts and of locations requiring managed access was also included in the Subsidiary Arrangements.

Procedures for handling any questions or inconsistencies arising from Canada's AP declarations were also established by the CNSC. These procedures ensure prompt distribution of questions and inconsistencies to the relevant site or location and the timely submission of appropriate answers and explanations to the IAEA. Finally, according to agreed procedures, notification is sent each November from the CNSC to senior management at each location requesting an update of the information previously submitted pursuant to Canada's Article 3 declaration. Upon receipt and after satisfactory review, the information is assembled by the CNSC and sent by electronic channels in a secure encrypted transmission to the IAEA.

In addition to establishing the necessary framework to ensure that Canada could meet its increased obligations following the entry into force of the AP, the CNSC had to contend with the inevitable increased verification effort undertaken by the IAEA. This effort was directed towards establishing the basis for the Agency to be in a position to draw the broader safeguards conclusion. It included increased verification activity at facilities and locations in Canada (e.g., complementary access), detailed analysis of information available to the Agency - particularly that pertaining to historical activities that pre-dated the safeguards agreement - and the adjustment of some practices and procedures that had been part of safeguards implementation in Canada in the pre-strengthened safeguards period. All of this led to extensive and continuous interaction between the CNSC and the IAEA in order to provide clarifications and to address questions and inconsistencies. The requirement to bring Canada's natural uranium refining and conversion facilities under safeguards as a result of the Agency's new policy concerning the starting point of safeguards was an additional challenge during this period.

In an attempt to establish a structured approach to ensure effective and timely resolution of substantive issues, Canada sought clarification from the Agency as to those issues which required resolution prior to drawing the broader safeguards conclusion and those issues that could continue to be addressed after the conclusion was drawn based upon an agreed course of action. The IAEA provided such a listing - commonly referred to as a "Roadmap" - identifying the first set of issues as major issues to be resolved and the second set as minor issues requiring a longer term approach. This development significantly improved the rate of progress as it enabled work to be prioritized and to be systematically completed.

3.2. Conceptualizing, Developing and Implementing a State Level-Approach

While considerable effort in the post-2000 period was expended on attaining the broader safeguards conclusion, a similar amount of effort was expended on contributing to the conceptualization and development of safeguards approaches that could be utilized in Canada once the conclusion had been attained. In fact, the beginning of this effort pre-dated the conclusion of the AP.

As early as 1990, the CNSC initiated industry outreach on the future development of safeguards in both the global and the Canadian context. This investigative look at the possible element for an effective and efficient strengthened safeguards regime was the beginning of Canada's deliberations on the optimization of safeguards measures. The outreach led to open and transparent trilateral discussions between the CNSC, the IAEA and Canadian industry resulting in field trials in Canada which tested the concepts that were being considered for strengthening the Agency's safeguards system.

In September 1998, the IAEA convened an expert group meeting on integrated safeguards. The Canadian representative at this meeting was able to bring to the table the results of Canada's experience in conceptualizing the integration of traditional safeguards measures with the new measures in the Additional Protocol. This meeting was followed by several additional consultative meetings between the IAEA and Member States on the development of concepts for specific State-level integrated safeguards approaches. Canada prepared a concept paper for a State-level integrated safeguards approach for Canada which emphasised openness and transparency and the optimisation of the safeguards approaches already being utilized at several Canadian facilities. The essence of this proposed approach was the provision of near real time information on nuclear material flows throughout the fuel cycle by means of mailboxes and the remote transmission of data from installed safeguards equipment, and the use of short notice and/or unannounced access to locations within Canada to verify this information.

In addition to focusing on a possible State-level integrated safeguards approach, the CNSC also undertook a Safeguards Support Programme Task to develop and test an integrated safeguards approach for transfers of spent fuel to dry storage at CANDU multi-unit stations in Canada – an activity that demands the utilization of a significant proportion of the Agency's inspection effort.

Subsequent preliminary discussions with the IAEA on a possible State-level integrated safeguards approach for Canada indicated that the continued use of instrumentation would remain an important element of such an approach. Accordingly, Canada proceeded to systematically upgrade these capabilities. At this time, all data from installed safeguards equipment at nuclear power stations (eg, Bundle Counters, Core Discharge Monitors and surveillance systems) is being transmitted remotely to the IAEA. In addition, data on the state-of-health of this equipment is being transmitted.

In September, 2004, Canada and the IAEA agreed to establish a working group to discuss specific facility and State-level parameters that could be relevant to the State-level integrated safeguards approach for Canada. Obviously, the responsibility for developing and approving the approach is vested in the IAEA. However, this consultation mechanism enabled the CNSC to have a better understanding of what the requirements might be and to keep Canadian industry informed. The working group met six times.

During this same period a working group was established to deal with the implementation of Policy Paper 18 specifically bringing the Cameco Blind River refinery and Port Hope conversion facility under safeguards and folding these two facilities into the State-level approach. This exercise was accomplished with outreach to industry and included the participation of Cameco senior management in meetings with the IAEA to discuss the operational parameters of the plants and specific characteristics of the material inventories and flows. This dynamic meeting proved invaluable in moving forward the implementation of safeguards and the development of the State-level approach.

Following the attainment of the broader safeguards conclusion in September 2005, the CNSC and the Agency began discussions on implementing the State-level integrated safeguards approach for Canada that had been developed by the Agency. It was agreed that implementation should be pursued based upon agreed priorities. It was also recognized that, since some elements of the Agency's State-level approach could require additional resources on the part of the CNSC, full implementation may not be possible until such resources are obtained. In other words, the extent of the CNSC's contribution to the exercise was dictated by available resources.

A CNSC/IAEA working group was established to begin detailed consideration of the implementation plan. To-date, the working group has focused on sector and/or facility-specific implementation issues and on the development of implementation procedures. In keeping with past practice, Canadian industry is being regularly apprised of the status of the exercise and have been provided opportunities to provide input, either in writing or through attendance at trilateral meetings. The goal is to begin implementing the State-level approach by the end of this year.

4. Conclusion

This paper has focused on Canada's experience and efforts in attaining the broader safeguards conclusion and in moving towards the implementation of a State-level integrated safeguards approach. While recognizing that some of Canada's experience along this road may be unique, some of the approaches and the lessons learnt may be relevant to other States. In this regard, three particular elements arising from our experience may be of particular interest:

(1) It is important to ensure that the necessary legislative framework to facilitate the implementation of the new measures arising from the Additional Protocol is established as soon as possible;

(2) Early and frequent consultation and cooperation with the IAEA in both the preparation and the implementation phases is essential to facilitate the process and to communicate expectations; and

(3) Early and frequent consultation with industry is essential to promote understanding, "buy-in", cooperation and practical advice.

Much has been achieved but much remains to be done. Having attained the broader safeguards conclusion, Canada must continue to work diligently to maintain it. Challenges remain with respect to the full implementation of the State-level approach as it has been identified by the Agency. Work is continuing on the agreed approaches to address the minor issues identified in the Roadmap document. Further, additional effort will be required to re-examine the conceptual framework for integrated safeguards that has been established by the IAEA in order to take into account experience gained in its implementation as well as relevant technological advances. This effort could lead to consequential changes in the State-level approach for Canada. Finally, the CNSC, as the national nuclear regulator and as the State System of Accounting for and Control of Nuclear Materials (SSAC), must be in a position to ensure Canadians as well as the international community and the IAEA that nuclear material in Canada is properly accounted for and that it is being used solely for peaceful purposes.

Activities of the ESARDA Working Group on Integrated Safeguards

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Abstract. ESARDA, the European Safeguards Research and Development Association, pursues as a main objective to assist the European safeguards community with advancing progress in safeguards and enhancing the efficiency of systems and measures. Key bodies of ESARDA are standing Working Groups dealing with various technical subjects.

The ESARDA Working Group on Integrated Safeguards was created in 2000 with the objective to provide the Safeguards Community with expert advice on methodologies and approaches to integrate INFCIRC/153 and INFCIRC/540 measures and to present a forum for the exchange of information, views and experiences in that regard. Its members represent inspectorates, national authorities, operators and research centres active in the field of safeguards.

The Working Group very soon realized that a first milestone on the road to Integrated Safeguards is the successful and functional implementation of the Additional Protocol. Discussions and activities concentrated on actions necessary to reach this end thereby taking into account the specific situation in European States.

Among the topics discussed were issues of

- how to establish a functional site definition for different types of installations, ranging from small locations with very small amounts of nuclear material to complex installations with a complex history;
- how to deal with different and even conflicting requirements in the context of unannounced inspections;
- how to interpret and handle the requirements for R&D declarations considering the needs and interests of all parties involved; etc..

The IAEA participated in most of the meetings as an observer and provided the group with valuable background information on Additional Protocol questions and received in turn a deep insight into our considerations, motivations and concerns. This procedure contributed much to a better mutual understanding that is not least reflected in the revised draft of the AP implementation guidelines.

1. Introduction

The Additional Protocol (AP) entered into force in the European Union (EU) Member States on April 30, 2004, just one day before the enlargement of the union to now 25 members. Already during the AP negotiating phase, and especially after the signing of the AP on September 22, 1998, activities have been undertaken in EU countries to assess issues and consequences of the new AP measures for states and operators. The implementation of the AP and the development of Integrated Safeguards have been under broad discussion within the European safeguards community.

ESARDA, the European Safeguards Research and Development Association, has given these developments and activities high attention by organizing special sessions and seminars on respective issues. Taking into account, however, that the whole nuclear industry and nuclear related infrastructure, including nuclear facilities, nuclear supplier, research centres and authorities within the European Union, will be affected by the measures of the AP and by the application of Integrated Safeguards there has been a common understanding that a permanent forum for specific and detailed discussions and for exchange of information would be needed. The ESARDA Working Group on

Integrated Safeguards (IS WG) was established in May 2000 to satisfy this need, i. e., to serve as a platform for information exchange and to provide assistance to operators, member states, and national, regional and international inspectorates on the implementation of Integrated Safeguards.

2. The Structure of the Working Group

Most of the members founding the group had already worked in the field of Integrated Safeguards, e. g., within the framework of the Member States Support Programmes to the IAEA. They were familiar with the objectives and regulations of the AP and the basic approaches for Integrated Safeguards as developed by the IAEA.

The focal point of the constituting meeting was to define the scope of activities and the main fields of work for the group. There was a common understanding among the participants that the work should mainly be oriented towards practical issues of the implementation of the Additional Protocol and Integrated Safeguards approaches taking into account the point of view of all parties involved in the implementation process, such as operators, as well as national, regional, and international inspectorates. It was clear from the beginning that there was no intention to develop own approaches since the participants were well aware that the group did not have the necessary capacity and resources at its disposal to do such a work. The group would rather concentrate on the assessment of documents available and draw its own conclusions.

Among the first activities after bringing the WG into being was to fulfil some administrative requirements, such as to establishing terms of references, an action plan, and to define success factors. These are instruments to be used in the ESARDA Working Groups to guide the direction of the group's activities and to survey the efficiency of their work. On the basis of the common understanding described above the terms of reference and the action plan were discussed and set up accordingly (see reference 2).

During its now six years of lifetime, the group has proved to be very active and productive. The first constituting meeting started with six participants present. This number quickly increased and is still increasing with the accession of the new EU member states to now more than 20 members or observers from more than twelve European countries, with representatives from the inspectorates of EURATOM and the IAEA. This indicates a great interest in the subjects of AP implementation and Integrated Safeguards. Most members have a strong background of practical experiences in implementing and carrying out safeguards measures, since most of them are representatives of nuclear operators or national authorities. The group meets regularly about 3 times a year at different locations, hosted by one of the sending organisations, and the meetings have always been well attended.

EURATOM representatives participated in the meetings from the start and the group soon recognised that the participation of the IAEA would be very fruitful for both sides, too. For the group it is a big advantage to receive first hand information on new considerations, concepts and developments associated with IS. For the IAEA it is very valuable to gain a direct and unfiltered insight into and thereby a better understanding of our reflections and concerns to implement the new safeguards elements into real life under the various given circumstances in our countries. The IAEA followed our invitation and has soon become an indispensable part of our group.

3. Topics of Discussions

First discussions concentrated on new elements and new approaches proposed for Integrated Safeguards, which the WG examined under the point of view of how to implement this in practice. The group reflected aspects of how the new elements could be translated to and carried out in real life, what effects this could have to current practice of plant operation, what administrative procedures and regulations were affected, etc.. During all these discussions the group drew up lists of open issues and questions that needed further consideration and solutions.

3.1. The Role of R/SSAC

In Integrated Safeguards, the increased co-operation with Regional and State Systems of Accounting and Control (R/SSAC) will be an essential element which is believed to have a high potential for savings in IAEA inspection efforts. The group discussed, quite from the beginning, the role the European regional and state systems could play in Integrated Safeguards. Input to the discussion on this issue was provided in several meetings by presentations from group members. Participants from the United Kingdom presented the results of their studies on the application of approaches to Quality Assurance and compliance used in other industries to R/SSAC's, group members from Finland and Sweden portrayed their respective State Systems of Accounting and Control und representatives from EURATOM and IAEA briefed the group on the results of their considerations and expert group work.

Good experience with respect to saving in Agency efforts has been gained through the co-operation between IAEA and EURATOM within the New Partnership Approach (NPA). EURATOM is currently under way to reconsider its mission and its own safeguards approaches. The group closely follows the discussions on these issues and will try to analyse the effect the changes in the new safeguards approaches may have to the implementation of Integrated Safeguards.

3.2. Safeguards Approaches under IS

The group also closely follows the development of Integrated Safeguards approaches for the different facility types. These approaches and the respective safeguards criteria, as far as information is available, are analysed with respect to its consequences and effect to the operators and the authorities.

Starting with the IS approach for LWR without MOX the group exchanged views on different elements, such as timeliness, C/S application, and unannounced inspections. In principle there has been a positive response to the base LWR approach since it provides for a reduced number of inspections and lower the risk of instrument failure, because C/S will be used only during the reloading of the reactor. However, implementation difficulties may arise in some states regarding the unannounced inspections (UI), which are seen as a key element of the Integrated Safeguards approach in these facilities. The group recognises the attractiveness of UI for safeguards concepts. These will allow the Agency much faster access to relevant locations in a facility than any other type of inspection. They will place a potential diverter in a permanent state of uncertainty and can be used to detect and deter from undeclared activities in a facility and, thus, can be an efficient and cost effective tool to cover a range of diversion scenarios. However, UI imposes additional burden to facility operators and increases the danger that inspections may interfere with planned operational activities. Unannounced inspections are already provided for in the INFCIRC/153 type Safeguards Agreements. They are also used in the concept of 'Limited Frequency Unannounced Access' (LFUA). This concept was developed for the access to cascade halls in gas centrifuge enrichment plants and has now been successfully applied for many years in the URENCO enrichment plants in the EU. A presentation from a group member working within the URENCO Company outlined the details of application and the experiences gained with UI in the LFUA safeguards schema.

The issues were, together with the feedback collected from reactor operators, summarised in a detailed list of questions that has been presented and handed over to the IAEA. This list has served as a contribution from the group to support the IAEA in defining the conditions for unannounced inspections. The results of our discussions are summarized in a topical paper worked out by the group (see reference 4).

3.3. AP Implementation

In their own organisations, many of the WG members have been involved in practical work to prepare for the implementation of the AP in their countries and this common interest directed our activities, as well. Within the EU Member States, there exist three different tripartite Additional Protocols concluded by the states and the European Commission with the IAEA: the two nuclear weapon states (NWS) each have their own type of AP and the remaining non-nuclear weapon states (NNWS) have a common AP. For some of the new EU member states that have not yet acceeded to the tripartite EU safeguards agreement with the IAEA, existing bilateral agreements are still in force.

The AP for the European NNWS contains a third annex that foresees the possibility that states entrust EURATOM with tasks assigned to the state in the Additional Protocol. A considerable number of the EU member states make use of this opportunity and therefore EURATOM has to play a main role in the implementation of the AP in these countries, as well.

A task requiring considerable effort in all countries was to work out the initial declaration according Article 2 of the AP. The UK had already submitted a voluntary Article 2 declaration since 1999 and had gained a lot of experiences for several years. They provided the group with very valuable information on the procedures chosen, the effort required, the problems encountered, and the results achieved. Although the Art. 2 declaration of the UK as a nuclear weapon state is special in nature, it clearly identified the areas where difficult problems are to be expected. Additional stimulations for our in-depth discussions of AP implementation issues came from briefings given by the inspectorates, from presentations made to group about field tests conducted in Finland at the VTT and in the Netherlands at Petten.

3.3.1. Site Definition

One of the most important practical issues when preparing for the implementation of the Additional Protocol is the definition of the site under the Article 2a(iii) declaration. The WG discussed this issue in length since there are many aspects to be considered. The site should be carefully defined as it has a direct influence on the amount of information to be provided to the IAEA, the Agency's rights for Complementary Access and the building owner obligations to grant this access in often at very short notice. Sites should be large enough to meet the objectives of the Additional Protocol. They should also be set in such a way that they do not include buildings not contributing to the nuclear mission of the site so that no information is collected that is useless for the IAEA. The site boundaries should, therefore, be set in such way that no unnecessary burden is put on building operators that have no functional relationship with nearby nuclear activities. This topic was one of the central points of our discussions during the last 3 years. The problems considered and the solutions found are discussed in detail in a topical paper worked out by group members (see reference 3).

3.3.2. R&D Declaration

Beyond any doubt, declaration of R&D activities is one of the most sensitive aspects of the AP. R&D declarations were treated intensivly by the Working Group and it is very likely that it will take some time to reach a universally accepted interpretation of the inferred obligations. A clear understanding of a State's commitments and the Agency's rights in relation to the declaration of R&D activities not involving nuclear materials are key elements to adequately address the requirements of the AP. Misinterpretation of the AP obligations could lead either to excessive burden for States and operators or to incomplete declarations, which may weaken the objectives of the AP and deserve further inquiry from the Agency. While the latter is true for most AP provisions, it is particularly relevant to R&D activities for several reasons, which have been discussed in depth in our group:

- The wording of the AP requirements is open to interpretation, mainly due to the broad concepts embedded in the definition of nuclear fuel cycle-related R&D activities;
- The role of the State needs to be discussed, in order to understand when it has a genuine control or knowledge of the specific research activities carried out by the involved players; and
- The declaration should focus on added-value information for the Agency's strengthened safeguards mission. Excessive R&D declarations would be an unnecessary burden for States, operators, and the Agency.

The critical points of the R&D declaration as identified in our discussions and the solutions contemplated are in detail described in a topical paper worked out by group members (see reference 6).

3.4. Other Topics and Activities

Besides the topics mentioned above, there were quite a lot of other issues touched in our meetings. It soon became a sort of a ritual to start the meeting with a “tour de table” where members from each country represented in our group and the inspectorates gave a status update of current developments, activities, and issues encountered in the process of preparing for the AP implementation. Within the framework of Integrated Safeguards, we discussed other subjects that not yet demanded our attention with the same urgency as the preparation of the initial AP declaration. Among these subjects were the IS evaluation criteria, the role of C/S in IS and the procedures for the resolution of anomalies. But among the subjects we felt it necessary to spend more attention on were the analysis of IS boundary conditions and the contribution to the revision of the IAEA AP guidelines.

3.4.1. Discussion of Boundary Limits

The WG also worked on the analysis of the Integrated Safeguards overall framework and its consequences. The Integrated Safeguards Conceptual Framework was established taking into account a bottom-up approach, starting from facilities, and with a number of self-imposed boundary conditions such as one PIV per year, recoverability of classical criteria within existing timeliness goal, no change in safeguards levels on Unirradiated Direct Use material, and non-discrimination between states exclusively based upon quantitative parameters. Looking back at what was expected from Integrated Safeguards and considering the current framework and the proposed approach, the design logic looks closer to combination than integration, and the above mentioned boundary conditions clearly put limits on what can be reasonably achieved. Consequently, it puts a strong limitation on savings that can be expected compared to “classical” safeguards verification measures. It constrains the value expected from a greater use of Additional Protocol measures and at the same time undervalues the synergy of Additional Protocol and “classical” verification of nuclear materials. The WG discussed some of those boundary limits and suggestions to reach a better balance between quantitative and qualitative tools and provided examples where real integration could be implemented and could bring more benefits than juxtaposition. The ultimate goal, in our view, is to grant Integrated Safeguards the strength of a real flexibility to enhance effectiveness and efficiency.

The considerations and suggestions of the Working Group are summarized in a topical paper prepared by group members (see reference 5).

3.4.2. Input for the IAEA AP Guidelines Seminar

In October 2003, the IAEA organised a seminar in London to discuss the proposed revised text for the AP guidelines. The Working Group contributed to the revision of the original text that was issued in 1997. One of the characteristics of this group is that problems are encountered under very different conditions in the different countries, thereby covering a wide spectrum of possible circumstances due to the different conditions existing in each country. The IAEA was present and participated actively in our discussions of AP issues und thus got a profound understanding of the problems. The group members produced a large effort to document the discussions and to prepare topical papers on all relevant issues. These documents were also handed over to the IAEA for the revision of the guidelines. We were very glad to notice that many of our views and suggestions have been adopted in the revised version of the guidelines.

4. Future Activities

According to some information received during the IAEA regional technical meeting on Additional Protocol implementation in August this year, the IAEA intends to draw the broder conclusion on the absense of undeclared nuclear material and activities for some of the EU states with few nuclear activities until the end of this year. The vast majority of the EU states will follow within 2 years. This broader conclusion is a prerequisite to implement Integrated Safeguards.

Although the IAEA will design the IS approach for a state by itself, there will remain a lot of details to discuss for the practical implementation. In our next meetings we will analyse such issues like the practical cirumstances on if and how unannounced inspections could be implemented in our different member states, what the procedures are to implement mailbox systems, etc. Here we can also take advantage of the experiences some of our members made alredy with IS.

In a long term step, the state based IS approaches in the EU may be complemented by an IS approach considering the European Community as a whole. As the nuclear industy in the EU states is heaviliy interwoven,and there exist practcally no internal boarders for the exchange of goods in the EU, as major nuclear players in the EU are running their facilities in several countries of the Union, and as the respective services of the European Commission act as the common system of accountancy and control, there are many aspects that call for a Community wide and not just state wide consideration. This will also be a field of our future activities.

5. Conclusion

From our Point of view, the Working Group on Integrated Safeguards has very well met the expectations that called for the setting-up of the group. The group has proved to be very active and productive and makes the results of its work available to the safeguards community. A key output is the intense information exchange between the group members that also leads to the emergence of an harmonised view on key issues related to the implementation of the AP and the development of IS. The relationship developed with the IAEA allows a very open discussion and thus a good mutual understanding. It has always been our endeavour to find harmonized solutions that take into account the view of all parties involved in the implementation process, such as operators, national, regional, and international inspectorates.

With the Additional Protocol now in force in the European countries, a milestone of our work has been accomplished, but our task is not at all completed. The need for an intensive information exchange continues with the preparation for the practical implementation if IS in our countries. All this belongs to the necessary groundwork on which the development of the Integrated Safeguards approaches can be based and further developed or complemented in future.

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The development and the implementation of integrated safeguards approaches for LWRs in Japan

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Abstract. Due to the large nuclear fuel cycle in Japan, the IAEA and Japan decided in 2001 to start preparation of facility-type specific integrated safeguards (IS) approaches. This decision was taken after the completion of the conceptual framework for IS by the IAEA. The process began with the development of IS approaches for LWRs because there were more than 50 LWRs in Japan.

After completion of a draft IS approach and associated procedures for LWRs without MOX, the IAEA and Japan carried out field trial tests of them. The main purpose was to test how to carry out random interim inspections. It was a big challenge for all parties. The trials were comprised of three phases and at the last phase, the IAEA selected the facilities along with the procedures. With co-operation from the facility operators, the trials were successfully completed in 2003 and Japan was ready for the implementation of IS in LWRs without MOX.

In June 2004, the broader conclusion for Japan was drawn and IS measures for LWRs without MOX and a few other facility-specific IS approaches were implemented beginning 15 September 2004. IS approaches for certain facilities or sites are still being developed. Even so, the IS implementation in Japan's LWRs was an important beginning. After a transition period, routine implementation started on 1 January 2005 and 2005 is the first year for the IAEA and the Government of Japan to assess the efficiency and the effectiveness of IS implementation in LWRs without MOX. We hope that these approaches and associated benefits will encourage other States to conclude additional protocols and to contribute to effective IAEA safeguards implementation and the strengthened non-proliferation regime.

1. Introduction

Since ratification of the Additional Protocol (AP) to Japan's Safeguards Agreement in December 1999, the complete set of strengthened safeguards measures have been applied in Japan.

IAEA completed the development of the conceptual framework of integrated safeguards (IS) in 2001. IS is defined as the optimum combination of all safeguards measures available to IAEA under comprehensive safeguards agreements and additional protocols to achieve maximum effectiveness and efficiency in meeting the IAEA safeguards obligations within available resources. IS is implemented in a State only when the broader conclusion is drawn.[1]

Due to the large nuclear fuel cycle in Japan, the IAEA and Japan decided in 2001 to start preparation of the IS approaches for light water reactors (LWRs), spent fuel storages (SFSs) and research reactors and critical assemblies (RRCAs). The IS approach for LWRs was especially important to Japan and the IAEA, because there are more than 50 LWRs in Japan and they provided a practical example to demonstrate the effectiveness and efficiency of the IS approaches.

2. Establishment of the working group of the development of IS approach for LWRs

In 2001, there were only two Japsnese LWRs which had fresh MOX fuel stores. Most of the LWRs in Japan were categorized as LWRs without MOX. Therefore, IAEA and Japan started discussions about the IS approach for LWRs without MOX based on the model IS approach for LWRs. Japan wanted to carry out field trials, called "field rehearsals", to aid the smooth transition from traditional safeguards to integrated safeguards. The rehearsals were beneficial for both the IAEA and Japan (Japanese Safeguards Office and facility operators) to test and demonstrate the effectiveness and efficiency of the IS approach and to prepare their own implementation framework. The Government of Japan provided an extra-budgetary fund to the IAEA to support the IAEA's participation in the rehearsals.

The 22nd Japan-IAEA Joint Committee Meeting (JCM) in 2001 agreed to the creation of a new working group, to deal with the development of IS approach for LWRs. Representatives of Japan’s Safeguards Office (JSGO) visited IAEA headquarters in January 2002 to discuss the first draft of an IS approach for LWRs without MOX prepared by Japan based on the model IS approach.

3. Objectives of IS approach for LWRs without MOX

The main difference between the IS approach for LWRs without MOX and the traditional safeguards approach is the introduction of random interim inspections. Under traditional safeguards, interim inspections are scheduled by the IAEA, JSGO and facility operators and carried out every three months in order to meet the timeliness requirement. Under integrated safeguards, the scheduled interim inspections are replaced by fewer random interim inspections (RIIs). The RIIs are triggered by IAEA alone, and is not predictable for JSGO and facility operators. Taking into account the unpredictability, the objectives of RIIs are as follows:

- contribution to the detection of diversion of spent fuel, including cases where diversion could be concealed by "borrowing" across facilities;
- contribution to the detection of diversion of fresh LEU fuel and the presence of undeclared nuclear material;
- contribution to the detection of tampering with C/S systems;
- contribution to the detection of diversion by pin removal and the undeclared production of plutonium; and
- confirmation of the operator's statement about continuous reactor operation without access to the core since the last PIV inspection.

Unannounced RIIs are preferable to the IAEA because it maintains the unpredictability. Taking into account the recent tightening of the security at nuclear facilities, short notification periods seem more practicable and reasonable. Shorter notification times may require further co-operation from States and facility operators in the implementation of RIIs. Successful implementation may require extensive consultations.

RIIs are performed with a 20% selection probability per reactor per year. This means that a relatively high number of facilities are not selected for an RII(s). The agreed inspection procedures are very important for continuity of knowledge relevant to RIIs. Therefore, IAEA, JSGO and facility operators worked to make the agreed IS approach and its implementation procedures consistent with the full and successful implementation of RIIs. They should be updated as needed.

4. Important elements of the IS approach for LWRs without MOX

IAEA and Japan started the discussion based on the IS model approach for LWRs and they focused on how to carry out random interim inspections. The model IS approach for LWRs without MOX contains three options for meeting intensive inspection requirements. According to the cost analysis and other implementation conditions, IAEA and Japan agreed to select option 3, announced RIIs supplemented by surveillance operated in an overwrite mode.

4.1. Safeguards Parameters under IS

The intensity of safeguards verification is controlled by four implementation parameters; significant quantities, timely detection goals, probability of detection and defect test levels. A reduction in verification intensity for some nuclear material types under IS is accomplished by an extension of the timely detection goal for irradiated direct-use material from three months to one year and some reduction in probabilities of detection. Other parameters are unchanged. Nuclear material accountancy remains a safeguards measure of fundamental importance and annual material balance evaluations supported by annual verified closures of material accounts and an annual safeguards statement will continue under integrated safeguards.

4.2. Provision of Advanced Information from Facility Operators

For increased transparency and improved IAEA planning, the facility operators provide more information about their operational schedule, safety inspection schedule, activities involving crane operations and so on. Some of them had already been submitted to the IAEA under traditional safeguards. The new items were requested by IAEA formally, pursuant to the Articles 2.a(ii) and 3.f of the additional protocol. Now facility operators provide a 15 month schedule in an agreed format, with the following;

- the operational programme,
- the schedule of shipments to foreign countries,
- the receipt of fresh fuel,
- the maintenance of crane and fuel handling machines,
- the movement of empty fresh fuel/spent fuel casks,
- the cask loading activities,
- the transportation of radioactive waste, and
- other relevant activities.

This schedule is updated every three months and is sent to IAEA through JSGO. Additionally, the respective company holidays, normal working hours, scheduled power outages and other schedule related information is updated to the IAEA as needed.

4.3. Notification Procedures

Notification procedures for RIIs is an important point for the successful implementation. The most controversial point was how and when the notification is transmitted to Japan. According to the model IS approach, option three allows an advance notification from 24 hours to one week.

Some activities carried out during RIIs need facility support, such as crane operations. Certain levels of crane operation capability are necessary for this operation in Japan. Japanese LWRs facility operators share qualified crane operators and their schedules are fixed at the beginning of a year. It is almost impossible to re-schedule qualified crane operators with short advance notification. Even one week is not enough for re-scheduling and the re-scheduling can result in unplanned costs. Based on understanding of the importance of short notice and quick access to strategic points, both IAEA and Japan agreed on 24 hour advance notification. Facility operators did their best to facilitate the smooth implementation of RIIs. They encouraged their own staff to keep the bridge placed over the spent fuel ponds, to get the necessary operation licenses and generally tried to reduce the constraints on IAEA's inspection activities. Even so, some constraints remained. Japan proposed that they summarize facilities' difficulties during RIIs with short notification that the summaries be included in the IS approach and procedures and it was agreed.

IAEA prepared the form for advance notification, including the additional NDA and C/S support and Japan set the notification window between 09:30 and 10:00 am (Japanese Standard Time, JST).

4.4. Support for the inspection activities during RIIs

On the day of an RII, IAEA and Japanese inspectors meet at 09:30 at the selected facility and the facility operator prepares for the access to the LWR, including health and safety services. IAEA and Japanese inspectors start their inspection work at 10:00 am (JST). In general, IAEA starts with a review of the overwrite mode surveillance during the period from the time of the notification to access to the strategic points. The result of the surveillance affects the following inspection activities. If the result is conclusive, IAEA will continue with the planned RII activities. If the result is inconclusive, the RII is stopped and IAEA will proceed to follow-up actions.

The facility operators prepare the following documents for a RII;

- the general ledger,
- the inventory list as of the day before the RII,
- the integrated thermal power diagram until the end of the proceeding month,
- the NIS (APRM) and S/W flow charts for the most recent available dates,
- the fuel location maps as of the day before the RII (could be updated manually for the RII),
- the list and maps of fuel assemblies in the casks/containers, if applicable, and
- relevant source documents.

The inspection activities consist of;

- a book audit,
- verification of the reactor core,
- verification of spent fuel assemblies,
- verification of fresh fuel assemblies,
- visual observation of the spent fuel pond, and
- environmental sampling.

The inspection activities themselves are not different from those under traditional safeguards.

The introduction of RIIs required that the IAEA modify some inspection support software packages and in-field equipment. One of the important developments was the "in-situ" surveillance data review using GARS software. As mentioned above, IAEA reviews the surveillance data first. It covers almost 24 hours, from the time of the notification sent to JSGO to the time of IAEA inspector's access to the strategic points. It takes approximately 5 to 10 minutes to review the surveillance record.

IAEA amended the computer-based inspection reporting system, because the timing of interim inspections is changed under IS. IAEA developed new software to select facilities for RIIs at random.

During the rehearsals, IAEA also prepared special RIIs suitcase which contains all necessary inspection equipment and documents. The IAEA can not send inspection equipment in advance of the inspection under IS.

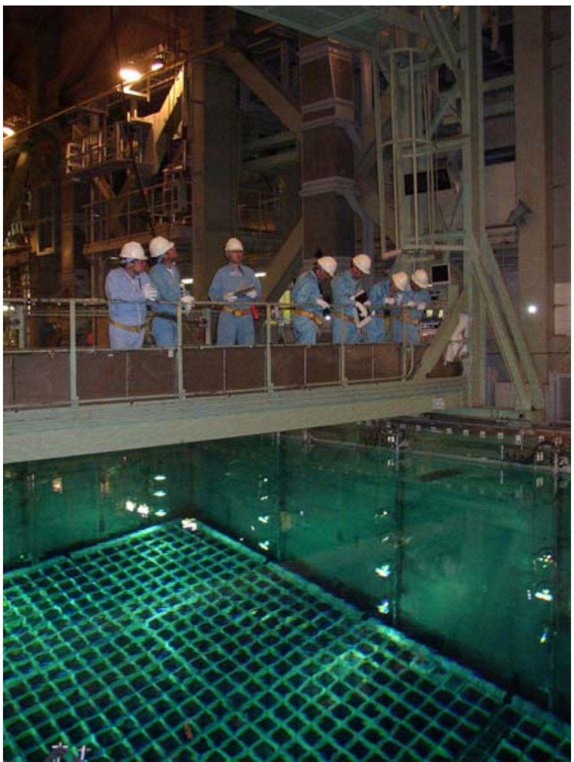
5. Field Rehearsals and their conclusions

There were 3 phases of field rehearsals from 2003-03-26 to 2003-11-30. Two facilities (one PWR and one BWR) were selected for phase 1 and 2 rehearsals. The phase 1 rehearsals were planned and conducted jointly by IAEA and Japan (JSGO and facility operators) on 2003-03-26 and 2003-03-28. The main purposes of the phase 1 rehearsal were to check the draft procedures and the necessary new inspection related equipment and to give a chance for all relevant facility operators to participate in the RIIs.

After the successful completion of the phase 1 rehearsal, phase 2 was carried out. A three month period (from 2003-03-31 to 2003-06-30) was set for the phase 2. During this period IAEA triggered RIIs for two facilities with 24-hour advance notification but JSGO and facility operators did not know the date. It was closer to the actual conditions for RIIs. There were 2 RIIs at each facility. IAEA and JSGO concluded that the procedures were suitable for practical use in Japan.

The phase 1 and 2 rehearsals were good opportunities for IAEA to test their new software packages and systems for supporting RIIs. After the rehearsals IAEA worked at headquarters to complete the evaluation process of integrated safeguards activities. IAEA was able to test them and obtained good results for practical implementation.

After taking all possible measures to assure the successful implementation of the IS approach for LWRs without MOX, Japan decided to carry out the phase 3 rehearsal for all LWRs without MOX except for the two facilities involved in phase 1 and 2. Forty-seven LWRs without MOX facilities were included in phase 3. Again, a three month period (from 2003-09-01 to 2003-11-30) was set and the IAEA selected three facilities. Some situations requiring follow-up were identified. IAEA and JSGO resolved these situations and agreed on the measures for dealing with them. These experiences were reflected in the finally agreed procedures. We believe that the phase 3 rehearsals were a very important part of the process leading to the practical implementation of the IS approach for LWRs without MOX.



The overall conclusion of the rehearsals was that the agreed IS approach for LWRs without MOX was practicable. The IAEA expressed their appreciation for Japan's co-operation in the implementation of the IS approach for LWRs without MOX. [2]

6. The Development of IS approach for LWRs with MOX

The definition of LWRs with MOX is that the LWRs have a store of fresh MOX fuel assemblies. There were 2 LWRs with MOX in Japan in 2003. Fresh MOX fuel had been moved to and were being stored at the facilities but not yet loaded to the core. The inventory of MOX fuel was static at the two facilities. IAEA proposed an IS option with remote monitoring. JSGO and the facility operator considered the proposal and found practical difficulties with the application of C/S measures proposed by the IAEA. Therefore, the IAEA and Japan agreed to apply an IS approach with attended mode surveillance for these two facilities. If other facility operators, in the future, for MOX use and it becomes more common in Japan, Japan will reconsider the IS approach for LWRs with MOX and start discussions with the IAEA.

Main features of the IS approach for LWRs with MOX are as follows;

- the timeliness for fresh MOX fuel assemblies is three months. Quarterly announced/scheduled interim inspections are carried out to meet the timeliness requirement for fresh MOX fuels;

- currently installed surveillance systems will continue to operate;

- after loading all fresh MOX fuel into the core and starting reactor operations, the LWR will follow the IS approach for LWRs without MOX and surveillance is changed to the overwrite mode;

- RIIs for LWRs with MOX are performed on a random basis with a 20% selection probability per reactor per year, (at least 1 facility should be selected) in order to detect diversion of irradiated and fresh fuel and to confirm the absence of undeclared nuclear material and activities. RIIs are carried out during closed core period, and

- IAEA will reaffirm annually the conditions for continued implementation of IS at LWRs with MOX.

The PIV consists of a pre-PIV, a PIT including human surveillance or unattended surveillance for the loading of the fresh MOX fuel assemblies into the core, and a post-PIV. There is no big difference from traditional safeguards.

7. A Site Approach for LWRs without MOX

There are some LWRs' sites with multi facilities in Japan, e.g. one site includes 6 facilities. In May 2005, IAEA proposed a site approach for LWRs without MOX intended to improve cost effectiveness and efficiency of safeguards. Two possible options were proposed. They did not include a change in the selection probability under current IS approach for LWRs without MOX. JSGO and facility operators are considering the two options.

In June 2006, Japan informed IAEA that some LWRs without MOX facilities are planning to introduce MOX fuel from 2008 or 2010. The specific facilities are not yet decided. This decision affects the consideration of the site approach for LWRs without MOX. After the LWRs with MOX facilities are identified, the site approach will need to be reconsidered. Time is needed to complete the site approaches for LWRs with and without MOX.

8. Application of IS approaches in Japan

In June 2004, the broader conclusion for Japan was drawn and IAEA informed JSGO that IS measures for LWRs without MOX, spent fuel storage and RRCAs would be implemented in Japan from 15 September 2004. The State-level IS approach for Japan is not complete and some of the IS approaches for facilities or sites are still being developed. Even so, it was an important beginning for the implementation of IS in Japan.

For a smooth transition from traditional safeguards to integrated safeguards, the IAEA proposed a transition period for LWRs without MOX. During the transition period, IAEA installed an appropriate surveillance system at each facility and closed the material balance under the traditional approach and evaluation. The full IS approach for LWRs without MOX was implemented at the beginning of 2005.

Under IS, a new PIV inspection regime was also introduced to LWRs without MOX. Facilities' PIT ~~is~~ are selected for a PIV with a 20% selection probability. Therefore the total number of IAEA's inspectors visit for PIVs under IS is less than that under traditional safeguards.

In June 2006, an electric company made a presentation about its IS experiences in 2005 at the Japan-IAEA technical seminar. This seminar is held annually in order for the IAEA to explain the implementation of safeguards in Japan. The facility operator clearly stated that the number of inspections in 2005 was reduced. But they also found an increase in follow-up inspections. They concluded that more co-operation among IAEA, JSGO and facility operators was needed for improvements in inspectorates' equipment and in the effectiveness and efficiency of IS inspection activities.

IS approaches for RRCAs were developed on a facility-by-facility basis. Every facility has facility-specific features and the nuclear materials existing at facilities vary. IS approaches for RRCAs consist of several facility specific IS approaches and they have been implemented since September 2004.

There is a spent fuel storage in Japan and the IS approach was developed based on the model approach and field trials. It has been implemented without problems.

Under traditional safeguards, IAEA, JSGO and Japanese low enriched uranium (LEU) fuel fabrication facilities have implemented short notice random inspections (SNRI) since 1996. SNRIs are the standard for Japanese LEU fuel fabrication facilities. Therefore the preparation of an IS approach for LEU fuel fabrication facilities was not necessary. Taking into account Japan's large nuclear fuel cycle, IAEA and Japan agreed to implement "borrowing inspections", which are carried out among the same type of fuel handling facilities during a facility PIV. It is not the standard based on the model IS approach for DNLEU conversion and fuel fabrication facilities. Japan believes that it will increase the transparency of its nuclear activities and contribute to drawing the broader conclusion.

9. New Challenges for the development of IS approaches

IAEA, JSGO and related facility operators are working hard to develop a site approach for the JNC-1 site and the completion of the State-level IS approach for Japan. It is a big challenge for all parties to develop an IS approach for unirradiated direct use (UDU) material. The approach focuses on the interrelation between facilities and the flow of nuclear material on the same site. Therefore the JNC-1 site is divided to sectors, groups of facilities that process and store the same types of nuclear material. Japan also wants to carry out field trials for the JNC-1 site approach. The results of the trials will be reported to the IAEA as a contribution from the Japan Support Programme for the Agency Safeguards, JASPAS. We hope that this task will contribute to the completion of the State-level IS approach for Japan and the further development of integrated safeguards concepts for UDU materials.

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Implementing an integrated safeguards approach at multi-unit CANDU stations: Potential savings

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Abstract. Under traditional safeguards a large inspection effort is required to verify spent fuel transfers from on-load reactors (OLRs) to dry storage. However, under integrated safeguards considerable savings are expected from implementing a safeguards approach that relies on unannounced random inspections and the operator's provision of advance information on planned facility operation, accounting declarations and operating records. Arrangements for this approach were tested during a field trial in 2004 at a multi-unit CANDU station. The experience gained during the 2004 field trial has provided the foundation for the integrated safeguards approach for multi-unit CANDU stations. For maximum unpredictability, the State and the operator are notified of an unannounced inspection for the transfer verification only upon the arrival of IAEA inspectors at the facility. Equipment for the remote transmission of surveillance and non-destructive assay (NDA) data has been installed at all multi- and single-unit CANDU OLR sites in Canada in order to further reduce on-site inspection effort. In 2005, the IAEA began evaluation of the remotely transmitted data, including comparisons with operators' records.

The relevant safeguards measures in place during 2006 for the implementation of integrated safeguards are summarized in this paper. An analysis of the potential savings under integrated safeguards is also discussed. Under traditional safeguards. approximately 250 person days of inspection (PDI) per year have been used for verification of the transfer of spent fuel to dry storage facilities. A reduction of 150 PDI per year is expected through the implementation of the integrated safeguards approach described in this paper. Both Canada and the IAEA place high priority on implementing integrated safeguards for transfer verification.

1. Introduction

An integrated safeguards approach for a State involves the optimum combination of safeguards measures selected from all safeguards measures available to the IAEA under comprehensive safeguards agreements (CSAs) and additional protocols, and results in the most effective and efficient implementation of safeguards in the State in order to meet the safeguards objectives [1]. Canada concluded a CSA with the IAEA on 21 February 1972, [2] The additional protocol to the CSA was signed on 24 September 1998 and was ratified on 8 September 2000 [3]. Canada submitted its first additional protocol declaration on 6 March 2001.

The IAEA performs the State evaluation process by analyzing information submitted by the State pursuant to its safeguards agreement and additional protocol or submitted on a voluntary basis; information obtained, for example, through IAEA verification activities; and information from open sources and satellite imagery. Complementary access may be carried out, inter alia, to assure the absence of undeclared material and activities as recommended in the State evaluation process. Based on these verification activities carried out for Canada, in 2005 the IAEA drew the broader conclusion that all nuclear material remained in peaceful activities.

In 2001, the IAEA and the Canadian State authority (Canadian Nuclear Safety Commission, CNSC) began the preparation of the integrated safeguards approach [4] and established an Integrated Safeguards Working Group (ISWG) for formulating the details of the approach. The State-level integrated safeguards approach for Canada was finalized by the IAEA in December 2005 [5] [6]; it is divided into four sectors, as described below in section 4.1.2. The approach relies on the safeguards conclusion and thus credible assurance of the absence of undeclared nuclear material and activities in Canada, which has been derived from the State evaluation process. With this assurance, the intensity of in-field safeguards verification can be reduced. The State-level approach for Canada calls for the provision of advance information by the State regarding the facility's operational programme. This information is then used to plan and implement randomized inspections, either without advance notice or with short notice. The findings from such inspections can be propagated to the facilities or to the period from which random selection has been made. Substantial savings can be therefore expected while safeguards effectiveness is maintained [7][8].

On the basis of the State-level integrated safeguards approach, the IAEA is preparing facility-level integrated safeguards procedures, and discussions with the CNSC and the facility operator are currently in progress. The IAEA and Canada have agreed to place high priority on the early implementation of integrated safeguards at the multi-unit CANDU stations in view of future potential savings.

In this paper, the safeguards approach developed for the multi-unit CANDU stations is described. This approach is based on the experience gained from a field trial carried out in Canada in 2004 [9]. Under traditional safeguards, a large inspection effort has been allocated for the verification of spent fuel transferred from the spent fuel bays to the dry storage facilities. The safeguards approach for the multi-unit CANDU stations will utilize unannounced inspections intensively, which is expected to lead to considerable savings in inspection effort and resources. These savings will be allocated to the processing and analysis of information, within the framework of the State evaluation process, and additional protocol activities, including complementary access.

2. Nuclear power in Canada

The generation of nuclear power in Canada is based on a natural uranium fuel cycle. It includes mines and mills for the production of uranium concentrate. The uranium concentrate is sent to a refinery for processing to UO_3 [10]. UO_3 is further converted at conversion facilities to UO_2 for CANDU fuel fabrication and to UF_6 for export.

Three fuel fabrication plants produce fuel bundles to support twenty-two CANDU OLRs of which eighteen are currently operational. There are two operational CANDU-600 station sites (Gentilly II, Pt. Lepreau, single unit each) and three multi-unit CANDU station sites (Bruce, Pickering, Darlington). There are two dry spent fuel storage facilities in operation at the Bruce and Pickering sites, and a third one will be commissioned at Darlington in 2007.

3. Traditional safeguards approach

3.1. Overview

Under traditional safeguards, two zone approaches have been implemented in Canada: the natural uranium fuel cycle zone, and the Chalk River Laboratories (CRL) zone with high enriched uranium (HEU). Outside of nuclear power generation, low enriched uranium (LEU) and plutonium are used in research laboratories and miscellaneous locations; however, with the exception of the CRL the quantity involved is small.

The natural uranium zone consists of conversion facilities, fuel fabrication facilities, OLRs and their associated spent fuel dry storages. The annual simultaneous physical inventory verification (PIV) at all facilities within the zone has been an effective and efficient way of verifying the amount of nuclear material at each facility. However, spent fuel is transferred from the CANDU reactors in Bruce, Pickering, Gentilly and Pt. Lepreau outside of the zone to their associated dry spent fuel storage facilities. The verification, under traditional safeguards, of nuclear material involved in the transfer has demanded substantial resources because of the need for continuous inspector presence.

Similarly, fresh HEU is stored and used primarily in the CRL and a zone has been established for the CRL facilities. Simultaneous PIVs in the CRL facilities are coordinated for facilities handling fresh HEU and for facilities without fresh HEU, also resulting in an effective and efficient implementation of safeguards.

3.2. Multi-unit CANDU stations

Under traditional safeguards, the inspection activities at the multi-unit CANDU stations include an annual PIV as a part of the simultaneous PIV for the natural uranium zone, quarterly inspections, the monitoring of core fuel discharges by core discharge monitors (CDMs) and the subsequent discharge to the primary bay by bundle counters (BCs). In addition, the primary and secondary bays are under optical surveillance.

Spent fuel shipments to dry storage involve the verification of spent fuel bundles by non-destructive assay (NDA) prior to loading, and observation of the bundles being loaded into a dry storage container (DSC). After the DSC has been prepared for transportation, an inspector provides human surveillance of the transfer until the DSC is placed under optical surveillance in the dry storage facility. At the dry storage, the inspector performs additional gamma ray scanning of the DSC. After the operator completes the DSC welding, the inspector applies dual seals. The DSC can then be moved from the processing area (under surveillance) to the storage area (not under surveillance).

Although the zone approach has optimized the implementation of traditional safeguards in Canada, the inspection effort involved in verifying the spent fuel transfers is considerable. For spent fuel transfer verification, the loading and transfer of one DSC requires 2-3 person days of inspection (PDI). In recent years, about 100 DSCs were transferred annually, resulting in some 250 PDI for the spent fuel verification at the two multi-unit CANDU sites with operational dry storage facilities.

4. Integrated safeguards approach

4.1. State-level approach

In order to optimise the safeguards efficiency and effectiveness, the State-level approach for Canada has been developed with specific features, as described below.

4.1.1. Credible assurance of the absence of undeclared nuclear material and activities

While the traditional safeguards approach primarily addresses facility or location-specific factors, the State-level approach evaluates the State as a whole. Findings from the analysis of information, especially that provided under the additional protocol and collected through inspection and additional protocol activities (e.g. complementary access), can be used to draw a soundly-based safeguards conclusion (and thus credible assurance) of the absence of undeclared nuclear material and activities. Based on this conclusion (and assurance) and on that for the non-diversion of declared nuclear material, the broader conclusion can be drawn that all nuclear material remained in peaceful activities.

This broader conclusion was drawn for Canada in 2005. The continuous State evaluation process will continue to support the validation of this conclusion. This resulting assurance has allowed for more effective safeguards and for the relaxation of traditional safeguards parameters and procedures. The timeliness goal of irradiated direct-use material has been extended from three months to one year. For example, the conclusion that there is no undeclared reprocessing plant and no undeclared spent fuel handling capability at a site has reduced the safeguards effort required to detect the diversion of spent fuel bundles from the DSCs during transfers within the site.

4.1.2. Fuel cycle characteristics

Canadian nuclear facilities and locations can be grouped into the following four sectors:

— Sector 1 - Fuel cycle for power reactors;
— Sector 2 – Chalk River Laboratories (CRL);
— Sector 3 – Miscellaneous facilities (research reactors, static dry storages and location outside facilities (LOF)); and
— Sector 4 – Non-safeguarded mines, mills and closed down facilities (mainly facilities where most of the essential equipment has been removed, i.e. in a decommissioning phase).

The establishment of these sectors has facilitated the design of random inspections for each group of facilities or locations, resulting in a significant reduction of the overall inspection effort.

4.1.3. Provision of information by Canada: Advance information and declarations

In order to increase the efficiency of safeguards implementation, Canada agreed to provide advance information about its operational programme, accounting declarations and operating records. The timing of inspections, without notice or with short notice, can therefore be selected more effectively. The amount of information required has also been kept to the required minimum to reduce the impact on the facility and data handling by the IAEA. The detailed advance information is only provided shortly before the operation is conducted. On the basis of the advance information, the timing of an unannounced inspection is determined by random sampling. The declarations may also be simple because access to the whole set of records is possible during the short notice random inspections (SNRIs) or at the announced inspections such as the PIV.

4.1.4. Unannounced and random inspections

Since the 1980s, Canada has hosted trials for unannounced inspections and accepted unannounced inspections in the CRL; these have included the verification of on-site transfers and the confirmation of the absence of unreported production of plutonium. Under the State-level approach, Canada agreed to utilize unannounced inspections and SNRIs, defined as follows [11]:

— An unannounced inspection is an inspection performed at a facility or LOF for which no advance notice is provided by the IAEA to the State before the arrival of IAEA inspectors; and
— A SNRI is an inspection performed at a facility or LOF for which less advance notice is provided by the IAEA to the State than that provided for under paragraph 83 INFCIRC/153 (Corrected) [12].

For an unannounced inspection to be effective, the operator should grant prompt inspector access to the strategic points and accommodate inspection activities so that the IAEA can meet it safeguards objectives.

4.1.5. Advanced safeguards equipment and remote monitoring

To enhance efficiency, all operating CANDU units have had CDMs, BCs and DMOs surveillance installed, giving the IAEA the capability to remotely monitor data transmissions from this equipment. Three main elements assure the reliability, authenticity and confidentiality of the data transmitted: (a) the equipment state-of-health, (b) data authentication, and (c) data encryption. The implementation of remote monitoring at OLRs contributes to near real time monitoring and continuity of knowledge of the discharged irradiated fuel bundles from the core to the spent fuel bay (i.e., the CDMs and BCs). It is also possible to keep track of DSC movements at both the spent fuel bay area after the loading is completed and at the dry storages processing area.

4.1.6. Toronto Regional Office

When the IAEA opened its Toronto Regional Office in 1982, the verification of inter-bay transfers was its main objective. Later on, the Office was expanded to include the verification of the spent fuel transfers from the CANDU reactors to the associated dry storages. The Office offers several advantages:

— Communication is enhanced by working in the same time zone as that of the Canadian facilities and the State authority;
— IAEA inspectors can travel freely in Canada and can perform unannounced inspections or SNRIs more efficiently; and
— IAEA inspectors can respond faster to last minute changes in the facilities' operation programmes that may impact the inspection plan.

4.1.7. Information-driven planning of inspection and additional protocol activities

The State evaluation process provides a basis for planning inspections and additional protocol activities. An annual implementation plan for Canada is prepared and used for evaluating the performance of inspections and additional protocol implementation. The frequency of complementary access and the intensity of random selection are based on the analysis of this information. For planning daily safeguards operations, the information provided by Canada (advance information and declarations) and the safeguards data collected by remote monitoring are used. Timing of the randomized inspections is determined according to the updated notification on the operational programme. Inspection reports are prepared using the remote monitoring data and the facility's declaration regarding the accounting and operating records. The Toronto Office has developed a data management system (DMS) to share the results of daily evaluation among its staff.

4.2. Integrated safeguards approach at multi-unit CANDU stations

The integrated safeguards approach for multi-unit CANDU stations is characterised as follows:

— Unannounced inspections and SNRIs are used intensively to meet specific objectives. For the efficient planning of such inspections, provision of information by Canada is arranged.
— The annual PIV is performed only at randomly selected facilities. The spent fuel is physically verified less frequently than under traditional safeguards. During a PIV, design information verification (DIV) is performed in these facilities. At facilities not selected for a PIV, the DIV is performed annually with advance notice.
— The core fuel discharge is monitored by the CDM; the spent fuel discharge to the primary bay is item-counted and verified by BCs; and the primary and the secondary spent fuel bays are under optical surveillance by DMOs.
— Under integrated safeguards, the timeliness goal for irradiated direct-use material is extended from three months to one year. Quarterly inspections for timeliness are replaced by remote monitoring and SNRIs. For evaluating remote monitoring data and performing record examination in the IAEA, Canada provides monthly declarations containing accounting and operating records. The service and maintenance of installed equipment are carried out as indicated by the routine evaluation of the remote monitoring data.
— The inter-bay transfer of the spent fuel bundles, i.e. from the primary bay to the secondary bay, may require verification if the transfer is not carried out under optical surveillance.
— Verification activities may involve observation and examination of activities involving shielded containers, such as those used for Co-60 production and post irradiation tests of fuel items.
— SNRIs are performed at facilities selected on a random basis from Sector 1. During such inspections, normal routine inspection activities are carried out. Such activities may include examination of records, fresh fuel receipt verification, service of installed safeguards equipment and environmental sampling. SNRIs ensure the probability of detecting the borrowing of fuel bundles by Sector 1 facilities.

— Spent fuel transfers to the dry storage are verified on a random selection basis. Each DSC transfer involves steps for the preparation, verification, loading, transfer and sealing of the DSC. For the random selection, a selection probability is assigned to each DSC and a selection probability is assigned to each step of the transfer process. A reduction to approximately 1 PDI per transfer is expected.
— Complementary access, with the possibility of environmental sampling, provides assurance on the absence of undeclared nuclear material and activities.

4.3. Data management system

A computerised data management system has been developed at the Toronto Office to support the handling and processing of information to be received at the IAEA. The system infrastructure is designed to streamline the work process into a coherent data storage framework. The system provides the following:

— A systematic approach for handling data originating from the IAEA's remote monitoring systems (i.e., CDMs, BCs and DMOSs) and the facility supplied information (i.e., declarations on accountancy records and operational schedules pertaining to movement of nuclear materials);
— A logical framework for the review and evaluation of the operators' declarations; and
— A work process that is streamlined for efficiency and transparency in delivering the output data, i.e. the Inspection Document Package (IDP).

5. Open issues and potential improvements and savings

Facility security requirements and access times have become stricter and remain an issue for discussion. The IAEA requires that access be granted as soon as possible and that sufficient time be allowed for the IAEA to achieve its safeguards objectives. Access times will be facility specific.

At present the IAEA has not authorised the Canada-specific procedures for the electronic transmission of encrypted data. Although the method for providing information has been left flexible, the contents have been defined. Currently, some declarations are hand delivered to the Toronto Office while some advance information is sent by regular e-mail. Improvement in this area is expected.

To avoid unexpected equipment failure and to confirm the correctness of the operator's declarations, it is essential for the IAEA to develop an environment suitable for timely information evaluation and analysis. The data management system developed by the Toronto Office may be used as a prototype for such an environment.

The results of testing remote monitoring data handling and review procedures confirmed the feasibility of the data transmission as a secure and reliable practical measure to be implemented within the framework of integrated safeguards.

Savings from the implementation of integrated safeguards at multi-unit CANDU stations are expected from:

— Reduction of PIV inspections, although additional DIV visits might be scheduled;
— Cancellation of quarterly inspections and the implementation of SNRIs in facilities selected from Sector 1; and
— Reduced frequency of spent fuel transfer verifications through the utilization of randomised unannounced inspections.

Under traditional safeguards approximately 500 PDI were used in the three multi-unit CANDU stations: 100 PDI for PIV, 150 PDI for quarterly inspections and technical activities and 250 PDI for transfer verification. With the introduction of unannounced inspections for transfer verifications, a reduction of 150 PDI is expected. Further reduction is also expected from randomized PIVs and by replacing quarterly inspections with SNRIs.

Some activities included in the integrated safeguard approach for Canada are already being implemented under the additional protocol, including complementary access, DIVs, remote monitoring evaluation, and information analysis of open sources and satellite imagery. Their rational use will further contribute to the expected savings in the implementation of integrated safeguards.

Inspectors' efforts will focus on the efficient handling of information for unannounced inspection planning, documentation of remote monitoring inspections, and planning and conducting DIVs and complementary access to support the broader conclusion reached for Canada.

6. Conclusions

Integrated safeguards will provide effective and efficient safeguards in Canada, by making better use of IAEA resources without compromising its safeguards objectives. Although the number of inspection activities under integrated safeguards is defined on a probabilistic basis, a significant reduction in comparison with the inspection activities under traditional safeguards is expected. The full implementation of integrated safeguards at multi-unit CANDU stations shows possible reductions of about 30% or more on safeguard efforts measured in PDIs. The Annual Implementation Plan assures that the requirements of the State-level approach are met. The unpredictability of the inspection activities under integrated safeguards is paramount to the effective implementation of the State-level approach in Canada.

Considering that Canada's nuclear programme is the largest natural uranium fuel cycle under safeguards, the implementation of integrated safeguards in Canada constitutes a high priority for the IAEA. The implementation of integrated safeguards in Sector 1 of the Canadian facilities and sites has been given top priority, in particular the implementation of integrated safeguards at the single- and multi-unit CANDU stations. The verification of spent fuel transfers from multi-unit CANDU stations to dry storages was identified as being the safeguards activity offering the greatest potential for savings under integrated safeguards. In line with this, the following priorities have been identified:

— Finalizing the procedures under integrated safeguards for unannounced inspections for spent fuel transfers at multi-unit CANDU stations in Canada and unannounced inspections and SNRIs at multi-unit CANDU nuclear power stations in Canada (to be achieved by October – November 2006);
— Implementing particular integrated safeguards measures by the end of 2006 (UI for spent fuel transfers at multi-unit CANDU stations);
— Full implementation of the integrated safeguards approach for CANDU multi-unit stations by the first semester of 2007.

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SAR imagery satellite applications

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Abstract. The potential of SAR imagery in the context of safeguards is analysed. We first review the characteristcis of SAR images and SAR interferometry that are of interest. Sar interferometry allows to measure topography and deformation of areas of interest. The so-called coherence pattern derived from interferograms provide information about the temporal changes. The technique is illustrated from images acquired over the Nevada Test Site (USA), Dead Sea as well as from simulated high resolution X-band images.

1. Introduction

SAR imagery provides complex images that are atmospheric turbulence independent, can be acquired whether by night or day and that are usefull for monitoring of ground displacements with centimetric accuracy [1]. Moreover, in the near future several SAR satellites will provide X-band images with metric resolution.

Those characteristics are of interest in the context of safeguards and SAR imagery can provide usefull information about the evolution of sites of interest.

In the following, we first present the principle and the potential of SAR imagery and SAR interferometry that are of interest for our applications. We then propose three major axes to study sites in the context of safeguards. The first application deals with monitoring ground subsidence induced by nuclear tests from SAR interferograms. The second deals with the anomaly detection based on SAR coherence images properties. The third describes the potential of near future X-band sensors from simulated images from the onera RAMSES system, courtesy of ONERA and CNES.

2. Background

2.1. Radar

The principle and the characteristics of SAR images [2] are illustrated in (*Fig. 1*). The SAR antenna embedded on the satellite plateform is both used in emission and reception. The antenna emits radar pulses in C-, L-, S-, P-, Ka- or X-band. Large wavelengths (e.g. metric wavelength in P-band) are usefull for ground penetration; medium wavelengths (C, L) are used for decametric resolution images (ERS, ENVISAT, JERS, Radarsat, etc.), short wavelengths are used for high resolution images (TerraSAR [3], CosmoSkymed [4], SARLuppe [5]) and Ka-band is used for imagery and data transmission. The SAR antenna measures the energy and the travel time of emitted impulsions that are reflected by the ground. Nominal resolution of radar images is poor (kilometric) and is limited by diffraction. The SAR technique consists in overcoming this drawback by the synthesis of a very large antenna from the numerous echoes of a given point on the ground acquired while the plateform moves on its orbit. Resolution of SAR images in azimuth equals half the SAR antenna length (10 m for ERS and ENVISAT, few meters for TerraSAR).

SAR images are complex. The amplitude of an image depends on viewing angles, ground slope and reflectivity. The histogram of a SAR amplitude image shows a Rayleigh shape as described by Goodman [6]. The phase term is the sum of satellite to ground distance and of a random pattern that originates from the interaction of the coherent incident wave with the roughness of the pixel at the radar wavelength scale. This pattern is know as speckle. It is noteworthy to understand that the speckle is linked to diffusion characteristic of the pixel : without roughness all the incident wave would be reflected in the specular direction and the antenna would not receive anny backscattered echoes in the emission direction. The non-zero backscaterred signal that is measured by the SAR antenna is induced by the roughness of the pixel and this roughness is also responsible for the speckle pattern. If standard deviation of the roughness is significant in comparision with the radar wavelength, the speckle is called fully-developped and the phase of the backscattered echoes follows an uniform distribution in the range [-π,+π]. This is generally the case with remote sensing imagery of natural scenes. Thus, the phase of a single image can not be used to recover the satellite-to-ground distance that includes the information about the topography and the ground deformation.

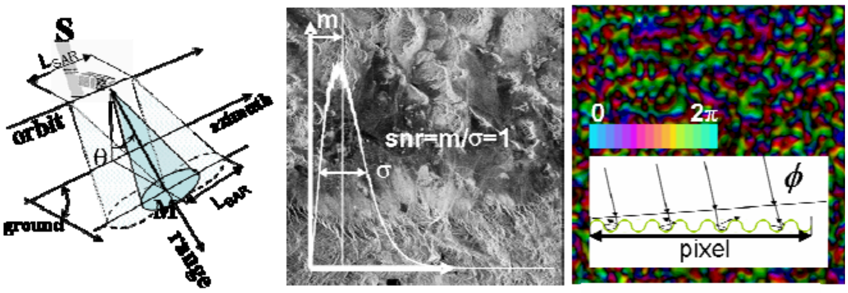

FIG. 1 Basics of SAR imagery.

Table 1. allows comparing the main characteristics of ERS and TerraSAR-X.

Sensors characteristics	TerraSar-X	ERS
Repeat cycle (orbits)	11 days (167)	35 days (501)
Orbits/day	15.2	14.3
Altitude at Equator	515 km	782 km
Inclinaison angle	97.44°	98.52°
Incidence angle	20-45°*	23°
Wavelength	3.1 cm	5.6 cm
Band with	150-300 MHz	15.55 MHz
Pixel spacing (azimuth)	1.5 m	3.9 m
Pixel spacing (range)	0.9 m	7.9 m
Swath	30 km	100 km

[*] Different stripmap beams available

SAR interferometry technique that consist in computing phase difference of images allows to overcome this drawback and is presented in the following paragraph.

2.2. SAR Interferometry

SAR interferometry consists in the phase difference $\Delta\varphi$ of SAR images (see *Fig. 2*). The phase difference at point *M* can be written as :

$$\Delta\varphi(M) \approx 4\pi\left(\frac{-B\cos(\theta+\theta_B)}{\lambda}+\frac{h}{h_a}+\frac{e}{\lambda}+noise\right), \tag{1}$$

where *B* is the distance between the two antennas called baseline, θ and θ_B are the viewing and baseline angles, λ is the wavelength, *h* and h_a are the elevation and the sensitivity to elevation and *e* is the deformation of the scene during the time span between two acquisitions. The first term in equation (1) yields fringes equivalent to those of the two wave young interferometer, the larger *B* is the smallest the distance between fringes are. The second term allows to recover Digital Elevation Models (DEM) from SAR interferometry (e.g. the SRTM mission), h_a equals 0 for *B* equals 0 (no sensitivity to topography) and is equal to 100 m for *B* equal to 100 m in the case of ENVISAT C band ASAR. Above a critical value B_c of *B* (B_c equals 1000 m for ERS) the bandwidths of the two images do not overlap and the phase only includes noise. The third term provides the deformation *e* of the ground at the wavelength scale and allows to measure subtle deformation of the surface [7], [8], [9], the process yieldind *e* is called differential interferometry. The latest term describes the noise and is of major importance for this study. Noise mainly originates from the difference in viewing angles (with a maximum *B* beyond which the interferogram can not be exploited), the difference in Doppler centroïds of the images and the change in apparent roughness of the ground (temporal decorrelation).

The radar beam has a slight interaction with the atmosphere as the atmospheric phase delay $\Delta\varphi_{atmosphere}$ can be written as :

$$\Delta\varphi_{atmosphere}(M) \approx \int_{satellite-ground} k_1\frac{P}{T}+k_2\frac{P_e}{T}+k_3\frac{P_e}{T^2}+k_3W_{cloud}+k_4W_{rain}+k_5\frac{TEC}{f^2}, \tag{2}$$

where *T* is the temperature, *P* and P_e the total and wet partial pressure, W_{cloud} and W_{rain} terms depending on clouds and rain, *TEC* is the electronic total content of the ionosphere and *f* the radar frequency.

$\Delta\varphi_{atmosphere}$ can contribute to several fringes per image and yields a signal that is partially correlated with topography. Several studies [10], [11] propose atmospheric filters. Though the atmosphere does not prevent high gradient deformation pattern analyse (e.g. earthquake co-seismic deformation) it may be a limitation to derive DEM from images acquired at different moment and to monitor subtle, low amplitude and large scale deformation.

The coherence function *C* describes the local changes in the speckle pattern. *C* is in the range[0,1]. It can be estimated as :

$$C(M)=\frac{E(ab^*)}{\sqrt{E(aa^*)E(bb^*)}}=C_{temporal}(M).\frac{B-B_c}{B_c}.\frac{\Delta f_{doppler}}{PRF}, \tag{3}$$

where *E* denotes mathematical expectancy and is generally computed from local averaging of interferogram, B_c is the critical baseline, *PRF* is the bandwidth along the orbit (azimuth) and

$\Delta f_{doppler}$ is the difference in Doppler centroïds of the images. $C_{temporal}$ denotes temporal changes of the scene and is of interest to detect some kind of activity of interest (see § 4). If the ground has not changed between the acquisition of the image, $C_{temporal}$ is close to 1. $C_{temporal}$ close to 0 is the case of major change in the roughness of the pixel [12]. Changes in roughness may be due to vegetation, crops, erosion, building, water content, etc.

Due to the near fractal change rate of roughness with scale, the shorter the wavelength, the higher the decorrelation rate.

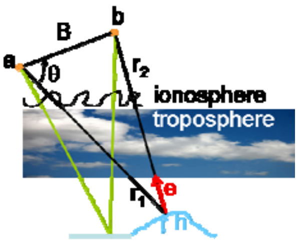

FIG. 2. Basics of SAR interferometry.

3. Nevada test site analysis

We used ERS SAR images to measure subsidence at the Nevada Test Site and get the results presented in *(Fig. 3)*. These two images were the same as presented in [9].

The interferogram shows a deformation pattern which geometry is correlated with the location of the major tests and that continues to evolve with time. The intereferogram includes some atmospheric artifacts that are however about ten time smaller in amplitude than the subsidence signal.

Profile on the deformation pattern shows the shape and the amplitude of local deformation of the studied area.

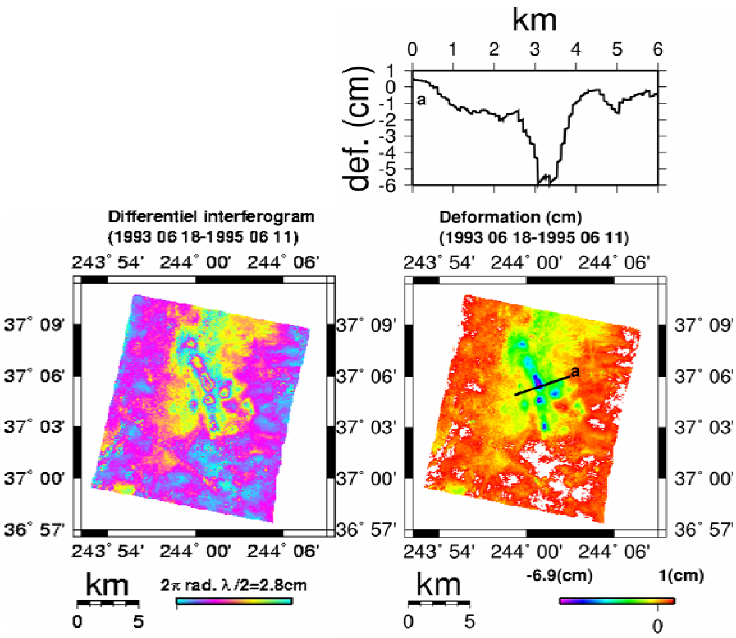

FIG. 3. Differential SAR interferometry for the Nevada Test Site.

4. Anomalic activities detection

4.1. Dead sea

(Fig. 4) is an illustration of the potential of coherence to detect changes. Actually, industrial devices appear on the coherence map *(Fig. 4.a)* and are not identified on the amplitude image *(Fig. 4.b)*. We can notice also on the zoom area, that the hydrographic structures are clearly identified on the coherence map. This kind of precision could be interesting in the context of safeguards.

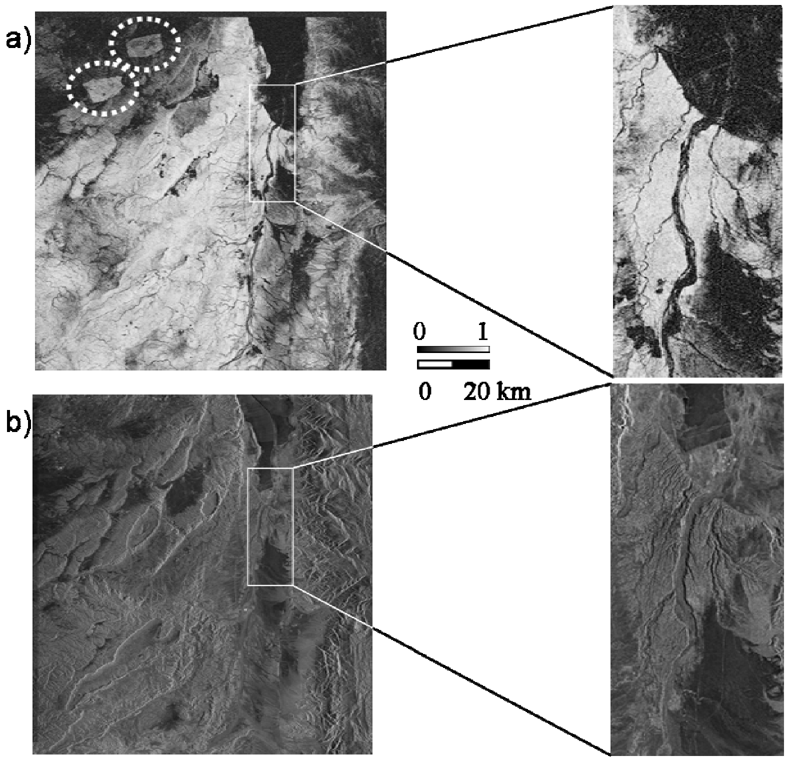

FIG. 4. Anomalies detection (devices and hydrographic …) on the Dead sea.

4.2. Anomalic large field detection

A procedure that allows to analyse anomalies from a set of SAR images acquired at different dates is presented (*Fig. 5*).

Temporal decorrelation *C* is derived from interferograms. Rates of temporal decorrelation are constrained from measurements, physical models (erosion, vegetation, etc.) and predictive filter and we get an estimate of what should be the coherence images derived from a new image of interest. The comparison between measured and estimated temporal decorrelation maps is the anomaly map.

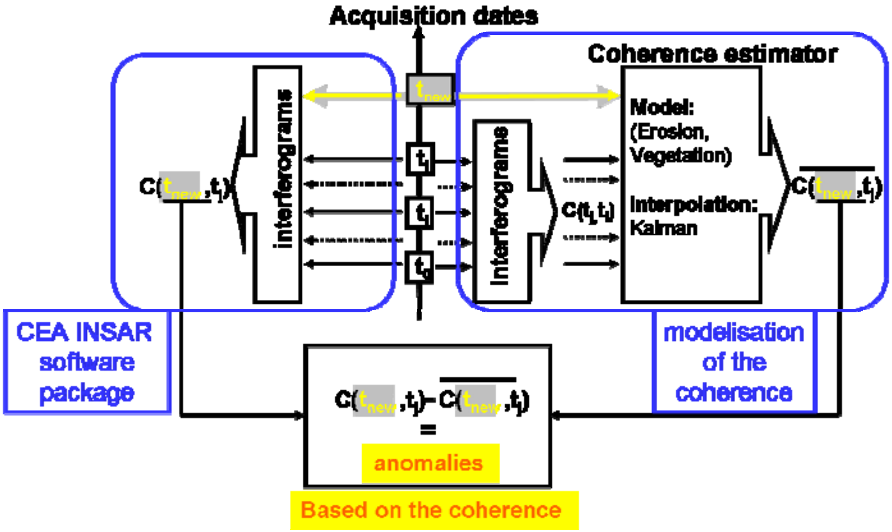

FIG. 5. Principle of anomaly detection from coherence images.

5. Potential of near Future X-band Satellites.

Several satellites will be launched in the near future and will include onboard X-band SAR antennas. The potential of those images in the context of sageguards is illustrated on *(Fig. 6)*. First, resolution of the image is metric and this is particularily of interest for the analysis of man made devices. Second, the phase difference will provide enough accurate topography measurements to make building height estimation. Third, coherence and phase images will be of interest for the detection of industrial changes. Those systems will also benefit from a smaller revisit time period (11 days for TerraSAR, 35 days for ERS). The major drawback of these systems is the lack of coherence because decorrelation rate may be important in X-band, especially over crops and natural vegetation. Another drawback is that these images will require more skills from the operator, mainly because the geometry of images is complex, including overlayed and shadowed areas. This figure illustrates also the potential of this kind of SAR satellite to recognize activities of a boat. Analysing the zoom image indicates that this boat is a frigate with Anti Submarine War capabilities.

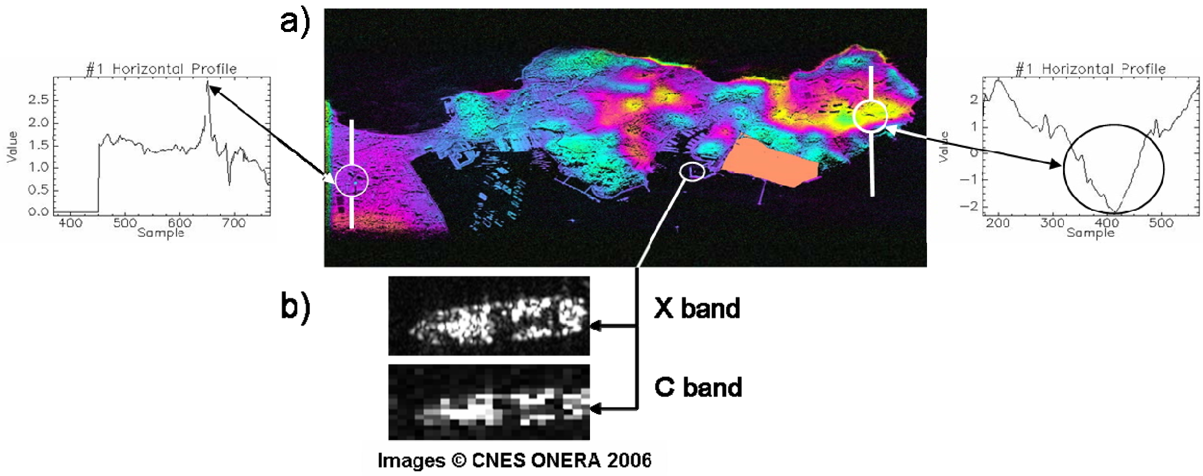

FIG. 6. a) corresponds to the unwrap interferogram, and b) present a zoom corresponding to a boat.

6. Conclusions

We have illustrated the potential of SAR imagery for the analysis of scenes of interest in the context of safeguards. SAR interferometry allows to estimate topography and deformation induced by nuclear tests. The coherence fonction provides usefull information about the activity and evolution of sites. Near future SAR systems operating at X band will open a new area for the exploitation of SAR images mainly because of the availability of images metric in resolution. Those images will also require to developp more skill and associate tools for the operators.

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Object-based analysis of hyperspectral and thermal infrared satellite imagery

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Abstract. The given paper proposes an object-based procedure for the combined analysis of high-resolution optical, thermal infrared and hyperspectral satellite imagery for different nuclear safeguards-related tasks. Some case studies using Hyperion, Landsat,QuickBird and Ikonos data will demonstrate the advantages of this approach.

1. INTRODUCTION

The use of satellite imagery for nuclear safeguards applications today is very much limited to visible, near-infrared data, due to at least two reasons: First, from a technical point of view, optical data provides the best spatial resolution in the sub-meter range for the monitoring of small-scale nuclear-related activities. Second, from the user's point, the (visual) interpretation of visible and near-infrared data is more obvious rather than analysing thermal infrared, hyperspectral or radar image data where extensive pre-processing and knowledge on the sensor is required. However, also satellite data from thermal infrared, hyperspectral and microwave sensors involve useful or even relevant information for nuclear monitoring. The given paper aims to demonstrate the potential of hyperspectral and thermal infrared satellite data for nuclear safeguards-related tasks from an image processing perspective. As safeguards applications, monitoring of uranium mining and milling and power reactors is considered.

Under Integrated Safeguards all States having signed the additional protocol are obliged to declare the whole states production of uranium. Information on the production of individual mines and concentration plants has to be provided on request to the IAEA. In order to be able to verify the States declaration and to guarantee the absence of undeclared mining and/or milling activities, satellite imagery data are also taken into consideration. Safeguards-related information identifiable by satellite imagery are signs of water, power and chemical usage, level of mining respectively milling activities, geographic extent of mining activities and the presence of other industrial activities associated with mining or milling respectively (Leslie et al., 2002). The question whether hyperspectral data may be used to determine the surface mineralogy of exposed uranium tailings, has been controversially discussed recently (Lévesque et al. 2001, Neville et al. 2001). According to those results, analysing solely hyperspectral image data will not lead to the identification of uranium activities or to the differentiation of uranium mining activities from other types of mining respectively. Nevertheless, hyperspectral information could support the interpretation of high-resolution optical imagery, and vice versa. An object-based procedure utilising both the hyperspectral information and the shape and topology features from the high-resolution imagery was developed. Methodology and application will be presented.

As different materials (soils, plants, water, man-made materials) selectively absorb short-wave solar energy and radiate the long-wave (thermal) energy in a specific way, it is possible to determine the type of material based on the thermal emission characteristics of the material - in case the atmospheric conditions and other influencing factors are known. Moreover, thermal data could be used to evaluate whether significant changes have taken place in the thermal characteristics of these materials over time. Thus, the use of thermal infrared imagery for the monitoring of (heat generating) nuclear activities seems to be reasonable, even though the spatial resolution of satellite-based thermal infrared sensor bands is still limited to 60m (LANDSAT 7) and 90m (ASTER) respectively. The image data given by the thermal infrared system enables the user to analyse thermal differences between the area of interest and its neighbourhood and thus to derive information on the operational status of the facility. Since environmental influences may overlay or modify the natural or artificial thermal radiation or even result in thermal anomalies, also thermal image data has to be analysed in combination with other data. High-resolution satellite imagery for instance provides relevant information on the land cover that could be utilised for the interpretation of thermal infrared data. An object-based approach for analysing high-resolution optical and thermal infrared image data was realised. Again, methodology and application will be presented.

2. DATA

2.1. Hyperspectral image data: Hyperion

Hyperion, the only satellite-based hyperspectral sensor today, is part of the Earth Orbiter – 1 (EO-1) Mission, launched November 21, 2000 together with two other revolutionary land imaging instruments on EO-1, the Advanced Land Imager (ALI) and Atmospheric Corrector (AC).

Hyperion provides a high resolution hyperspectral imager capable of resolving 242 spectral bands in 10-nm (average) sampling intervals over the contiguous reflected spectrum from 0.4 to 2.5 µm with a 30-meter resolution. The data product distributed by the USGS EROS Data Center (EDC) consists of 198 calibrated channels with radiometrically corrected (Level 1R) data. No geometric correction is applied, thus the data is not georeferenced. The instrument images a 7.5 km swath width in the cross-track direction, with a variable along-track swath width, either 42 km for a standard scene or 185 km for an extended scene.

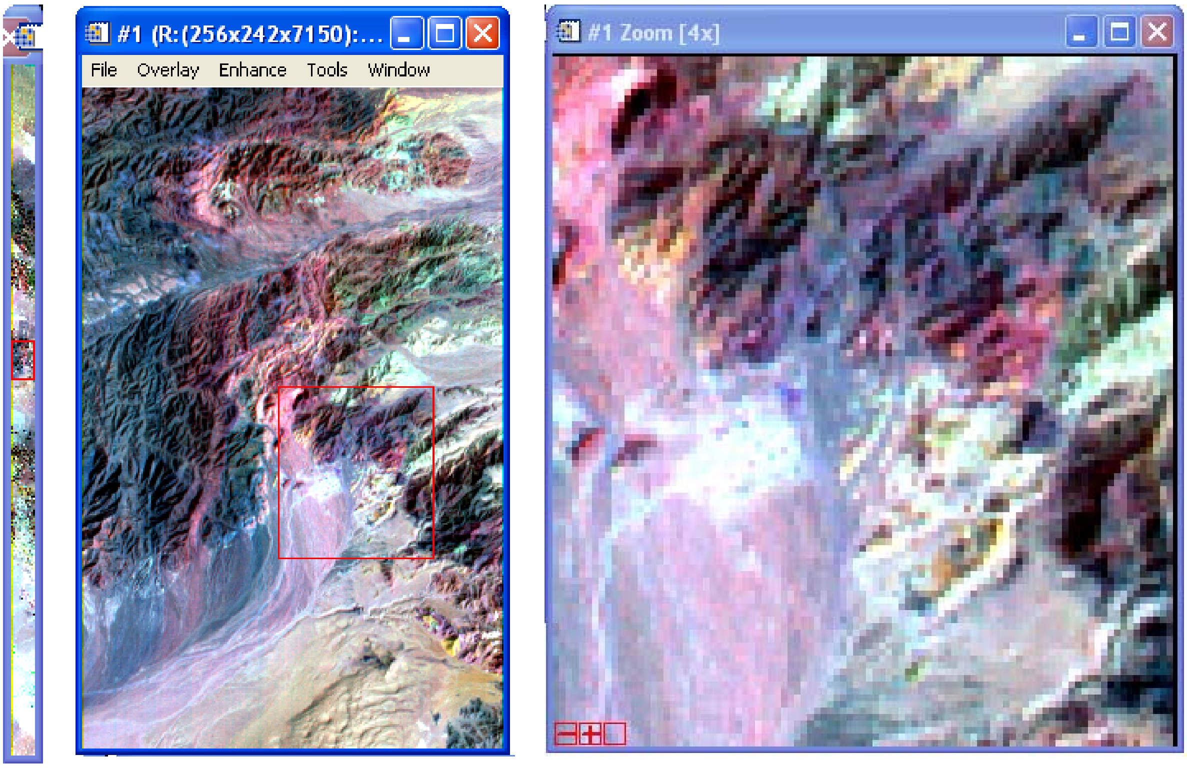

Figure 1. Hyperion scene, Saghand Uranium Mine, September 24, 2003.

2.2. Thermal infrared data: LANDSAT and ASTER

At the present time, only the Landsat and Aster (Advanced Spaceborne Thermal Emission and Reflection Radiometer) satellites offer image data from the thermal infrared region.. Landsat 7 ETM+, launched in 1999, holds one thermal channel in the range of 10.40 to 12.50 µm with a spatial resolution of 60 m and a temperature accuracy of 0.5°K. ASTER, also launched in 1999, features the thermal subsystem TIR with five channels between 8.125 and 11.65 µm at a spatial resolution of 90m and a temperature accuracy of 0.3°K (Abrams et al. 2002, Kramer 2002).

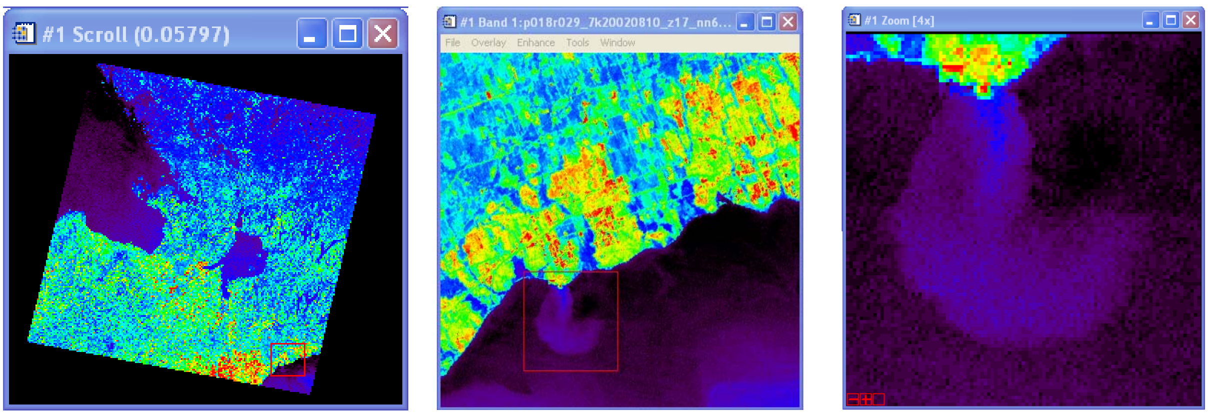

Figure 2. Thermal band of Landsat 7 scene, color mapped, Pickering, August 10, 2002.

2.3. High-resolution optical data: Ikonos and QuickBird

Ikonos, launched in September 1999 by Space Imaging (US), was the world's first commercial satellite producing data almost comparable to arial images. The sensor produces imagery in four mulitspectral bands (blue, green, red, near infrared) with a spatial resolution of 4m and a panchromatic image band in 1m resolution (Kramer 2002).

Two years later, in October 2001, QuickBird was launched by DigitalGlobe (US) and is still the only commercial satellite providing image data in submeter resolution. The spectral resolution however is comparable to Ikonos, i.e. four mulitspectral bands (blue, green, red, near infrared) with a spatial resolution of 2.44m and a panchromatic image band in 0.61m resolution (Kramer 2002).

Figure 3. Pan-sharpened QuickBird data , Saghand, August 24, 2004.

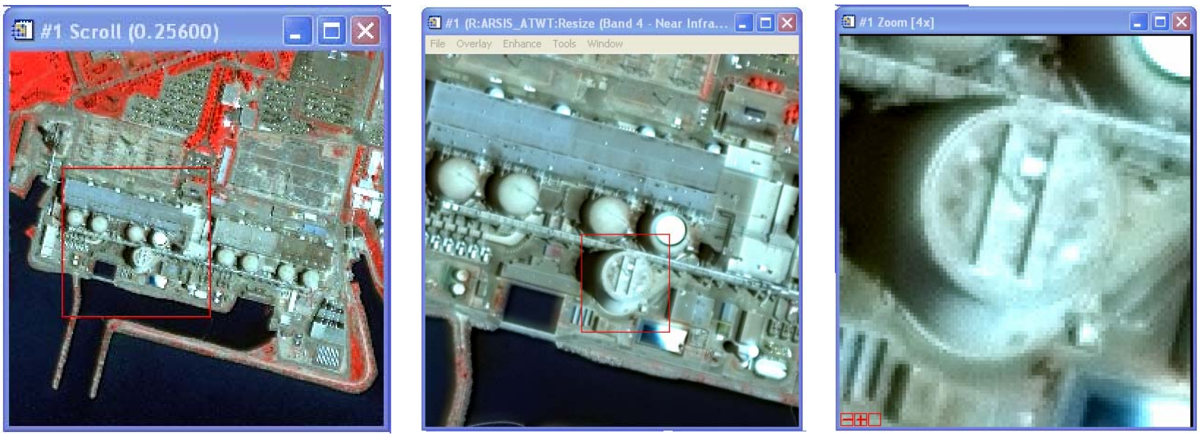

Figure 4. Pan-sharpened Ikonos data , Pickering, July 20, 2000.

3. METHODOLOGY

3.1. (Pre-)Processing of data

3.1.1. (Pre-)Processing of the hyperspectral data

- Removal of spatial and spectral artifacts (Staenz et al. 2002);
- Atmospheric corrections using FLAASH/ENVI;
- Dimension reduction using Minimum Noise Fraction Transformation (MNF);
- Endmember extraction and identification,
- Linear spectral unmixing;
- Mixture tune matched filtering (MTMF)
- Production of an abundance map of each endmember.

3.1.2. (Pre-)Processing of the thermal infrared data

- Atmospheric correction;
- Converting to Emissivity and Temperature.*(Pre-)Processing of the high-resolution data*

- Pan-sharpening by wavelet transformation (Ranchin andWald 2000);
- Calculation of NDVI

3.2. *Integrated analysis*

3.2.1. *Integrated analysis of the hyperspectral or thermal infrared and the high-resolution data*

- Image-to-image registration
- Object extraction by segmentation based on the high-resolution data
    - Using the site map;
    - By manual editing;
    - (Semi-) automatically.
- Semantic modelling using spatial and hyperspectral /thermal object features;
- Classification and interpretation.

4. DISCUSSION

Remote sensing data ranging from the visible to the microwave region of the electromagnetic spectrum could provide safeguards-relevant information. If the spatial resolution of the given spectral data is low, the integrated use of spatial high-resolution should be considered. The object-based approach offers a possibility to combine both form/topology features from spatial high-resolution data and spectral features from multispectral/hyperspectral data. The accuracy of the final results is first of all dependent on the (pre-) processing of the spectral data and the extraction of image objects based on the high-resolution data.

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Identification of key features of some nuclear facilities for interpretation of imageries from remote sensing satellites*

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Abstract. Using aerial photographs and satellite imagery, typical features of research reactors and pressurised water and boiling water power reactors were determined. It was found that, in satellite images, there are features that could be recognised and used to distinguish between research and power reactors and between a fossil fuelled power plant and a nuclear power reactor. From such data it is also possible to differentiate a civil reactor from that used for defence purposes. Moreover, it was found that from satellite images it is possible to distinguish between a pressurized water reactor (PWR) and a boiling water reactor (BWR). Using the above developed features, object-oriented classification techniques were used to identify automatically in wide-area satellite-based images some of the above reactor types.

Introduction

After a number of studies on the use of satellite-based remote sensing images in the safeguards procedures [1], the IAEA has now begun to use this technique [2] to check declared facilities and for planning of on-site inspections. However, for the detection of undeclared facilities, it is important to determine whether there are identifiable signatures related to nuclear facilities that could be used in the interpretation of non-intrusively acquired satellite-based imageries. Initially, in this study nuclear research and power reactors and some enrichment facilities were examined to determine if any identifiable features emerged which could subsequently be used in algorithms developed to automatically detect facilities within a large area satellite image. The object-oriented classification techniques have been developed for this purpose.

It is recognised that under the new extended safeguards agreement, Member States are obliged to give detailed information on their nuclear activities. Such information is verified by extensive on-site inspections but they tend to be costly and time consuming. Satellite imageries could minimise such problems particularly if a reference "Key" is developed.

Consider research reactors first.

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A “Key” for research reactors

Over 270 research reactors are in operation in 56 countries today. Such reactors are not generally used for producing electrical power, although they have a very high power density in the core requiring special design features. They are used for research and training, materials testing, as neutron generators and for the production of radioisotopes for medical and industrial applications. Many such reactors are located within universities and research centres.

Typically a research reactor produces power in the range between 10MW(th) and 100MW(th). In comparison, nuclear power plants produce 3000 MW(th) (or 1000 MWe). The operating modes are also different producing energy that may be steady or pulsed as in research reactors. The latter are simpler and smaller than power reactors requiring less fuel and operate at lower temperatures.

A number of nuclear research reactors were examined using aerial photographs to identify such parameters as the security fences, the shapes of various buildings associated with research reactors and also their relationships with each other. The “Key” constructed from such photographs was then used to interpret and identify such features in a satellite image with a resolution between 0.61m and 1m.

Consider the following for the development a “Key” for research reactors. A number of aerial images over several research reactors were studied to develop the “Key” and then applied during the interpretation of the satellite images. Below, a German nuclear research reactor is used as an example for discussion.

The following key features could be concluded:

- Neutron beam production reactors are rectangular measuring about 40m x 40m;
- If the reactor is a PWR, then the building is cylindrical with a dome-shaped roof of about 10m diameter that is much smaller than that used for a PWR power reactor;
- Contiguous to the reactor is a long neutron guide hall measuring 30m x 50m;
- The reactor also has mechanical draft cooling towers measuring approximately 30m x 30m;
- An exhaust stack is associated with research reactors;
- They have perimeter fences; and
- A research reactor that is used for training purposes and to investigate materials, does not have a large, long hall to accommodate neutron-related experiments.

Below, the aforementioned “Key” was used to describe a facility in a satellite image.

A test case - BER-II research reactor at the Hahn-Meitner-Institute (HMI) Berlin

The BER-II reactor which was photographed by the US QuickBird (0.61m resolution) on 13 May 2002 is shown in Figure 1. Using the above key as a guide, the image was interpreted and various features are annotated in the figure. The reactor building (17m x 25m), the experimental halls, the two cooling towers and the perimeter fence have been identified in the image. From the image, the experimental halls appear to consist of two buildings, which form an L-shape. The sizes of the buildings are 30m wide x 16m long and 23m wide x 33m long. From the shapes and the sizes of these buildings, it can be surmised that the reactor is used as a neutron generator for various experiments. The water heats up to about 40ºC and the excess heat from it is removed by circulating it through heat exchangers. The heat is finally removed and released to the atmosphere through two mechanical draft-cooling towers (Figure 1). These are constructed on the same rectangular base measuring 7m x 7m. The smaller dimensions indicate that the reactor is not generating large amounts of heat.

A similar exercise was carried out for nuclear power reactors. Below is a summary of the "Key" developed for such plants as an aid to interpretation of satellite images over such reactors.

Summary of the key for nuclear power reactors

Following are some of the important key features that were identified for a PWR:

- The reactor core, pressurised-water, primary coolant system and the steam generator are housed in a cylindrical containment building, the top of which is a hemispherical dome.
- The diameter of the dome and hence the cylindrical containment building is about 60m;
- Close to the reactor are a number of rectangular buildings housing, for example, the reactor control system, spent fuel storage pool and the turbine generator system;
- The size of, for example, the turbine and electricity generator building () is approximately 50m X 90m;
- Excess heat is carried away to the environment by either high (~150m) cooling towers (base diameter of 120m) or by very short cylindrical ones (base diameter 160m and diameter of the top of the short tower 70m) or rectangular (190m X 30m) cooling towers;
- Reactors are generally located close to either a sea, river or lake; and
- The civil PWRs examined did not have fuel-reprocessing plants in the reactor complex.

In order to test the above criteria to identify a power reactor in a satellite image, consider a high-resolution (0.61m) image of the Emsland PWR (Figure 2) near the town of Lingen in Germany. The image was acquired by the US QuickBird satellite.

The Emsland PWR is located at 52° 28"N, 7° 19"E near the town of Lingen. The construction of the reactor began in September 1982 and the commercial operation started in June 1988. It is expected that the reactor will be shutdown in June 2028.

Some characteristics of BWRs

From the above a number of important characteristics could be identified for BWRs:

- The reactor core, primary coolant system and the steam generator are housed in two rectangular containment buildings;
- The size of the outer containment building, on an average, is 35m X 45m;
- As in the case of the PWRs, close to the BWRs are the reactor control system, spent fuel storage pool and the turbine generator system all of which are housed in rectangular buildings;

Scale 1:4,200
Source: Digital Globe

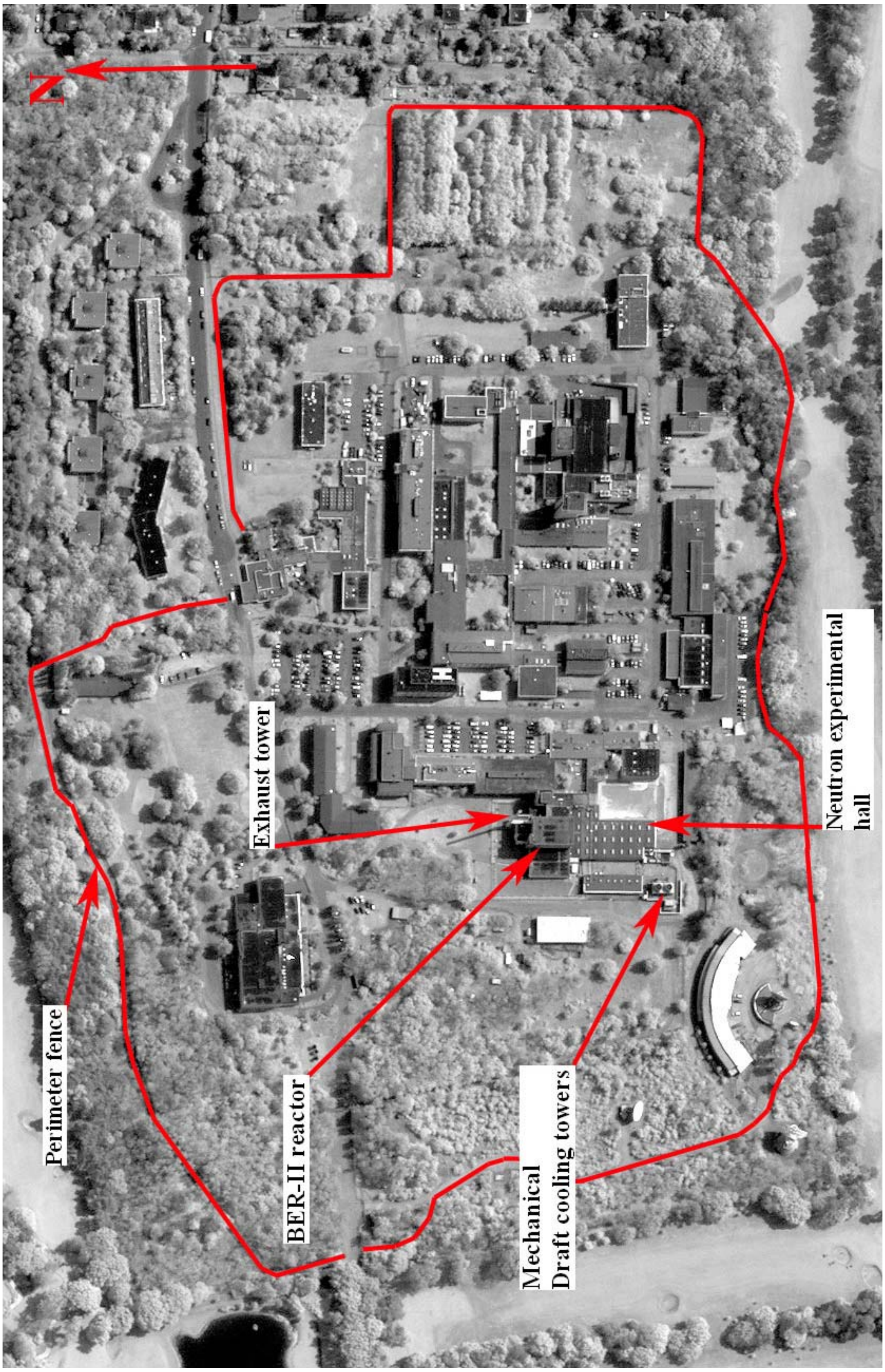

FIG. 1. An extract from the QuickBird (0.61m resolution) image shows the BER-II reactor complex. Various features can be identified using the previously established key.

- The size of, for example, turbine and electricity generator building is about 40m X 75m;
- Excess heat is carried away to the environment by discharging warm water into either a sea, lake or a river generally located near the reactor complex or via cooling towers; and
- The civil BWRs examined also did not have fuel-reprocessing plants in the reactor complex.

It is worth considering the characteristics of conventional fossil fuel power plants.

Scale 1:6,900
Source: Digital Globe

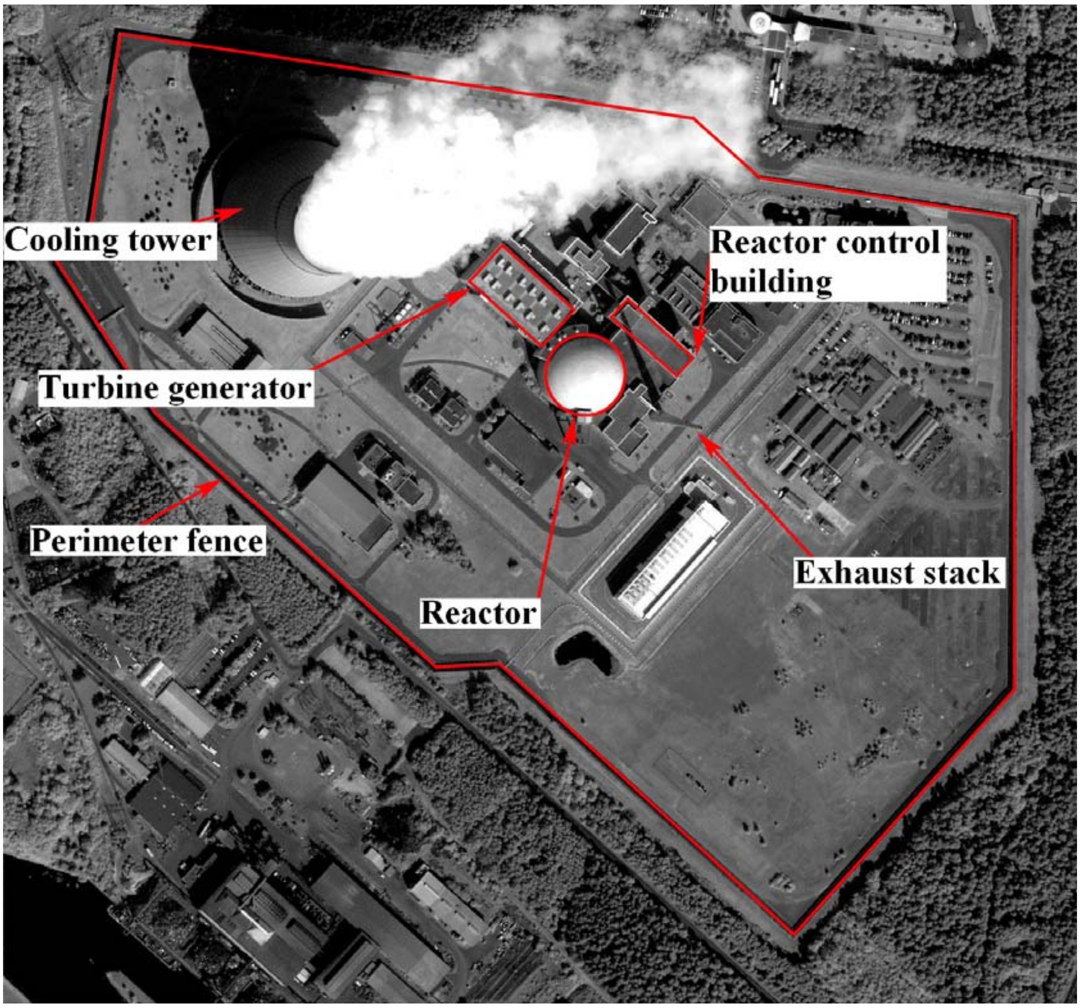

FIG. 2. This shows the Emsland PWR with some of the features identified.

Conventional power plants

It is recognised that cooling towers are a prominent feature of conventional fossil fuel power stations. They operate on the same principle as those used for nuclear power plants. Moreover, a fossil fuel power plant complex has a number of rectangular shaped buildings. However, there are two main differences: fossil fuel plants do not have cylindrical buildings similar to those used for the PWRs; and fossil fuels are placed in the same complex as the power plant in a recognisable way (heaps of coal or large cylindrical tanks for oil or gas) or they are built near the source of the fuel (such as a coal mine); fossil fuel plants have tall thin exhaust stacks but some, unlike nuclear power plants, have them on top of the power generator buildings; there is usually an extensive network of roads or railway lines in the vicinity of the power plants because, in the case of a solid fuel, the large quantities of fuel involved needs to be transported to the power plants; and, in the case of coal fuel, there is a network of conveyor belt systems to carry coal to the power plants from, for example, trains or boats carrying coal from its source. Thus, some of the following characteristics can be summarised:

- fossil fuel is stored outside, at the site of a plant as heaps of coal or gas and oil cylindrical (about 60m diameter) containers;
- gas cylinders can vary in height depending on the amount of gas present;
- unlike in the case of nuclear reactors, gas and oil containers are well separated from the power generator buildings;
- coal fuel plants have extensive conveyor belt systems and railway lines to transport coal to the power plants; and
- exhaust stacks are either built on top of the generator buildings or near them as in the case of nuclear reactors.

As mentioned above, the US QuickBird satellite image acquired on 26 June 2003 over the Emsland PWR, contained an image of a conventional fossil fuel power plant. The fossil fuel plant can be seen in an extract of the image in Figure 3. There are two buildings with exhaust stacks on top of them suggesting that these are the power plants measuring 45m x 45m. The size of the turbine building located between them is 200m x 40m. The diameter of the base of the cooling tower measures 105m and that of the top is 75m. These dimensions are similar to what would be expected for such a power station.

The dimensions of the cylinders, containing the fuel, range between 10m and 30m in diameter. Several such fuel containers are located within the perimeter fence, away from the power station. The pipeline carrying the fuel can be identified.

Thus, many of the key features identified earlier for conventional power stations can be used in the interpretation of a high-resolution satellite image. This is important, as one needs to be able to distinguish between a nuclear power plant and a conventional fossil fuel power station.

Some conclusions

It can be seen from the above, that the key features observed in high-resolution satellite images acquired over a nuclear research reactor and a nuclear power reactor are in general agreement with those developed above. A “Key” developed for a conventional power plant was also in agreement with the observed features in a satellite image. These results are encouraging to proceed with their use in algorithms developed to automatically detect facilities in a large area satellite image. The object-oriented classification techniques have been developed for this purpose.

Scale 1:11,500
Source: Digital Globe

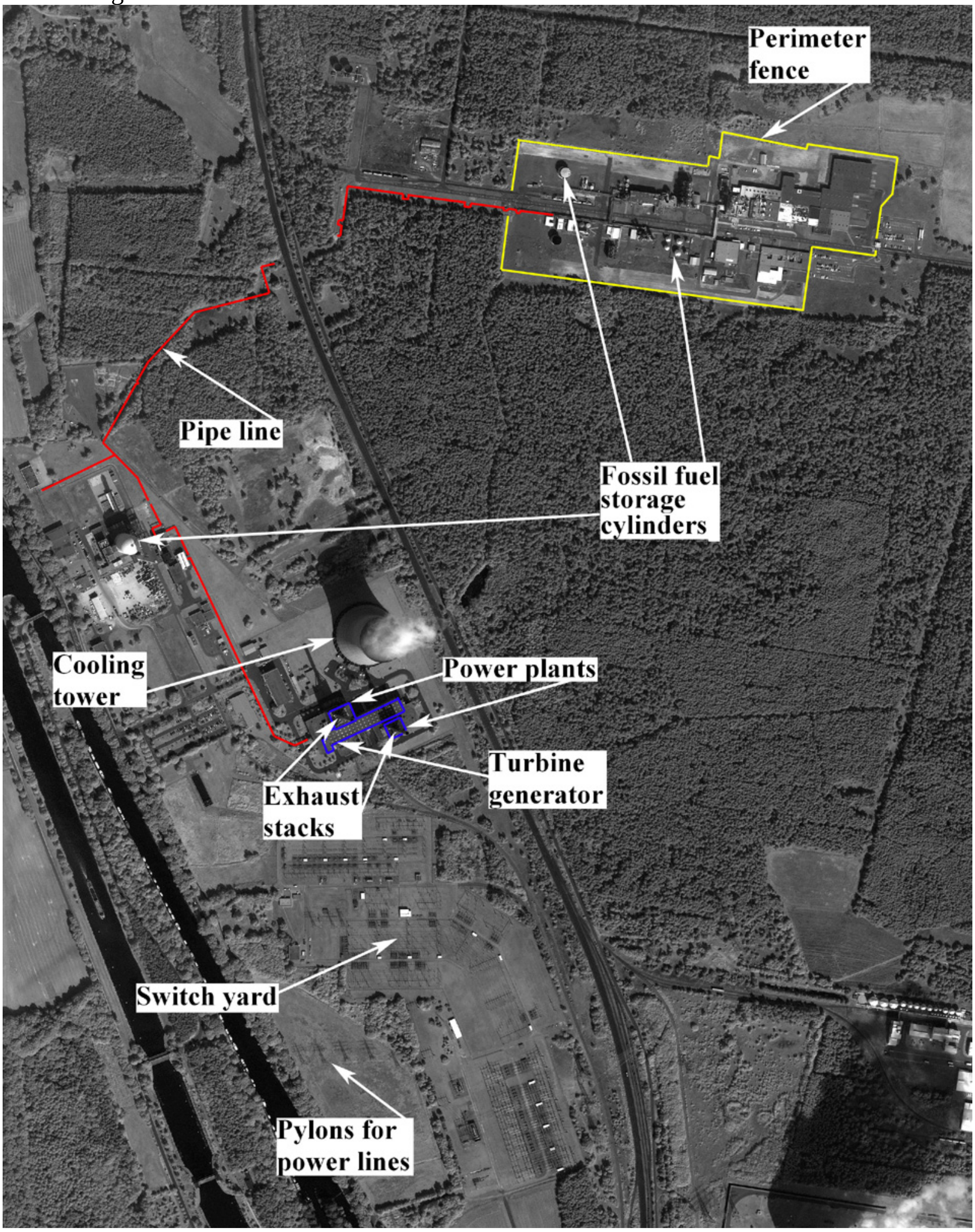

FIG. 3. This shows the Emsland conventional power plant near Lingen photographed by the US QuickBird satellite (resolution 0.61m) on 26 June 2003. Some of the fuel tanks can be seen within the perimeter fence (in yellow). Two exhaust stacks can be seen on the top of two buildings indicating the power plants. Other features such as the pylons are also identified in the satellite image.

Computer-based analysis of satellite imageries

Computer-based, object-oriented image analysis is, to a first approximation, comparable to visual perception. An image interpreter recognises, along with the colour of an image, also shapes, textures of objects and its environment, and associates meaningful objects around them. A similar goal is pursued in object-based image analysis, although the complexity and effectiveness of human perception are, of course, far from being achieved. The extraction of the objects from a pre-processed image takes place at the lowest level by segmentation, at which stage the primary segments should ideally represent the real object. Feature recognition is carried out by an analysis tool, *SEparability and THreshold* (SEaTH) [3] that provides the basis for image classification.

Data Pre-Processing

Radiometric correction procedures are necessary to obtain absolute surface radiance or reflectance by removing atmospheric effects. Assuming that the relationship between the at-sensor radiances measured at two different times can be approximated by linear functions, a relative radiometric normalization seems to be sufficient. Here, a relative radiometric normalization based on the so-called no-change pixels is applied to the image data. [4]

Wide-area monitoring using medium-resolution satellite imagery

For the object classification, a standardised and transferable semantic model being able to perform satisfactory results for a given Aster image (see Figure 4) was developed. In the given project the optimal object features and the range of its membership functions were automatically determined by the feature analyzing tool SEaTH. By this means, a classification model was defined for the object classes “industrial sites”, “mountainous areas”, “settlements”, “soils”, “vegetation” and “water”, and applied to a scene over the Bushehr reactor region. Figure 5 shows the classification result for the scenes of Bushehr.

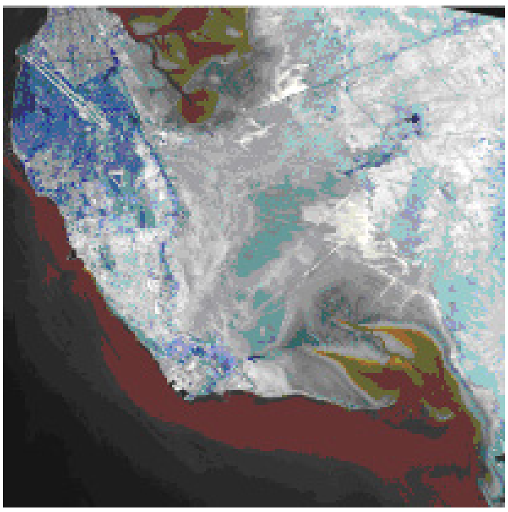

FIG. 4. An Aster image containing the Iranian nuclear power plant. The image was acquired on 26 June 2002.

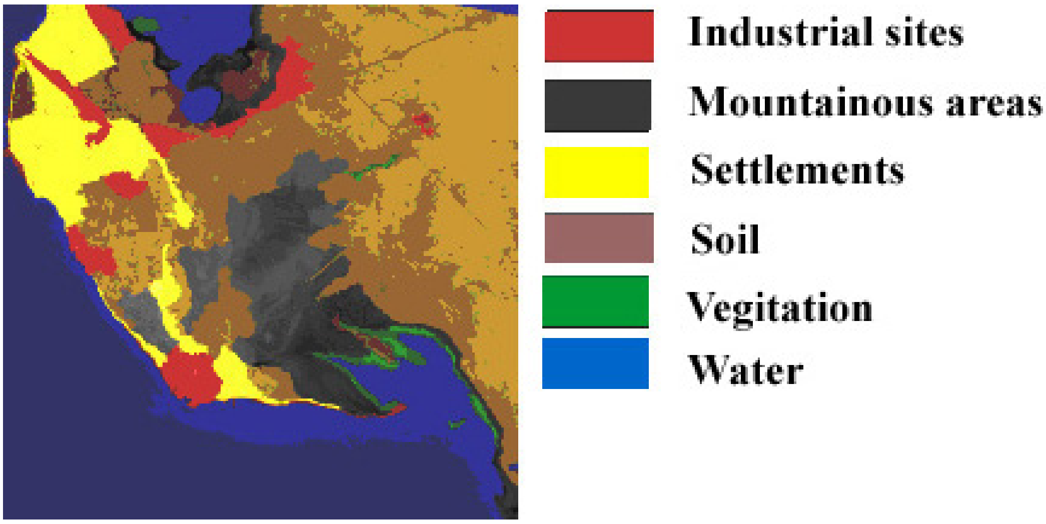

FIG. 5. This shows the image of Figure 4 after a computer-based analysis was carried. The area around the reactor site is now classified as shown in the image.

The image was classified by six classifications that include areas consisting of industrial activities. It can be seen that these areas are numerous. Thus, a method is needed to further focus on areas containing nuclear facilities. Here the above developed “Key” should be implemented. [5] However, for this to work, an image with better resolution than the Aster data (15m resolution), is required.

Some conclusions

It can be seen that the “Key” developed for PWRs, BWRs and conventional power stations using aerial photographs could be applied to identify facilities of interest in high-resolution satellite images. These results are encouraging enough to proceed with the development of “Keys” for other facilities such as the uranium enrichment plants.

As for computer-based automatic identification of such objects in a large area image, the method needs to be fully tested using higher resolution images than the ones used within this study (Aster images were used). For example, for future studies, improved results would be achieved from the algorithms via the use of SPOT and IRS image data, which have resolutions ranging from 5m to 2.5m.

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Remote sensing technology for nuclear verification

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Abstract. In this study, general remote sensing concepts are presented with some techniques for detecting and extracting information of ground activities and an advanced technology for accessing satellite imagery data of huge size more than several GB is introduced. The brief algorithm and test results of this new technique are presented, and compared with those of a typical data access method. The capability of analyzing spatial information is expectd to be highly enhanced with this new method, which can greatly reduce the time requried for loading and processing imagery data.

1. Introduction

Recently, International Atomic Energy Agency (IAEA) has focused its efforts on strengthening nuclear verification capability to find undeclared nuclear activities. In this context, the Agency has proposed measures for enhancing these capabilities in the areas of : (i) environmental sampling, (ii) satellites imagery, and (iii) information on nuclear procurement and supply.

Although spatial imagery technologies have been applied in mainly environmental or military sides, its application areas are expanded to general purposes such as construction, geographical information system, insurance, agriculture, etc. Nowadays, satellite imagery is routinely used for safeguards purposes such as the analysis of the member states' declarations and the planning of inspections and CA. It is also used in safeguards verification to find undeclared nuclear facilities or activities where the member states do not provide the access for the Agency inspectors due to some political issues or so.

It is not unusual to access high resolution satellite image data (~0.6 m) with the start of the commercial services of advanced earth observation satellites such as IKONOS and Quickbird. With satellite photographs of high resolution (~0.6m), information of objects on the ground can be accessed without visiting the place of interested. RS technology can help the Agency in confirming the operational status of nuclear facilities, and in understanding the features and changes of building in nuclear sites.

Since the most works are currently limited to just observing the ground features with raw satellite images, however, so the useful but invisible information inside the imagery has not been used sufficiently. Satellite imagery has something more than a simple photograph if supported by an appropriate software tool to analyze that inside information. Thanks to the recent development of spatial information technology using computer graphic skills, it can reveal hidden features weaved into the surface and construct virtual reality with three dimensional rendering skills using DEM (Digital Elevation Model). To get the whole benefits of the satellite imagery, it is necessary to manipulate the GB size image data readily, which requires special data processing tricks not in general-purpose graphic tools.

Since the Agency has plans to enhance its analytic capabilities with the member states' assistance, so it is important to review various and advanced RS technologies to support the Agency. In this paper, basic concept and general applications of remote sensing technologies using satellites imagery are reviewed and a new skill to manipulate huge size image data is presented.

2. Concept of Remote Sensing

2.1. Definition of Remote Sensing

Remote sensing is the science of obtaining information about an object, area, or phenomenon through the analysis of the data acquired by a device that is not in contact with the object, area, or phenomenon under investigation. But it is usually limited to the technology to extract information from the image data which are achieved by sensors mounted on the airborne or space borne platforms.

2.2. Resolution Category of Spatial Imagery

In order to collect data of ground objects or earth surface, sensor should gather reflecting electromagnetic waves from the targets. The reflecting electromagnetic waves have unique characteristics dependent on surface conditions and environmental parameters such as humidity, temperature, or content of air pollutants. The achieved image can not convey the real value of the surface data due to the physical limitation of the sensors. The quality of the data is determined by the resolving power of sensor, which might be classified into four categories as described below.

(1) Spatial resolution : a measure of the smallest angular or linear separation between two objects that can be resolved by the sensor.

(2) Spectral resolution : the number and dimensions of specific wavelength intervals in the electromagnetic spectrum to which a remote sensing instrument is sensitive.

(3) Radiometric resolution : the sensitivity of a remote sensing detector to differences in signal strength as it records the radiance flux reflected or emitted from the terrain.

(4) Temporal resolution : frequency of data acquisition over the area.

4m x 4m resolution 16m x 16m resolution

Figure 1. Image Comparision (Spatial Resolution).

3. Application Using Spatial Imagery

3.1. Land Coverage Classification

Each pixel of satellite imagery contains specific data representing the surface information. If appropriate software technique is used, whole image pixels can be categorized automatically into land cover classes or themes based on the pixels' attribute, i.e., spectral pattern.

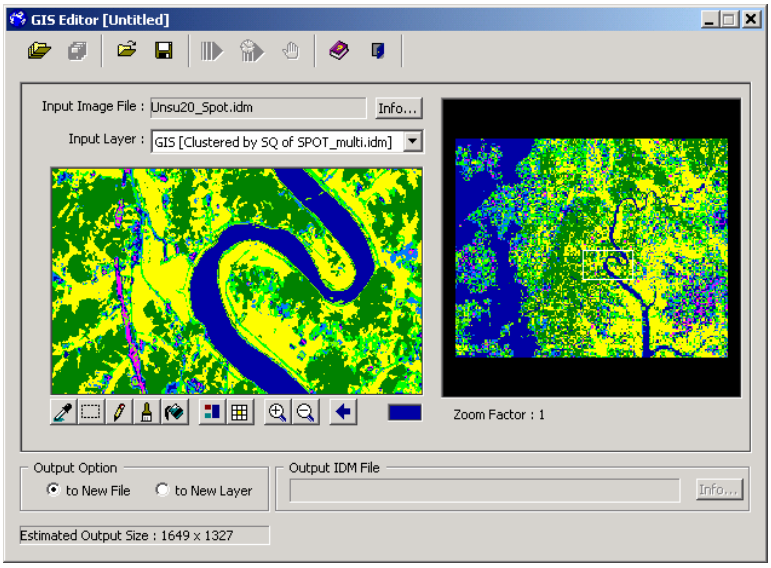


Figure 2. River Classification Example.

3.2. Feature Extraction

As an advanced technique of Classification, Feature Extraction is still under development for full automatic execution. The concept is to take out typical images such as building, road, or specified area based on user selection and pixel attribute.

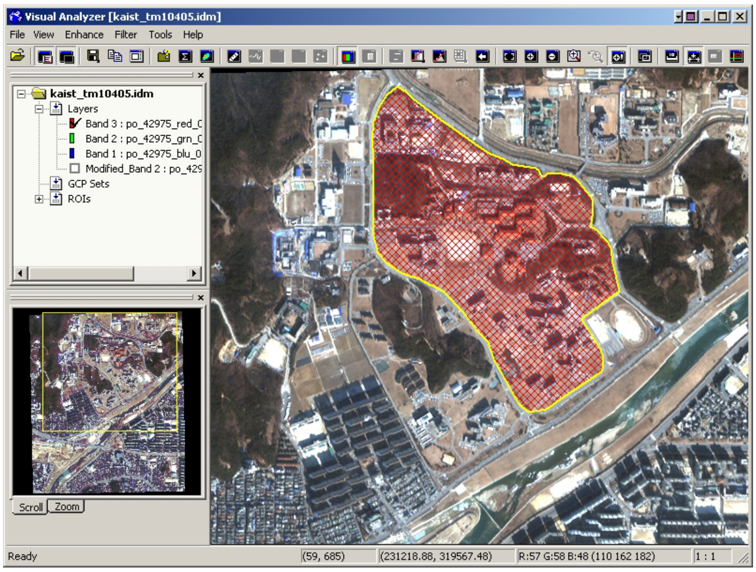


Figure 3. Feature Extraction (Delineating Area).

3.3. 3-D Terrain Analysis

Using stereo-typed images or electronic map with contour data, DEM(Digital Elevation Model) data can be constructed and used to make 3-D space in PC. Within this virtual space, terrain analysis can be performed which is not possible at the 2-D image.

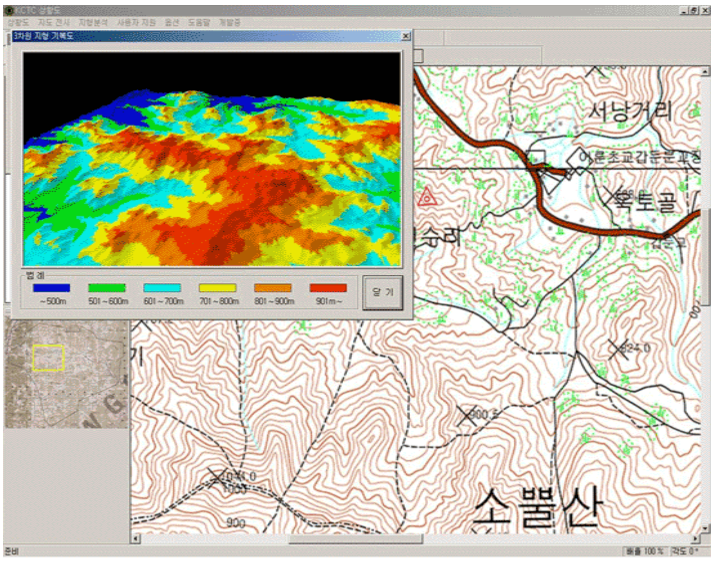


Figure 4. 3-D Terrain Analysis.

3.4. Change Detection

This is a technique to find the change of land coverage characteristics using arithmetic operation between images taken with time interval. It can be effectively applied to the structures which should be regularly monitored such as suspicious military bases or nuclear facilities.

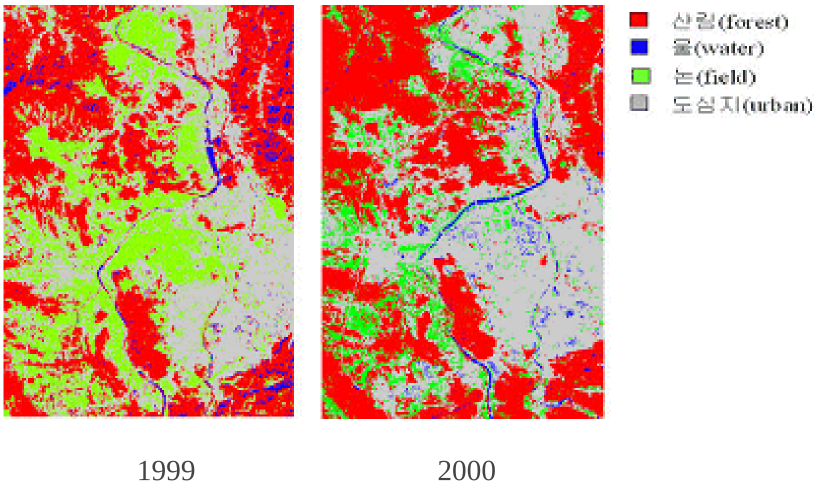


Figure 5. Land Coverage Classification of Daejeon.

4. Real Time Processing Technique for Huge Size Image Data

Since the data size of spatial imagery tends to increase by geometric progression, it is required to develop efficient data structure for real time processing of huge size data. The reason that the current computing power of typical desktop PC is able to treat GB size image data is due to the limitation of the display area. In the view of application program, low resolution of small size data is required when the whole area is displayed and only partial area is required when the image is displayed with original resolution. So it can be understood that the relation of Level of Detail (LOD) of data and Region of Interest (ROI) is in inverse proportion, and the data size which should be processed is less than a certain value, as shown in Fig. 6.

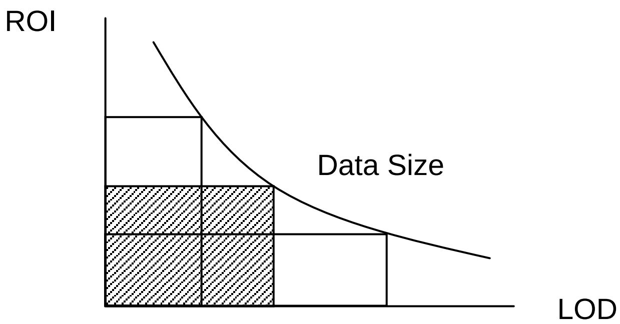


Figure 6. Relation between LOD and ROI.

In general, each LOD image data are prepared in advance in order to load and process huge size satellite image in real time, which is called image pyramid or MipMap. But it is required to make additional data about 40% of the original image size. Data contraction technique can be an alternative, but data loss is accompanied inevitably.

A new method which doesn’t require preparing additional data is suggested using LOD and image tiling tricks. With this new method, the GB size satellite imagery data can be processed in real time without data loss. The key point is to transform the pixel order within the data and rearrange the pixel data to reflect all LOD in one data set. Fig. 7 shows that the access time is greatly reduced when the data tiling is used.

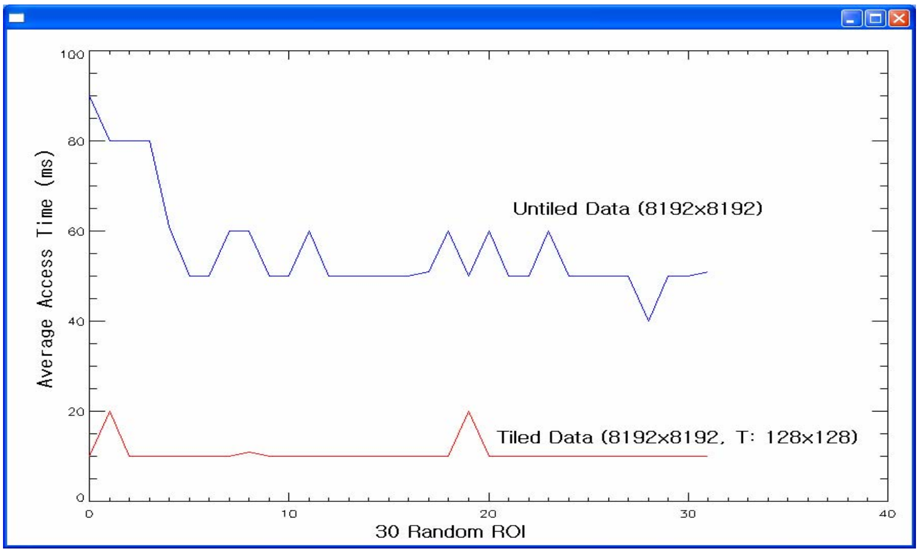
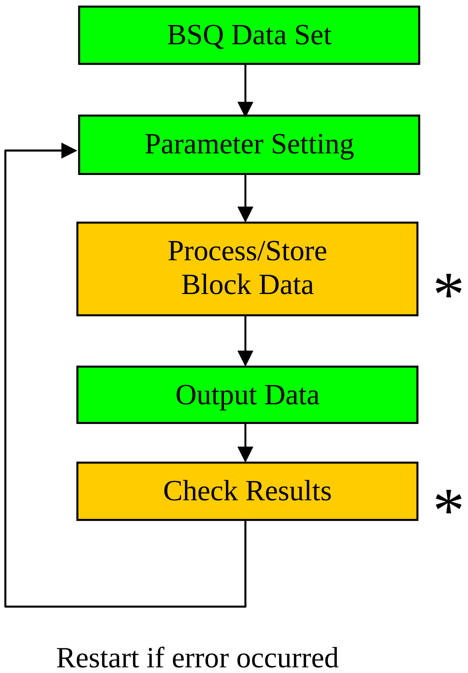


Figure 7. Comparison of Data Access Time.

Currently, most big image data are stored in BSQ format, which arranges its pixel data from the first one to the final one in sequential series. Manipulation with this format requires long time for extracting data of each resolution, so it is not possible to process data in real time. The data processing flow chart is shown in Fig. 8.

The data processing time is shortened with the new technique, and the processing results can be checked before making output file. This type of data processing reduces the overall working time greatly required for manipulating satellite imagery data. Fig. 9. shows the data processing flow chart for the new method.



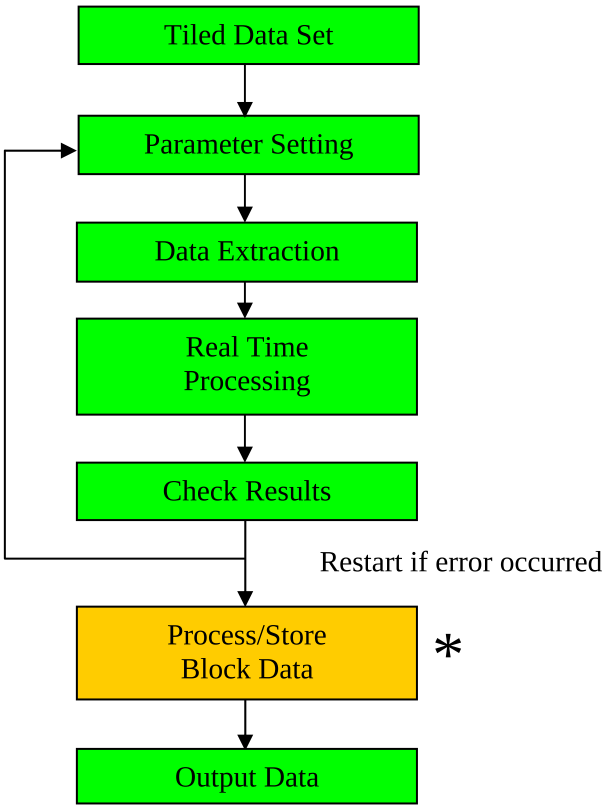


Figure 8. Flow Chart for BSQ data. *Figure 9. Flow Chart for Tiled Data.*

5. Performance Test

In order to confirm the effectiveness of new technique, performance tests were carried out under the following condition.

[Test Condition]

- Transformed Data with Tiling Structure and 4 LOD with 8192×8192 size
- Column Ordered Data with 8192×8192 size
- Test Frame : P IV Mobile 1.2GHz, 512MB Ram, Windows XP

Table 1. Extraction Time for Whole Area.

Level of Details	Avg. Time for Extraction (ms)	
	Column Ordered	Tiled
0 Lv : 8192x8192	160.2	119.2
1 Lv : 4096x4096	422.7	44.0
2 Lv : 2048x2048	118.2	10.0
3 Lv : 1024x1024	34.0	4
4 Lv : 512x512	13.0	0.1

Table 2. Loading Time for Partial Area.

Level of Details	Avg. Time for Loading 40 ROI (ms)	
	Column Ordered	Tiled
0 Lv : 512 x 512	70.1	10.2
1 Lv : 256 x 256	21.0	2.0
2 Lv : 128 x 128	7.0	0.9
3 Lv : 64 x 64	3.0	0.4
4 Lv : 32 x 32	2.8	0.8

The results show that the new method with tiled data structure is very effective for the real time processing of the huge size data and reduces the working time.

6. CONCLUSIONS

Although it is generally recognized that RS technology has contributed to find undeclared nuclear activities, its usage is still limited to visual analyzing satellite images due to lack of appropriate software technique. The new data processing technique introduced in this study will enhance the capability for analyzing satellite imagery if adopted.

Since the skill for hiding clandestine structures or activities will evolve in line with the advancement of verification technology, it is required to develop new techniques to draw more correct information from imagery. The verification capabilities with RS techniques are needed to be enhanced by developing state-of-art technologies including SAR (Synthetic Aperture Radar), Multi Spectral Image, and Automatic Change Detection, etc.

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IAEA safeguards information system re-engineering project (IRP)

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Abstract. The Safeguards Information System Re-engineering Project (IRP) was initiated to assist the IAEA in addressing current and future verification and analysis activities through the establishment of a new information technology framework for strengthened and integrated safeguards. The Project provides a unique opportunity to enhance all of the information services for the Department of Safeguards and will require project management 'best practices' to balance limited funds, available resources and Departmental priorities. To achieve its goals, the Project will require the participation of all stakeholders to create a comprehensive and cohesive plan that provides both a flexible and stable foundation for address changing business needs. The expectation is that high quality integrated information systems will be developed that incorporate state-of-the-art technical architectural standards, improved business processes and consistent user interfaces to store various data types in an enterprise data repository which is accessible on-line in a secure environment.

1. Background

In a statement to the Board of Governors in February 2005, the IAEA Director General affirmed that the Secretariat has been re-engineering the IAEA Safeguards Information System (ISIS) in order to improve the effectiveness and efficiency of information analysis and to reduce the risk of failure associated with antiquated computer systems.

The ISIS, developed in the late 1970s, encompasses a network of computer systems used by the Department of Safeguards to collect, store, analyze and evaluate safeguards-relevant information. Over the past decades, to meet new verification and analysis challenges the ISIS has been enhanced through the use of modern software technologies, rather than through investments of additional resources for the outdated mainframe. Although a significant amount of resources has been required to maintain the complex system of interrelated applications operating in a mainframe and client-server architecture, new functionality has not been easily integrated with applications running on different environments. System enhancements needed to meet new business requirements relative to strengthened and integrated safeguards call for a higher level of interconnectivity and a wider degree of flexibility than can be met using the existing computer system infrastructure.

To resolve this issue, the ISIS Re-engineering Project (IRP) was initiated to integrate, replace or re-engineer all elements of the ISIS, including the technical infrastructure and other safeguards applications. In July 2005, the IRP commenced in order to provide the Department of Safeguards with quick and reliable access to all required information through advanced integrated systems.

2. A win-win situation

In December 2005, the Director of the Division of Safeguards Information Technology (SGIT) stated that, in the context of the move to information-driven safeguards, the IRP provided a unique opportunity to achieve an improved modern IT architecture that was urgently needed to support the IAEA's verification mandate. It is the responsibility of SGIT to make this happen, although the quality of the results will depend greatly on overall Departmental engagement.

A primary goal of SGIT is to adequately support the Department's current and future verification and analysis work. To this end, information technology should be an integral part of safeguards' business processes and the IRP should adopt a comprehensive, business-centric approach to address all of the Department's information needs. The expected outcomes of this multi-year project include:

— Reduction in the risk of failure in drawing erroneous conclusions as a result of faulty safeguards IT applications;
— Improvement in the information systems to support both State-level and facility-level safeguards approaches;
— Improvements in safeguards' business processes and reduction of evaluation turn-around times;
— Implementation of workflow systems for more efficient process handling (e.g. inspection documentation package (IDP) tracking system);
— Creation of a secure, stable state-of-the-art IT infrastructure; and
— Improvement in data quality and reduction of data complexity and redundancy, facilitating the analysis of stored data and reducing the costs of using and maintaining safeguards IT applications.

3. Building a stable foundation for safeguards information technology

The IRP Solution Design projects (phase I) and the Foundation projects (phase II) aim to create a stable framework which can easily adapt to meet the changing needs of the Department of Safeguards. Key objectives of the phase I and II projects include:

— Development of a service-oriented architecture to optimize the re-use of common services and reduce redundancies of services across applications;
— Use of reverse engineering tools, as appropriate, to extract the requirements and business logic of existing applications for future integration in the new architecture;
— Development of architectural standards that define application development and standard tools for the implementation of all safeguards information systems;
— Development of a modern hardware and software platform that integrates current and future systems and provides users with a one-stop-shop service;
— Development of a data migration strategy to optimize the replacement of old applications with new ones;
— Development of an enterprise data model that provides a complete, consistent and coherent representation of the Department's data systems, to minimize data redundancy and ensure data integrity;
— Design and implementation of a common and consistent user interface which would allow role-based access to all required functionality and data from any supported location (where security restrictions permit); and
— Development of a central security system to control access to applications and data, thus avoiding several user-ids/passwords per user (single-sign-on).

4. Supporting safeguards today and tomorrow

When the IRP statement of work was written, various assumptions and references were collected and provided as inputs for the 'rough plans' to the Implementation Plan (phase III). Applications were identified, grouped and scheduled, and best 'guesstimates' were made about resources, costs and project durations. It was expected, and planned, that a re-assessment of phase III would be conducted near the end of phase II to update the activities and resource management plans. When the phase III Implementation Plan was revised to meet the changing business requirements, various factors were considered: (a) acknowledge urgent user needs, (b) inject acquired project knowledge, and (c) synchronize the current business processes with the newly documented process improvement activities. In addressing these needs and in seeking to continuously improve the efficiency and effectiveness of SGIT's project management activities, the new comprehensive phase III+ Implementation Plan now supports a larger scope of applications and provides for more robust coordination and integration. The updated plan also included consideration of the needs of all stakeholders (i.e., Departmental users, Management, and Member States) and all variables (i.e., priorities, funding, resources, dependencies and constraints). The objectives of the revised phase III+ Implementation Plan include:

— To produce a business-centric approach that aligns the information systems with the related safeguards processes;
— To optimize limited funds, available resources and Departmental priorities; and
— To provide a detailed financial and resource management plan.

5. The IRP impact on stakeholders

The expected benefits of the IRP on stakeholders include improvements in the collection, analysis and verification of all safeguards information. To facilitate the management of the IRP, four primary business areas have been defined, as described briefly below.

The ‘State-supplied data’ business area contains the applications required to capture, validate and report on declarations received from Member States. Planned as an early phase III deliverable, the Safeguards mailbox information system will allow Member States to securely send data electronically to IAEA headquarters in Vienna.

The ‘Verification’ business area includes all activities related to inspection planning, implementation and reporting resulting from verification activities. The IRP will build upon the successful results of the CIR-mobile inspector tool to formulate the foundation for the future integrated field inspection toolbox.

The ‘Analysis’ business area is concerned with evaluating the effectiveness of verification activities as well as contributing to the drawing of soundly based conclusions at the facility and the State levels — conclusions based on all source information, including data provided by the State-supplied business area and the Verification business area. For the Analysis area, the IRP seeks to develop a flexible infrastructure to ease the development of custom solutions needed for specialized analytical tools. This flexibility will facilitate the integration of external systems, i.e., the synchronization of external data from the nVISION project as well as from satellite imagery, environmental sampling, illicit trafficking and nuclear trade information systems. As a result, the new environment should provide all required data to the desktop of the analyst.

The ‘Support’ business area includes the logistical and planning systems needed to support the activities of the Department, such as equipment, funding and other decision support systems. The enhanced computerized equipment requisition system will be an early-return application that complies with the architectural and programming standards produced in the phase I and phase II projects.

6. Managing IRP resources and risks

The implementation of the IRP is a multi-year project with high investment costs. Although the Plan involves a modular development process to ensure a full migration from the mainframe, current project management ‘best practices’ should be followed to minimize risks such as the limited funds and staff resources required to complete a deliverable. Since collaboration is important to fulfil project goals, it is essential that all users (i.e., inspectors, analysts and support staff) contribute in the design of the future system. A successful team environment requires effective communication, trust and mutual understanding. These joint efforts will contribute to the usability and quality of the next generation safeguards information systems.

Participation from the business area experts and users, particularly those with significant field experience and analytical experience, will greatly affect the success, cost and duration of the IRP. Consolidated and validated requirements, coupled with optimized business processes, will reduce costs and ultimately produce the ‘product’ the Department needs. To achieve better results, in coordination with the Safeguards Departmental Quality Management System (QMS), the IRP will help to define safeguards' business processes.

Efficient resource planning and management requires detailed plans, budgeted resources, and coordinated tasks to prepare for the dynamics of a large project. Shifts in priorities, staff and unexpected events can adversely impact the project. To achieve a stable basis for system development, the users and customers should communicate both current and future system requirements. The validated requirements will be vital to the project since changing requirements would adversely affect the timeline and budget of the programme.

Securing the funds needed to meet the project goals will require active support from IAEA management and Member States. Only with the support of all stakeholders (i.e., users, management and Member States) can the programme deliver the expected benefits.

Risks are inherent in any large, complex re-engineering programme and need to be professionally managed. The IRP follows best practices in project management to continuously monitor risks that impact the success of the projects. In this context, the IRP's Programme Management Office works to define, analyze and mitigate risks on a regular basis.

The major risks facing the IRP today include: (a) the need for additional resources to complete the projects on-time while business continues 'as usual';(b) a new data centre to house the new development, test and production environments; (c) a standard set of improved business processes; (d) a major transition to the implementation phase of the IRP; and (e) the management of the change in our business activities.

7. Conclusions

In summary, the success of the IRP requires successfully addressing the following major issues:

— Availability of sufficient funding to support the full duration of the IRP project (based on a pre-arranged funding strategy).
— Full commitment of senior management and the regular, open communication of this commitment to ensure the acceptance of the project and resolve problems. The outcomes of the IRP will result in wide-ranging changes at all levels of Department of the Safeguards. Success will depend on the ability to manage change.
— The implementation of effective programme management to ensure the availability of adequate resources, manage communication at all levels and meet project milestones in a timely and cost-efficient manner. The IRP project involves parallel projects and multiple project dependencies; thus the availability of suitably skilled, trained and experienced resources is critical. The lost of key staff and critical resources could result in delays in the work and pessimism among stakeholders.

A common key management infrastructure for safeguards data

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Abstract. Risks to the security of safeguards electronic data, the need for secure communications with Member States, and the growth of systems with distributed security functions, have emphasized the need for a common key management infrastructure. This infrastructure must be flexible enough to support the expanding range of secure applications and must adopt the technology and procedures necessary to ensure that sufficient trust can be placed on the keys being managed. A common key management approach, based on international standards, will minimize the need to duplicate key management functions by each application, will ensure the more consistent adoption of uniform and formally approved security policies, and will simplify and encourage the use of shared secure credentials, including smart cards. This paper discusses the requirements for a Public Key Infrastructure (PKI) to support the protection of safeguards sensitive data. It describes the work underway to develop the policies, procedures and technical infrastructure required, and to provide initial support for a small number of public key applications, including an "electronic mailbox" system being deployed to support the secure receipt and acknowledgement of operator supplied information.

1. Introduction

It is the policy of the IAEA to preserve the confidentiality, integrity, and availability of information [1]. Information Security constitutes a necessary part of the foundation upon which the integrity of the Agency as a credible and effective organization must stand. The Public Key Infrastructure (PKI) described in this paper will be an essential element of the Information Technology (IT) security framework being deployed by the Safeguards Information Technology Division (SGIT) [2].

Today, many systems and procedures are in place to secure Agency data. These include cryptographic systems to sign data generated by surveillance equipment, to control access to equipment and systems, and to encrypt data for transmission over public data networks or for storage on portable media (e.g. memory sticks). In addition, Member States are demanding better means to secure their electronic communications with the Agency, and some of the procedures being developed to support Integrated Safeguards will require electronic means to securely submit and time-stamp information.

While some legacy systems continue to use private key cryptography, and this will continue for some select applications, an increasing majority of systems are relying on the use of public key technology. Public key approaches greatly simplify key management, but the trusted generation, distribution, maintenance and destruction of keys remains a crucial requirement shared by all systems.

Over the years, the Agency has deployed many approaches for the management of keying material. While meeting the requirements of their day, their continued proliferation will increase costs and make it difficult to allocate the attention needed to each one. However, existing mechanisms for the secure transfer and storage of information must continue to be supported, and the migration to a common PKI will only occur as new compatible applications are deployed, and with the approval of Member States.

A common key management approach, based on international standards, will:

- Permit the full harvesting of the benefits of public key cryptography by minimizing duplication and providing a full range of key management services to all applications,
- Enhance security by ensuring the uniform application of formally approved security policies to all supported applications,
- Simplify operations by encouraging the use of shared secure credentials useable by multiple applications, for example using hardware tokens or smart cards, and
- Provide a standard key management service that may be exploited by new applications as they are developed and deployed by the Agency.

A PKI, along with the associated approved policies and practices, is required for the creation and management of private keys and public key certificates[1].

Keys must be created securely, backed-up when used by applications requiring key recovery, and periodically replaced so that they are not used for too long. Certificates corresponding to private keys must only be issued to fully identified entities (persons or devices) for authorized purposes.

Information embedded within certificates permit them to be tailored and restricted to specific uses and applications. Applications can rely on the signed content of certificates to determine whether the corresponding keys have been issued by a key management system which uses technology and procedures that are sufficiently trusted to support the security required. Additional information within certificates define the lifetime of keys, whether keys should be used for signing only, or encryption only, or only for specific purposes, e.g. only for the identification of secure web sites. Proper enrolment procedures will ensure that only authenticated entities receive the electronic permissions (as tailored within the certificates they receive) needed.

2. Components of the Public Key Infrastructure

Public Key Infrastructures can be considered to consist of four major components:

- A Certification Authority (CA) – a system and associated personnel trusted to issue keys (and certificates) for specific purposes. The CA will issue keys; revoke keys upon suspected or real compromise; provide notice to certificate users when keys are revoked; and provide for the recovery of keys when requested by authentic individuals.
- A Registration Authority (RA) – a person and associated workstation trusted by the Certificate Authority to identify individuals or devices that are authorized to receive certificates. The CA relies on the RA to validate requests and provide the interface to users. Registration Authorities may be co-located with the CA or physically located in local Agency offices.
- A Publication capability, which is responsible for publishing the public key certificates and other information required to support ongoing operation. For example this capability may include a directory or interface to a corporate "electronic address book" within which

[1] In public key cryptography, pairs of keys are generated with one of the keys, the "private" key, used for decryption or signature generation, and the other, "public" key, used for encryption or signature validation. The private key is kept secret by its owner while the public key is placed within an electronic certificate which, along with other identifying information, is signed and issued by the public key infrastructure.

certificates are published, and a web location where signed lists of revoked certificates are published.

- A Management Authority – typically a group of senior management staff who are given the authority to approve the operation of the infrastructure, to judge the appropriateness of the procedures used for its operation, to approve the types of certificates that may be issued (and the security policies associated with them), and to make agreements with external authorities when cooperative security requirements must be met.

Also essential to the operation of PKI is the concept of role separation. Typically, responsibilities to perform actions within the infrastructure are limited to certain individuals who are assigned roles in a manner to prevent collusion. For example, an Audit Role is defined and allocated to one or more individuals who do not have any other PKI related responsibilities. In some cases “two out of three” persons may be required to authorize certain activities.

The overall operation of the systems, including the the configuration of the systems, the kinds of keying material generated and maintained, the roles and duties of staff, the procedures used to identify individuals and to validate requests for key recovery, and to periodically assess the security of the systems, must be fully documented. This represents a very significant task since it covers not only the technical specification of the systems, but also the management and procedural controls that will be used to maintain the security of the system.

In cases where it is necessary for Member States and the IAEA to exchange information in confidence, or when Member State and IAEA data is to be protected by the other party, there may be a need to communicate the level of assurance that can be placed on the PKI, and on the certificates and keys it issues. This adds another dimension to the concept of “interoperability”. Not only must these secure systems interoperate technically, it may also be necessary to compare the relative levels of “assurance” that can be placed on the key management infrastructures[2] on each side of the interface.

3. Standards Based Approach

Clearly the deployment of a PKI by the Agency will rely on the extensive use of standards to allow its most widespread use. Interoperation with Member States, and the promotion of the use of Commercial off-the-Shelf (COTS) security solutions, will also depend on the use of standards.

Thus the Agency has selected an approach which will maximize the use of standards-based COTS equipment, and which will provide a stable framework against which COTS and custom secure applications may be developed (should custom development be required).

Important standards and de facto industry agreements relevant to the IAEA PKI include:

- The International Telecommunications Union (ITU) and International Organization for Standardization (ISO) X.509 [3] standard which defines the content of public key certificates and certificate revocation lists. This is universally accepted as the standard for certificate content. X.509 uses the ITU and ISO ASN.1 [4] standard to represent its data and this provides flexibility in the kinds of information that may be included in a certificate. This flexibility allows the X.509 format to be used by a range of applications.
- The Internet Engineering Task Force (IETF) RFC 3280 [5] document provides a set of recommendations for specific content of X.509 certificates, certificate revocation lists, and for the way they should be used by secure applications. It identifies preferred alternatives and recommends against certain uses which may be less secure. It is widely used in the industry and its adoption allows certain assumptions to be made by certificate-using applications which can simply their implementation.

[2] In this case we might assume that the Member State, or facility operator or state system of accounting and control within the State, also deploys a PKI, or purchases PKI services, to issue keying material to their users.

- The IETF RFC 3851 [6] document, and the documents to which it refers, define a commonly used standard for secure electronic mail. This standard is being used by the IAEA secure mailbox system (discussed later). Specifically relevant to the Agency PKI, this standard defines minimum requirements of X.509 public key certificates for use by electronic mail.
- The IETF RFC 3647 [7] document defines a common presentation format for *Certificate Policy* and *Certification Practice Statement* documents. The use of common formats facilitates their assessment, and the comparison of equivalent documents issued by other PKIs should the equivalency of security measures need to be compared.
- The United States National Information Assurance Partnership CIMC Protection Profiles [8] define minimum certificate management security requirements for a range of environments. These profiles may be used to guide the specification of the systems and procedures needed.

4. Current Program

Work is underway in conjunction with the SGIT and with input from the Department of Management, Division of Information Technology (MTIT), to define the types and assurance levels required for public key certificates. It will develop the technical, policy and procedures needed for the deployment of a PKI to address safeguards security requirements and which will be compatible with future deployments to meet non-safeguards IAEA-wide secure applications.

This work is leveraging international standards and profiles for public key infrastructures, and aligning safeguards requirements with accepted international practice. Policies and procedures are being developed to meet both IAEA and Member State requirements and the resulting PKI will become an integral part of the new IT security framework being deployed by the SGIT.

This paper describes ongoing work which is still in its formative stages and many final decisions have yet to be made by the Agency. Thus this paper provides preliminary results which may not be carried through to the final system deployments.

Figure 1 provides a preliminary view of the target architecture for the IAEA PKI. This architecture is designed to be responsible for the management of keys and certificates at two assurance levels:

- The Enhanced Assurance Level to support applications such as safeguards confidential applications, and
- The Basic Assurance Level to support applications such as those at the restricted level.

The term “assurance” is used to refer to the level of confidence that may be placed on the subject identity and the binding between this identity and the public key contained within the certificate. Coupled with certificate-using security applications with equivalent levels of assurance, and using sufficient cryptographic strength (e.g. a large enough key size), it is intended that Enhanced Assurance certificates be used for the protection of confidential and safeguards sensitive data. Added mechanisms may be used to address higher security levels.

As a practical matter, it is expected that a smaller number of Enhanced Assurance certificates will be issued, and only to staff requiring access to the more sensitive data. A larger number of Basic Assurance certificates will be needed for a wider population of IAEA staff[3]. This will reduce overall costs since the corresponding Enhanced Assurance certificate procedures will only be used when

[3] Enhanced Assurance certificates will also be capable of being used for basic security application, thus a user will not require both types of certificates.

needed. Devices (IAEA internal[4] web servers, secure virtual private network devices, etc.) will be issued Enhanced Assurance certificates since these systems will typically require higher trust.

Since the most significant number of highly secure applications reside in the Division of Safeguards, it is appropriate to centre work related to the issue of Enhanced Assurance certificates in SGIT. For other secure applications, as might typically be encountered by most IAEA staff, it is expected that the MTIT will deploy components suitable for the issue of these certificates. The “root” CA, which issues certificates to the subordinate CAs responsible for issuing end user certificates must operate at the highest assurance level and thus is shown co-located with the Enhanced Assurance CA.

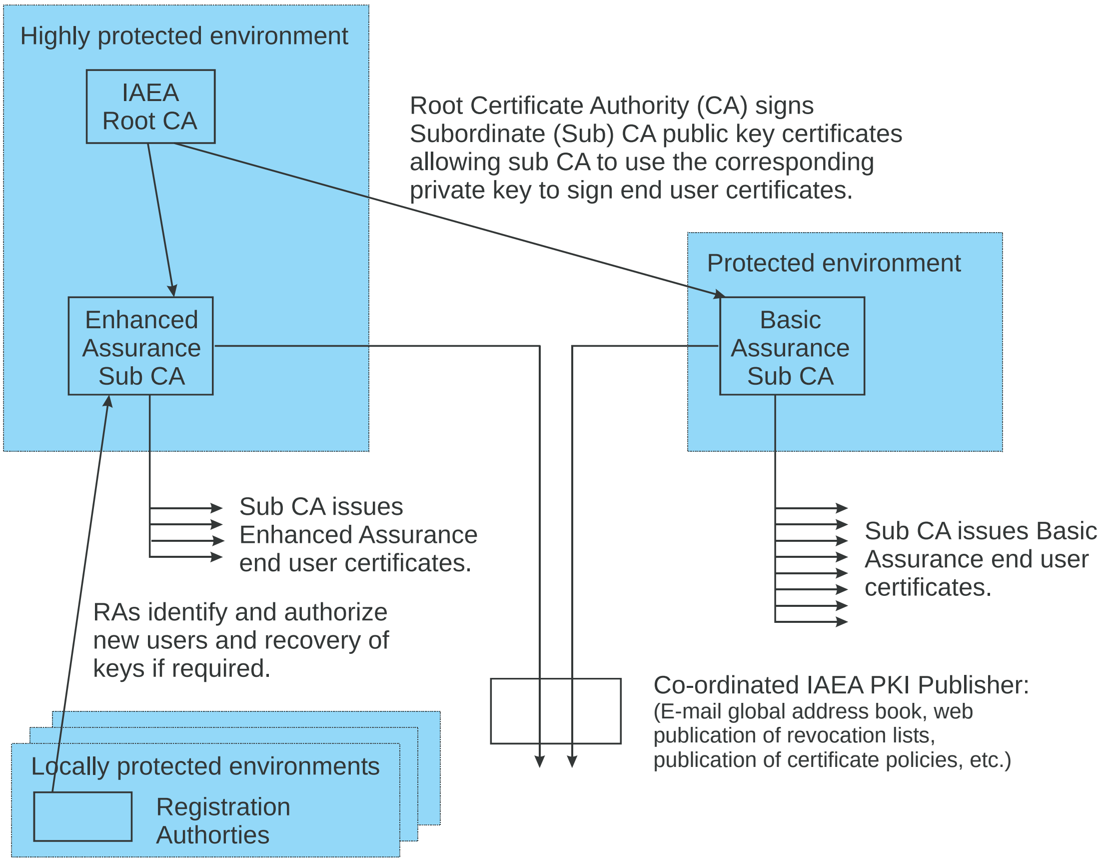

Figure 1. Proposed Target IAEA PKI Architecture.

Draft certificate policies have been developed and are under review by the Agency. Technical systems have been identified and a pre-production configuration, using COTS PKI software, established to validate the procedures. This architecture will provide a standards-based framework which can be leveraged by Agency secure applications which use public key cryptography. Work is currently ongoing to refine the policies and practices, and to deploy the technical systems.

The pre-production system, suitable for the issue of Enhanced Assurance certificates to a small range of applications, is currently operating on a trial basis. It is expected that initial deployments of full production environments will occur in the spring of 2007.

Also in pre-production testing is a “secure mailbox” system to support the electronic receipt of facility operator supplied information, and to return secure time-stamped replies to the operator and to the corresponding state system of accounting and control. This system, under development by the SGIT is deploying a standards-based electronic mail approach which will address the previously identified

[4] IAEA public web servers requiring certificates will likely continue to use commercially purchased certificates since this allows external public users to validate these certificates to existing public trust anchors.

requirements for the automated transfer of operator information [9]. This secure mailbox system will be one of the first applications to utilize keying material managed by the PKI.

5. Certificate Policies

Currently under consideration are the certificate policies associated with the Enhanced Assurance certificates needed to support safeguards applications. Currently five policies are under active development:

- *Confidentiality, Enhanced Assurance Level* – to define a policy for the management and use of certificates containing public keys used for encryption. Certificates issued under this policy are for use by applications such as electronic mail where private key recovery is required to support data recovery should the decryption private key be lost by the user.
- *Digital Signature, Enhanced Assurance Level* – to define the management and use of certificates containing public keys used for signature verification, to provide the associated identification, authentication and integrity protection of the signed data. Certificates issued under this policy are for verifying the signature on electronic mail or for verifying the identity of persons accessing electronic services. Private Key recovery is not supported under this policy[5].
- *File Encryption, Enhanced Assurance Level* – to define a policy for certificates used to provide confidentiality protection for local file encryption applications such as for inspector laptop computers.
- *Server Authentication, Enhanced Assurance Level* – to define a policy for certificates issued to IAEA Intranet servers[6].
- *Device Authentication, Enhanced Assurance Level* – to define a policy for certificates issued to devices to support device authentication, for example to support secure virtual private networking and embedded computer appliances.

Separate policies are defined so that the key management lifecycle requirements can be separately tailored to each application. Each certificate policy is documented using the format defined in RFC 3647 as previously indicated.

The first three certificate types are provided for use by human users. The last two are for use by devices. The corresponding applications will automatically select the certificate and keys to be used based on information within the certificate[7]. For example, when signing electronic mail, the mail client automatically uses the private key corresponding to the signing certificate. For encryption, the mail client uses the public key from the confidentiallity certificate of the intended recipient. The selection of the correct certificate and key is automatic and does not rely on human user intervention.

The Certificate Authority established to issue certificates according to these Certificate Policies must use procedures which are designed to address the corresponding requirements (as identified in the policies). These procedures are documented in the Certification Practice Statement (CPS) document which defines the procedures used by the PKI to meet the requirements of all of the certificates it issues. This document also uses the format defined in RFC 3647, although unlike the Certificate Policy documents which are normally made public, the CPS may be maintained in confidence since it may describe procedures and techniques which should not be made public.

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[5] Key recovery is not required since the loss of the private key only means new signatures cannot be created. Recovery from private key loss is accomplished by issuing a new private key and certificate. Old signatures can still be validated using the old certificate. By avoiding the requirement for key recovery (and corresponding key archive/escrow), and allowing the end user to generate the key, it is possible to avoid the need to transport the private key outside the end user's environment, resulting in enhanced protection for the user's signing key.

[6] These certificates are for use with internal IAEA servers accessed via the IAEA Intranet. Servers providing services to the public are expected to use commercially issued server certificates.

[7] The certificates include both a "permitted use" attribute and an explicit identifier pointing to the corresponding certificate policy. Most current COTS applications use the permitted use attribute to select the certificate.

6. Conclusions

Risks to the security of safeguards data, the need for more rapid and secure communications with Member States, and the growing use of systems with distributed security functions based on public key cryptography, have emphasized the need for a comprehensive and common means to generate and manage keying material. A common Public Key Infrastructure (PKI) must support levels of protection appropriate to the risks and threats to safeguards applications, while at the same time being compatible with a framework which can address Agency wide security requirements.

The Department of Safeguards has established a plan to deploy a PKI for the life cycle management of the electronic keys and public key certificates required for safeguards applications. The approach is being co-ordinated with the Department of Management to ensure that an integrated approach is taken appropriate for eventual Agency-wide use.

The number of applications requiring support is growing and the availability of a standard PKI will ensure that a secure and common infrastructure for key management is available. Absent such an infrastructure, each application would need to develop their own solution, resulting in increased cost and the risk that inadequate attention is paid to its design, potentially resulting in lower security.

Industry has adopted the X.509 standard as the specification for the content of public key certificates, and the use of this and similar international and de facto standards will ensure that the widest range of applications can be supported. Key material managed by such a standards-based PKI will offer enhanced security properties and will reduce dependency on vendor-specific encryption and digital signature products.

ACKNOWLEDGEMENTS

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Development of a safeguards verification method and instrument to detect pin diversion from Pressurized Water Reactor (PWR) spent fuel assemblies

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1. Introduction

A technical safeguards challenge has remained for decades for the IAEA to identify possible diversion of nuclear fuel pins from Light Water Reactor (LWR) spent fuel assemblies. In fact, as modern nuclear power plants are pushed to higher power levels and longer fuel cycles, fuel failures (i.e., "leakers") as well as the corresponding fuel assembly repairs (i.e., "reconstitutions") are commonplace occurrences within the industry. Fuel vendors have performed hundreds of reconstitutions in the past two decades, thus, an evolved know-how and sophisticated tools exist to disassemble irradiated fuel assemblies and replace damaged pins with dummy stainless steel or other type rods.

Various attempts have been made in the past two decades to develop a technology to identify a possible diversion of pin(s) and to determine whether some pins are missing or replaced with dummy or fresh fuel pins. However, to date, there are no safeguards instruments that can detect a possible pin diversion scenario to the requirements of the IAEA. The FORK detector system [1][2] can characterize spent fuel assemblies using operator declared data, but it is not sensitive enough to detect missing pins from spent fuel assemblies. Likewise, an emission computed tomography system [3] has been used to try to detect missing pins from a spent fuel assembly, which has shown some potential for identifying possible missing pins but this capability has not yet been fully demonstrated. The use of such a device in the future would not be envisaged, especially in an inexpensive, easy to handle setting for field applications.

In this article, we describe a concept and ongoing research to help develop a new safeguards instrument for the detection of pin diversions in a PWR spent fuel assembly. The proposed instrument is based on one or more very thin radiation detectors that could be inserted within the guide tubes of a Pressurized Water Reactor (PWR) assembly. Ultimately, this work could lead to the development of a detector cluster and corresponding high-precision driving system to collect radiation signatures inside PWR spent fuel assemblies. The data obtained would provide the spatial distribution of the neutron and gamma flux fields within the spent fuel assembly, while the data analysis would be used to help identify missing or replaced pins.

Monte Carlo simulations have been performed to help validate this concept using a realistic 17x17 PWR spent fuel assembly [4][5]. The initial results of this study show that neutron profile in the guide tubes, when obtained in the presence of missing pins, can be identifiably different from the profiles obtained without missing pins. Our latest simulations have focused upon a specific type of fission chamber that could be tested for this application.

2. Methodology

In order to study the effect on missing or replaced spent fuel pins, simulation studies were done using a Monte Carlo code MCNP5 [6]. The fuel assembly modeled was the Takahama-3 17x17 PWR spent fuel assembly [7], which was loaded with 248 UO_2 fuel pins, 4.1 w/o U-235 enrichment, 16 UO_2-Gd_2O_3 pins (2.6% wt U-235 and 6 w/o gadolinium) and 25 water rods The assembly was irradiated for three cycles with a power of 38.6 W/gU, and cooled for 2 years. The depletion of the assembly was achieved using MONTEBURNS [8] to approximate the isotopic distribution after operation at end of cycle (EOC) and after two years of cooling.

Figure 1 below shows a diagram illustrating the 39 independent regions depleted in MONTEBURNS, in which the color red highlights primarily non-depletable regions of water and the guide tubes (larger diameter circles). 1/8 bundle symmetry was used for the depletion process taking advantage of its symmetry. The fuel assembly was also assumed to have reflective boundary conditions surrounding the outer surface of the bundle.

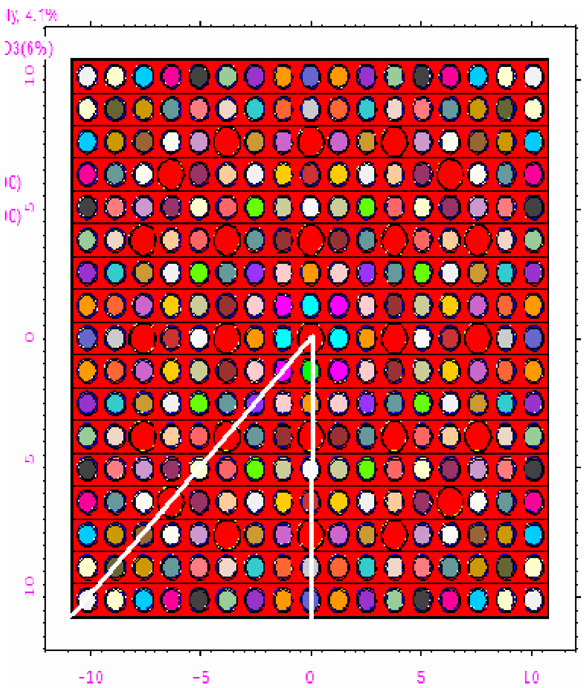

Figure 1. MCNP5 Visual Editor Image of the Takahama-3 17x17 PWR Bundle.
Note that 1/8 of the fuel were used for depletion taking advantage of its symmetry.

2.1. *The Source Term Distribution*

Separate MCNP5 cases were run for neutron and gamma studies. For the neutron flux source in these assemblies, we targeted the Cm-244 distribution in the assembly. This is because for typical commercial power spent fuel assemblies, the neutron flux inside spent fuel assemblies is expected to be dominated by the spontaneous fission neutrons from Cm-244 after two years of cooling time. Accordingly, the pin-by-pin neutron source strengths were established in the bundle in proportion to the Cm-244 relative accumulation. The neutron source was sampled by the Watt fission spectrum and divided in 23 groups between 1.0E-05 and 20MeV, plus a total count.

2.2. *Preliminary Results*

Results from pin-diverted cases were compared against cases with all spent fuel pins present, and the absolute difference and percent difference were calculated. Using standard error propagation, relative errors were calculated for both quantities. A key indicator was whether the differences observed were greater than could be accounted by the margin of error of the results. Assuming Maxwell statistics a hypothetical 60-second count was constructed from the MCNP5 flux. The absolute error was calculated as the square root of the count, and the relative error of the difference from the case where all spent fuel pins are present was calculated.

The preliminary Monte Carlo simulation studies showed that indeed two dimensional neutron data, when obtained in the presence of missing pins, have data profiles distinctly different from the profiles obtained without missing pins. Replacing a single spent fuel pin in the assembly resulted in detectable differences in the neutron flux greater than the designated threshold in at least one energy group for most of the guide tubes, as summarized in Table 1. Substitution of a pin by fresh fuel pin or Fe pin can show the difference in neutron measurement up to 2%. Substitution of two pins by fresh fuel pin can show the difference in neutron measurement more than 4.%. Figures 2,3 and 4 illustrate diagrams of the total neutron flux perturbations due to a corner and central pin diversion, providing some evidence of possible detectability.

Table 1. Summary of Neutron Results.

Pin replaced	MCNP5 statistics	Maxwell statistics
(10,9)	> Threshold in 7 tubes	> Threshold in 13 tubes
(17,1)	> Threshold in 3 tubes	> Threshold in 5 tubes

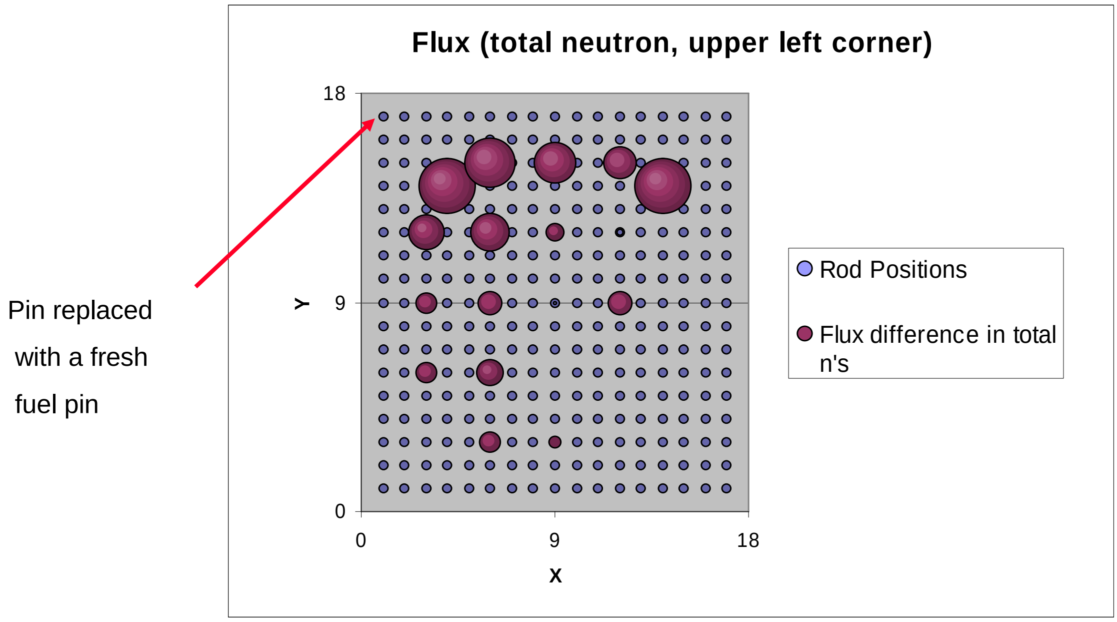

Figure 2. Neutron Flux Perturbation in Guide Tubes due to a Corner Pin Diversion.

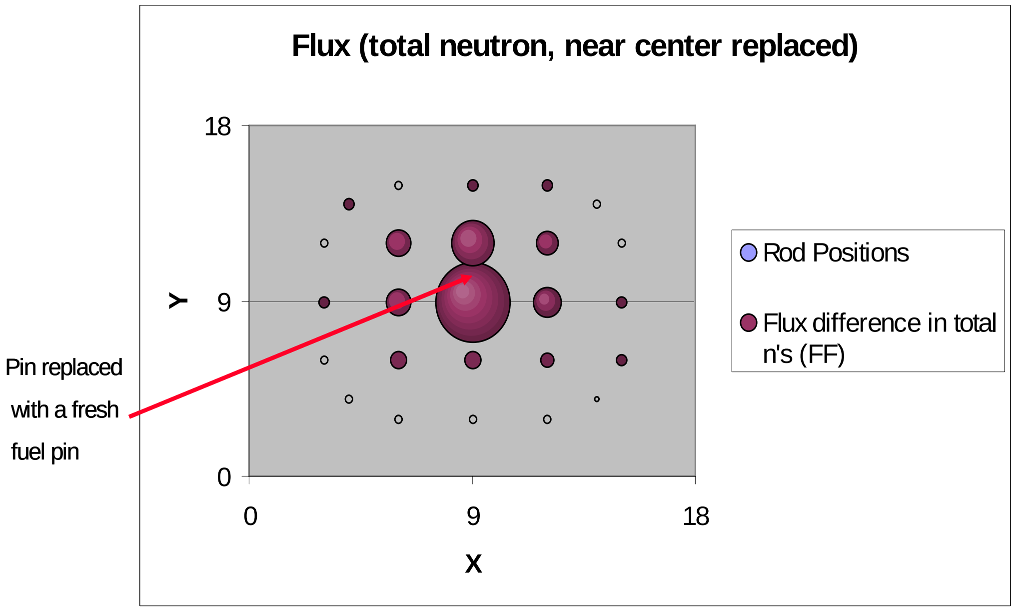

Figure 3. Neutron Flux Perturbation in Guide Tubes due to a Central Pin Diversion.

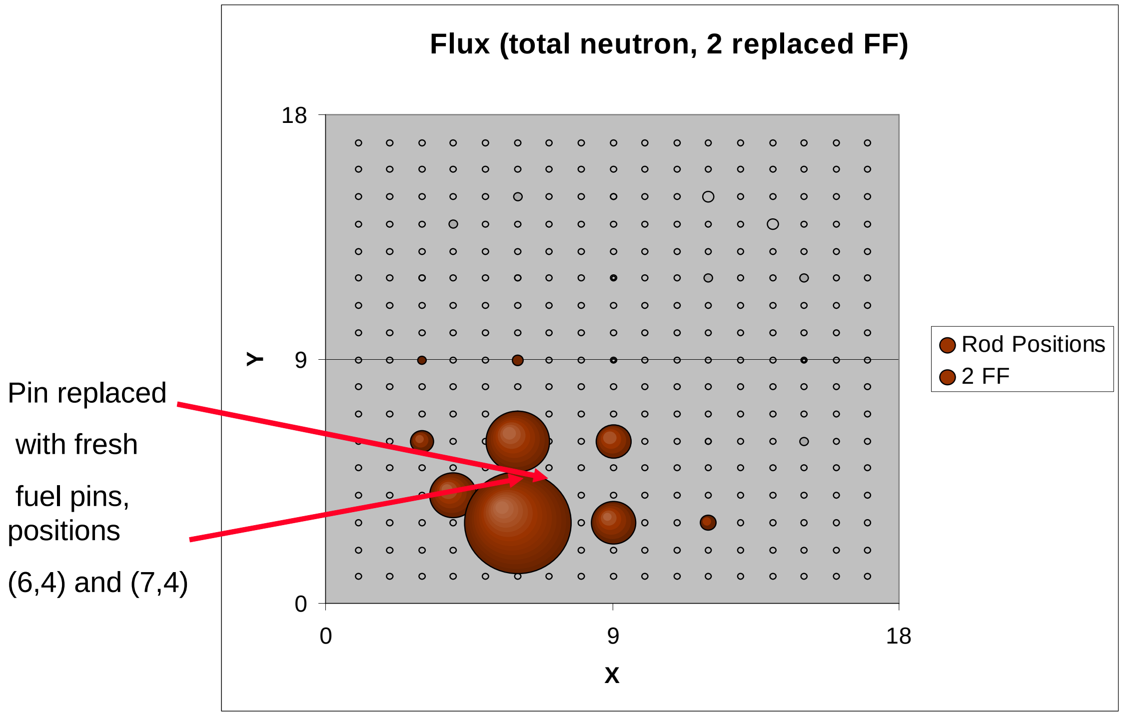

Figure 4. Neutron Flux Perturbation in Guide Tubes due to Two-Pin Diversion.

3. Recent Results with Fission Chamber Model

Our early simulations focused primarily on MCNP tallies of neutron and gamma flux profiles within the guide tubes, and in assessing whether distinguishable differences could be detected in the gamma or neutron flux at each of the energy groups or bins selected. However, in practice, available radiation measurement tools do not usually have the luxury of finely divided energy tallies. Thus, we proceeded to model a typical fission chamber such as the LND 30753 shown in Figure 5 as a diagram and as an MCNP model, side-by-side, with the general specifications provided in Table 2.

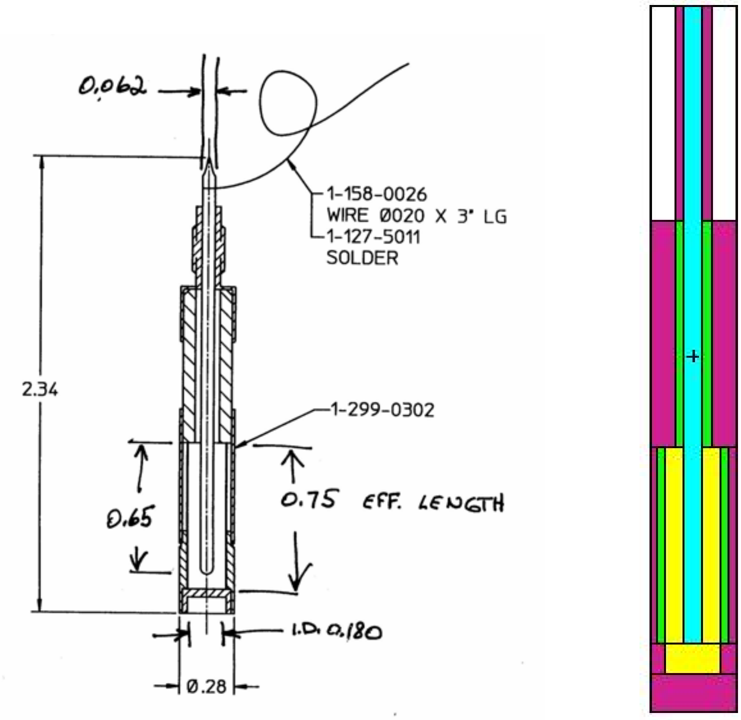

Figure 5. LND 30753 Fission Counter.

Table 2. General Specifications of LND 30753 Fission Counter.

Maximum Diameter (inch/mm)	0.28 / 7.1
Effective Diameter (inch/mm)	0.18 / 4.6
Maximum Length (inch/mm)	2.34 / 59.4
Sensitive Length (inch/mm)	0.75 / 19.1
Cathode Material	Nickel
Fill Gas	Argon
Fill Pressure (Torr)	760
Connector	Flying Lead

Figure 6, below, Shows cross-sectional views of the MCNP model of this fission chamber inserted into the central guide tube of the assembly previously described.

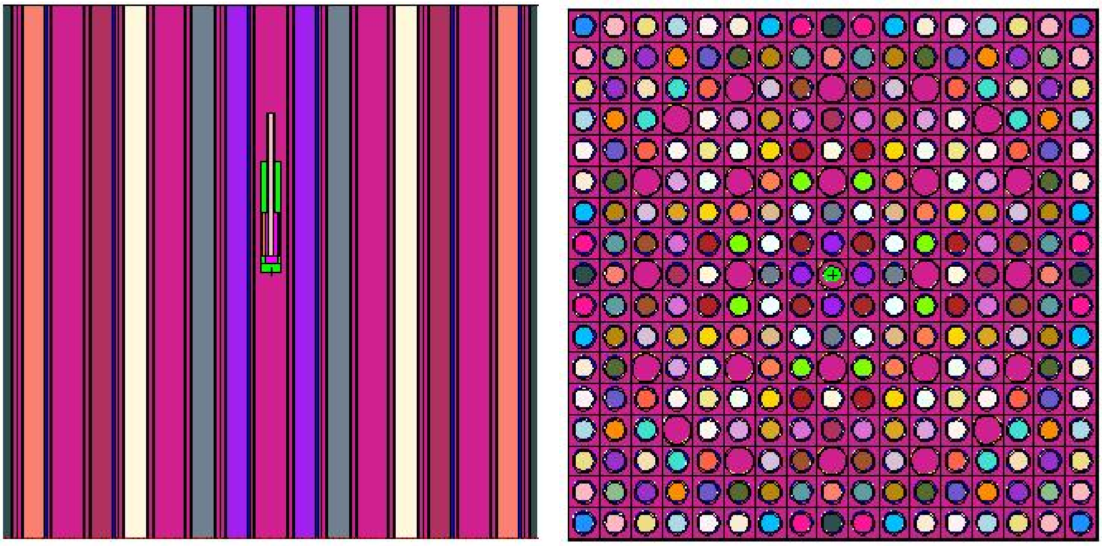

Figure 6. Cross-Sectional View of LND 307 Fission Counter in Central Guide Tube.

Likewise, Figure 7, illustrates the expected impact of the detector's presence within the guidetube upon the measurements.

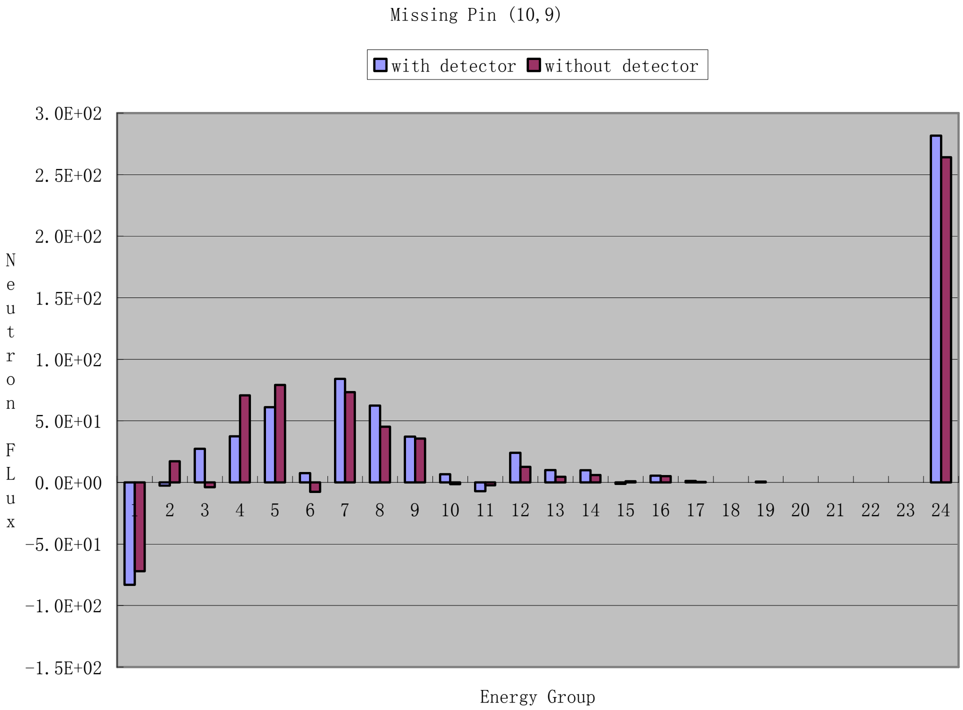

Figure 7. Neutron Flux Spectrum with and without Fission Chamber.

4. Conclusions and Future Work

Monte Carlo simulations have been performed to help validate this concept using a realistic 17x17 PWR spent fuel assembly. The preliminary results of this study show that neutron profiles in the guide tubes, when obtained in the presence of missing pins, can be identifiably different from the profiles obtained without missing pins.

There is still much work to be done in this area to establish a real experimental test. Ongoing activities include:

- Uncertainty analysis related to the operational and cooling history, and the type of PWR assembly (15x15 or vintage models), in particular, assessment of asymmetric depletion upon detection ability.

- Assembly depletion in 3D using TRITON and/or MCNPX/CINDER'90

- Monte Carlo analyses performed in 3D

- Study of detector design and efficiencies associated, such as study of thin fission chamber position within guide tube, and axial displacement of chamber, multiple detectors (cluster).

ACKNOWLEDGEMENT

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Is the FORK detector a partial defect tester?

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Abstract. Validation calculations for the simulation methodology of FORK detector measurements first showed in some cases discrepancies as high as 10% between calculations and measurements. Additional calculations have been done to find the reason for these discrepancies. Acceptable results were obtained when the He-3 and ionisation tubes were moved a few centimeters backwards, thereby placing the backends of the tubes at the same level instead of the centres of the tubes. The observed discrepancies were now at maximum 4%, while the majority of the discrepancies was less than 1%. The assumed positioning errors are considered as likely to happen and are accepted as a cause for the previously observed large discrepancies. The simulation methodology is now considered as reliable and can be used for simulations of measurements on spent fuel assemblies. This paper describes the results of the simulations of spent fuel assembly measurements with the FORK detector and assesses the applicability of proposed additional FORK measurements for spent fuel characterisation.

1. Introduction

A Coordinated Technical Meeting on Spent Fuel Verification Methods held in Vienna in March 2003 stressed the need for partial defect testers for spent fuel in both wet and dry storage.
A candidate for partial defect testing is the FORK detector (FDET). The performance of the FDET as a partial defect tester was investigated experimentally in a Joint Task for the IAEA by the Finnish, Swedish and Belgian Support Programmes. Fission chambers in the FDET were replaced by He-3 tubes and spent fuel was simulated with fresh MOX-fuel. It was concluded that the FDET was not suitable as a partial defect tester, since fuel pin removal could be not be detected in an unambiguous way for various possible scenarios [1, 2].
Later analysis of the results revealed the possibility to interpret the measurement results unambiguously, provided that some additional measurements at 90° are performed.
It was decided to perform MCNP calculations for some of the configurations that were investigated during the Joint Task for validation of the simulation methodology and to investigate the applicability of the proposed additional measurements for an unambiguous partial defect test. This work is performed under task BEL A 1493 of the Belgian Support Programme to the IAEA

2. Measurement principles of the FDET

The FDET has two arms that are placed around a PWR or BWR fuel assembly (see figure 1). Each arm contains three detectors: one ionisation chamber for measuring gammas and two fission chambers for measuring neutrons (see figure 2). The fission chambers are placed closest to the assembly. The detectors are connected by long cables via a watertight tube to the counting electronics. The fuel assembly and the detectors remain under water during the measurement.
A neutron measurement is performed for checking the consistency of the declared burnup. For long cooling times the neutrons are assumed to originate mainly from ^{244}Cm. The measured countrate is corrected for decay of ^{244}Cm (half life 18.1 years) by using the operator declared cooling time of the assembly. The corrected neutron countrate is related to the burnup by the following expression:

$$N_c = \alpha(BU)^{\beta}$$

where $N_c = N_m e^{0.0383 \times T_c}$, N_m is the measured neutron countrate, T_c is the declared cooling time (in years), BU is the burnup of the assembly to be verified (in MWd/tM) and α and β are fitting parameters, with β between 3 and 4

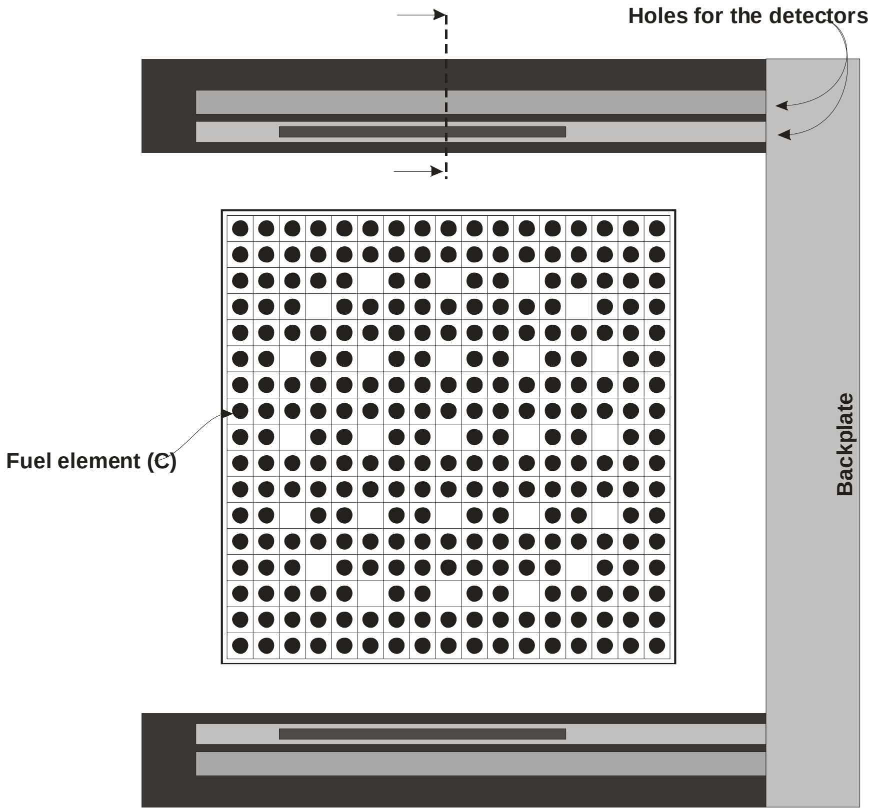

Figure 1. Horizontal cross-section of the FDET with a 17x17 PWR fuel assembly

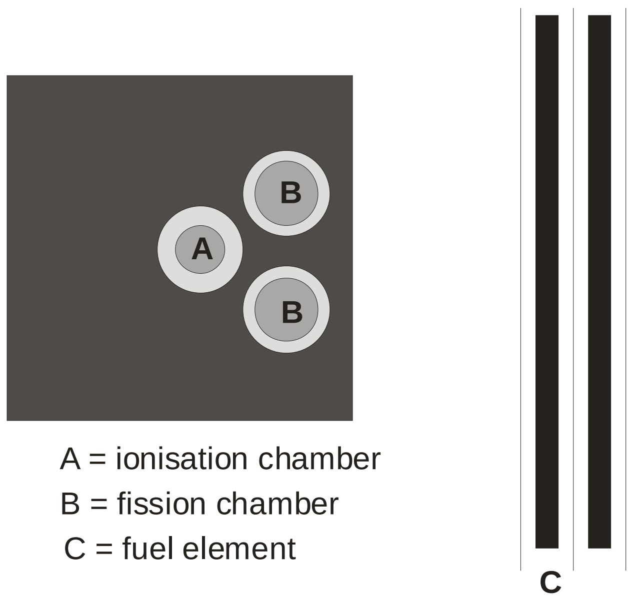

Figure 2. Vertical cross-section of an arm of the FDET, showing the position of the three detectors

The gamma measurement is performed for checking both the declared cooling time and burnup.
The cooling time check is performed by fitting an expression of the form

$$\frac{G_m}{BU} = a(T_c)^b$$

where G_m is the measured gamma signal and a, b are fitting parameters, with b between -0.5 and -1.2

An individual gamma measurement is checked for consistency by comparing it to a fitted curve.

The decay corrected gamma measurement G_c is used for a consistency check of the burnup. This is done via the relation

$$G_c = c \times BU + d$$

where c and d are fitting parameters.

The physical explanation of these relations lies in the fact that the neutrons from an irradiated fuel assembly with sufficiently cooling time originate mainly from ^{244}Cm, an activation product of ^{238}U. The production of ^{244}Cm is not a linear function of the burnup, but a higher order power due to the increased formation of mother isotopes with increasing burnup.
The gammas for long cooling times are mainly due to the amount of ^{137}Cs present in the fuel and are therefore linearly proportional to the burnup (after correction for decay of ^{137}Cs during cooling).

3. Previous work and validation of calculation method

Previous calculations reported in [3] revealed that for some configuration large discrepancies of up to 10% existed between measured and calculated neutron and gamma signals. Additional calculations have been performed in order to investigate the influence of a possible positioning error of the FDET or the individual gamma and neutron detectors. Both the measured and the calculated signals for the configurations with missing pins were normalised with respect to the full assembly.

$$R_{m,c} = \frac{A_{m,c}(conf.X)}{A_{m,c}(conf.32)}$$

Where configuration 32 is the full assembly and configuration X an configuration with missing pins. The indices m and c stand for measured and calculated, resp.

Acceptable results were obtained in the case that the detectors were shifted in such a way that the back-end of the detectors were in the same line in stead of the center of the detectors. The results of these calculations are shown in figure 3. The observed discrepancies are at maximum 4%, while the majority of the discrepancies is less than 1%.

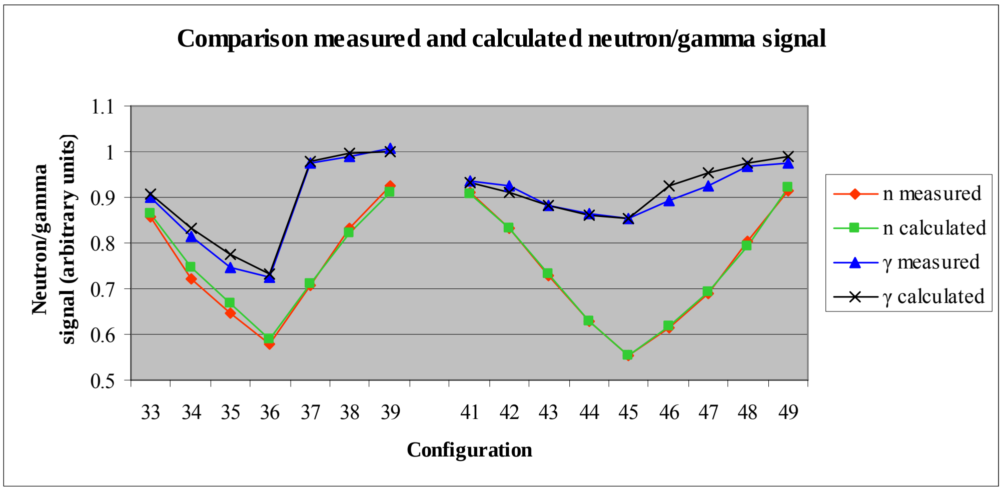

Figure 3. Comparison between measured and calculated neutron and gamma signals. Both measured and calculated signals have been normalised to the signal of the full assembly. The neutron detectors were shifted 2.7 cm and the gamma detectors 6 cm compared to their standard position.

On basis of these results the conclusion was drawn that the used calculation method was valid for FDET measurements on PWR fuel assemblies with missing fuel pins. It should be noted that the validation could only be performed for measurements on fresh fuel, whereas the calculations presented here deal with spent fuel. This is discussed in the following section.

4. Description of input for spent fuel calculations

4.1. Fission products

A list has been made of the most relevant gamma-emitting fission products and it appeared that after a cooling period of 5 years most of these fission products have decayed, except ^{137}Cs. This fission product has been taken into account in the gamma source of the MCNP input. The main gamma energy used is 662 keV. ^{137}Cs is assumed to be distributed homogeneously in the fuel, which in reality is not the case. However, for these calculations the assumption is fair, since the results are always compared to the reference full assembly. Possible calculation errors due to inhomogeneous distribution will be greatly cancelled by this comparison.

For short cooling periods other fission products are still present in the fuel. ^{95}Zr, ^{103}Ru and $^{106}Ru/Rh$ will contribute to the gamma radiation. However, since the energy of the main gamma peaks for these isotopes are all in the range 500-700 keV, the absorption of the gammas will not be significantly different from the absorption of the 662 keV gamma of ^{137}Cs. The results of the calculations will therefore also be applicable to short cooling periods.

4.2. Neutron emitters

Based on an ORIGEN II calculation of a PWR fuel assembly with a cooling time of 10 years and a burnup of 25 GWd/tM the relative amount of ^{242}Cm and ^{244}Cm in the fuel has been estimated [4]. These two isotopes provide more than 90% of the neutrons emitted by the fuel assembly.

For short cooling periods the contribution of the Cm isotopes to the total neutron signal will be significantly less. However, the behaviour of the neutrons produced by spontaneous fission of other isotopes will not be significantly different from those originating from the Cm isotopes. This might be less the case for neutrons produced by α, n reactions. Again, the fact that the results are compared to calculations on the reference fuel assembly will largely cancel out any errors.

4.3. MCNP version

Version 2.5.0. of MCNPX has been used for the calculations.

4.4. Fuel assembly

The modelled fuel assembly was a 17x17 PWR UO_2 assembly with an enrichment of 3.5%. The fuel pins contained apart from the UO_2 ^{137}Cs, ^{242}Cm and ^{244}Cm. The water in the fuel pond contained 2270 ppm boron.

In the annex the configurations are described that have been calculated. The configurations with missing rows or columns of fuel pins have also been measured in the experimental campaign described in [2, 3] with fresh fuel pins. These configurations have been used for the validation of the calculation methodology.

Additionally calculations have been performed on the configurations 69-76 as listed in figures A.2, A.3 and A.4 in the annex.

4.5. FORK detector

The modelled parts of the FDET consisted of the polyethylene arms and backplate. The holes in the FDET contain air. The modelled ionisation chamber was an LND 52 113 and the fission chambers an RS-P6-0805-134.

5. Results of spent fuel calculations and discussion

The calculations performed in the framework of this study aimed at checking the applicability of the FDET as a partial defect tester. Essentially this means that in case of a diversion of fissile material the neutron and gamma signal deviate significantly from their normal behaviour. Therefore a reference assembly has been taken (a full 17x17 PWR assembly with no diversion of fissile material) and several assemblies with different degrees of diversions.

Based on the expressions for N_c and G_c a diversion will be observed when the ratio N_c/G_c^4 will change significantly.

Figure 4 shows that this is the case for some diversion scenarios (conf. 33 to 36), but not for all. It was suggested that an additional measurement of the fuel assembly at 90° could overcome this problem.

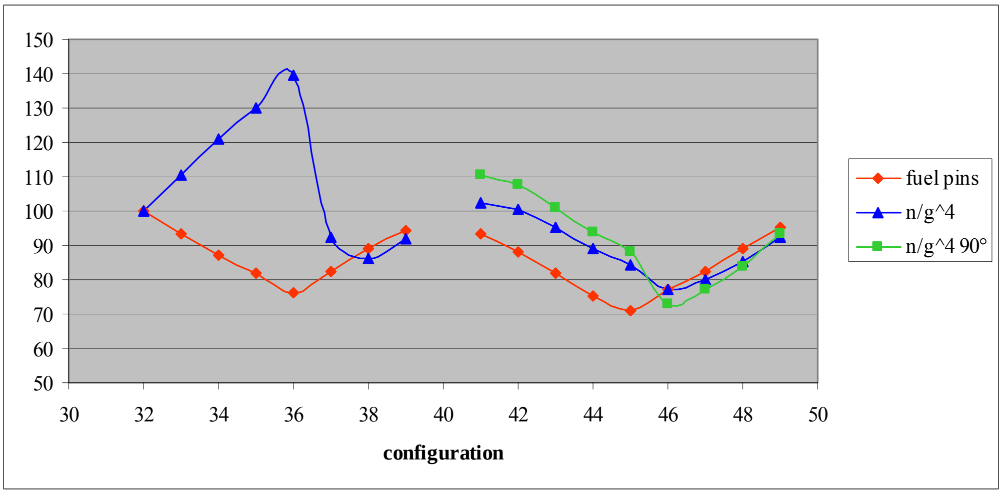

Figure 4. Results of calculations of N_c/G_c^4 for configurations with missing rows or columns of fuel pins.

The results of calculations on the additional measurement configuration at 90° for configurations 41-49 are also shown in figure 4 (squares) and it is clear that a measurement at 90° will not solve the problem of insufficient indication for diversion in all cases. Especially for those configurations where rows of pins are left between rows that are diverted, the change of the ratio N_c/G_c^4 when measuring at 90° is not significant for various configurations.

However, a partial defect test for a spent fuel assembly is defined as 50% of the fuel pins missing. All of the calculated configurations mentioned above have less than 50% pins missing, so conclusions about the partial defect testing capacities of the FDET cannot be drawn only on basis of these calculations.
Configuration 45 is the configuration with the highest diversion of pins, i.e. 30%.

Therefore some configurations have been calculated with 50% missing pins and a geometry that is symmetric to a 90° measurement. The configurations have been chosen so that the missing pins were mainly at the inside or at the outside of the assembly, while also an intermediate configuration has been taken. Figures A.2, A.3 and A.4 in the annex show the studied configurations.

Figure 5 shows the results of these calculations. It appears that configurations 74 and 76 show an extremely high value for N_c/G_c^4 whereas the other configurations show values that are significantly lower than 100%. The high values for the ratio N_c/G_c^4 for configurations 74 and 76 can be explained by the fact that only one or a few pins are in the column close to the detectors and that gammas from the pins in the following columns are shielded by the pins in the closer columns. A closer examination of the calculation results confirmed this explanation, showing that taking away a column of fuel pins that is shielded by a closer column does not affect much the observed gamma signal. Shielding by the water is far less effective than shielding by fuel pins. This explains why configurations 74 and 76 show extremely low values for the gamma signal. The influence of these low values is amplified by the fourth power in the expression N_c/G_c^4.
The neutron signal is much less affected by the form of the various configurations and its influence on the ratio N_c/G_c^4 is much less.

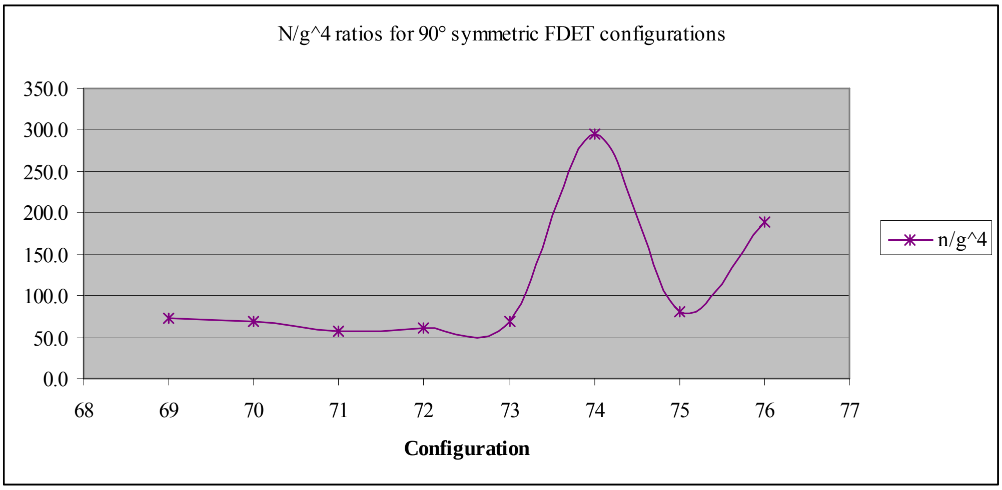

Figure 5. Results of symmetric configurations 69-76 with 50% fuel pins missing**.**

To conceal a diversion a diverter should obtain a ratio N_c/G_c^4 of 100% in order to have a signal similar to the full configuration. From the obtained results it is clear that a configuration with 50% fuel pins missing can be designed that will result in a ratio N_c/G_c^4 of 100%, since both configurations with a ratio higher and with a ratio lower than 100% exist. An intermediate configuration will therefore exist resulting in a 100% ratio of N_c/G_c^4.
These results should be verified and compared with IAEA FDET measurement data and additional checks that are performed during real inspections.

6. Conclusions

Simulations with MCNPX have shown that 17x17 PWR fuel configurations can be designed so that a twofold FDET measurement at the normal measurement position and a position at 90° will not notice a difference between a full assembly and an assembly with 50% fuel pins missing.

Previously reported measurements with fresh fuel have been used to validate the used calculation methodology.

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Annex. Description of the different fuel configurations

The fuel configurations that were calculated consist of a standard 17 by 17 PWR fuel assembly with 264 fuel pins and 25 water holes.

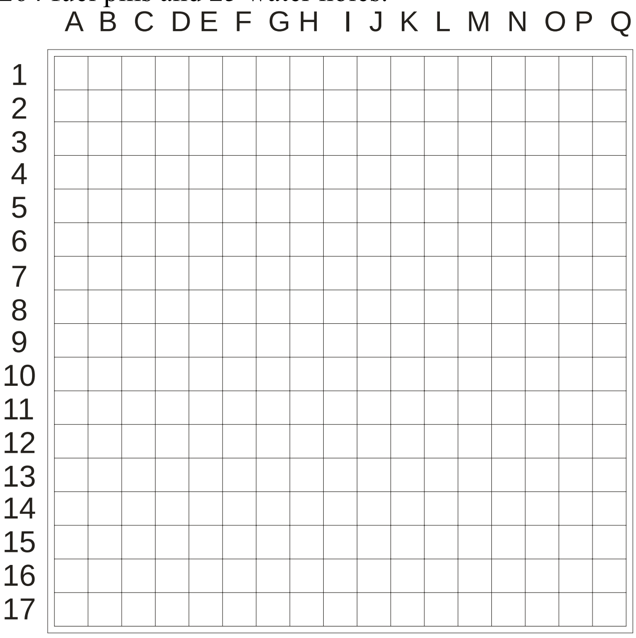

Figure A.1. Scheme of 17x17 PWR fuel assembly indicating missing rows and colums of fuel pins. The arms of the detector are parallel to the columns, indicated here by numbers (and not by letters). The backplate of the detector is located on the right.

Columns are indicated by a number, rows by a letter. The detectors of the FDET were placed parallel to the columns.
In the investigated configurations fuel pins were removed from either columns or rows. The following table gives the configuration number and which rows or columns were removed in that configuration.

Configuration	removed columns	# pins left	Configuration	removed rows	# pins left
32	none, full conf.	264	41	A	247
33	17	247	42	A, C	233
34	17, 16	230	43	A, C, E	216
35	17, 16, 15	216	44	A, C, E, G	199
36	17, 16, 15, 14	201	45	A, C, E, G, I	187
37	16, 15, 14	218	46	C, E, G, I	204
38	15, 14	235	47	E, G, I	218
39	14	249	48	G, I	235
			49	I	252

Table A.1. Rows (letters) or columns (numbers) that have been removed in the investigated configurations

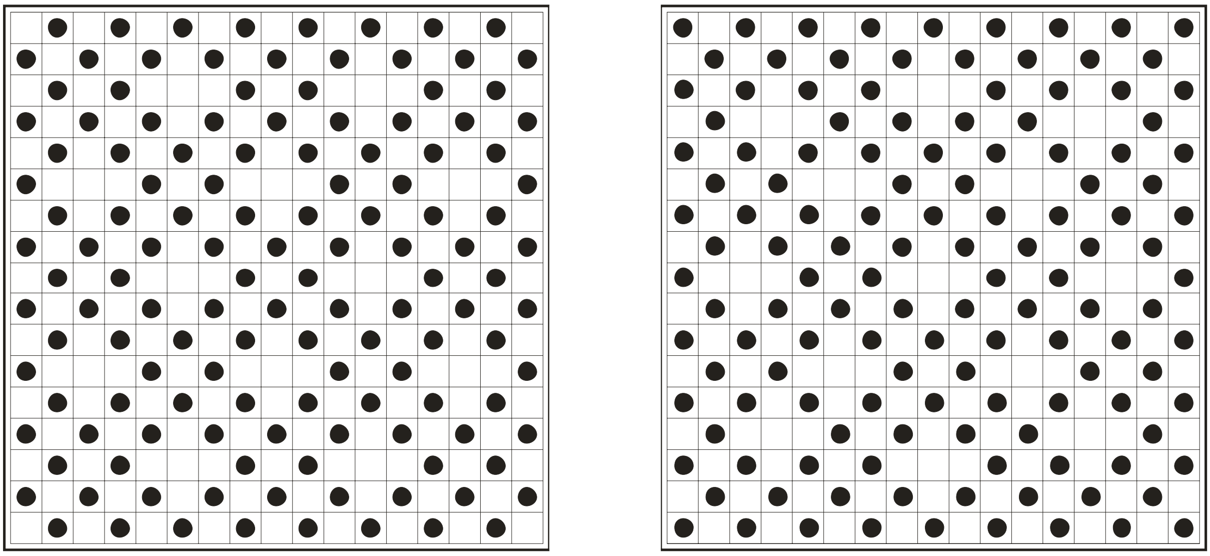

Figure A.2. Configuration 69 and 70, where 50% of the fuel pins has been removed in a homogeneous way.

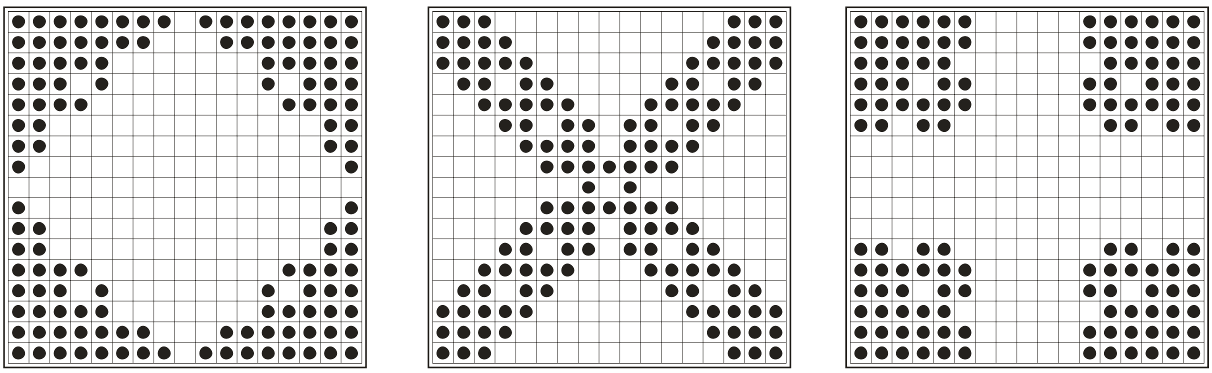

Figure A.3. Configuration 71, 72 and 73. 50% of the fuel pins has been removed with the purpose to have pins close to the detectors (71), far from the detectors (72) and an intermediate configuration (73).

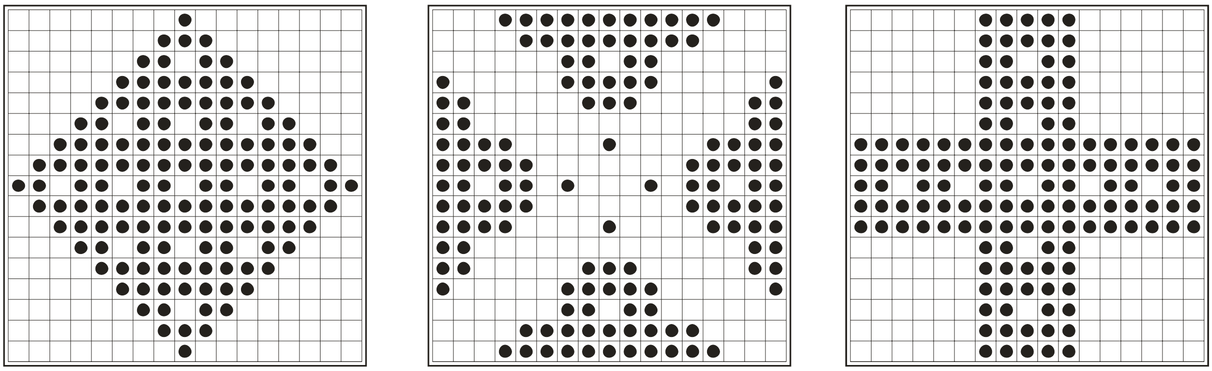

Figure A.4. Configuration 74, 75 and 76. 50% of the fuel pins has been removed with the purpose to have pins far from the detectors (74), close to the detectors (75) and an intermediate configuration (76).

Partial-defect detection using a digital Cerenkov viewing device and image processing

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Abstract. Fuel assemblies with substituted Zircaloy rods, water rods, missing fuel rods and substituted uranium rods were measured using a digital Cherenkov viewing device. Substituted Zircaloy rods can be detected visually from the DCVD image if they are located in the periphery of the assembly but are more difficult to detect when located in the central area of fuel assemblies. Empty positions can be detected from the DCVD image because of the higher light intensity in adjacent water gaps. For GE12S and Atrium 10B fuel assemblies, the missing fuel rod area is much brighter than the light above the partial-length fuel rods. The light intensity can be 88 percent higher and is easily detected. The light intensity is slightly lower than the partial-rod area when the missing rod is in the periphery of the assembly where fewer near-neighbour rods are present to generate the Cherenkov light. Data from the substituted uranium rods are still being analyzed.

1. Introduction

The digital Cherenkov viewing device (DCVD) is used by IAEA inspectors to non-intrusively verify low-burnup spent fuel with 10 000 MWd/t U and 40 years cooling time. The digital nature of the DCVD makes image processing possible. Image enhancement and noise reduction assist in the identification of fuel designs and offers the potential to automatically detect missing and substituted fuel rods (partial defect). A preliminary study has investigated the ability of the DCVD to detect missing rods, substituted Zircaloy rods, water rods and substituted uranium rods. Additionally, the Cherenkov characteristics of fuel assemblies with partial-length fuel rods were examined.

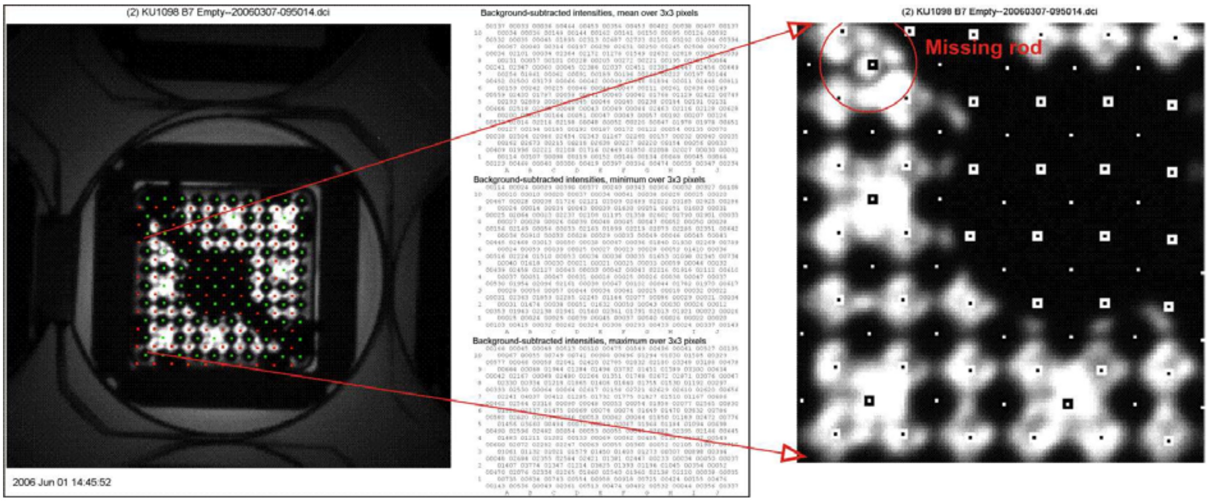
2. Instrumentation

A redesigned DCVD [1] that incorporated a new camera model (a custom Andor Technology DV887) was used for these measurements. The new camera uses a 512×512 pixel (16×16 µm), back-thinned, lumogen-coated CCD with on-chip gain, hence, an electron-multiplied, charge-coupled device (EMCCD). The chip has a quantum efficiency of 28 percent at 310 nm. This DCVD is now integrated into a single unit. The system uses a Mini-ITX computer board (VIA EPIA) with a 1.3-GHz processor and 1 GB of memory. A 2-GB compact flash card replaces the standard hard drive to improve the ruggedness of the system. Images are saved on a removable 1-GB USB flash drive. A 20-cm touchscreen monitor (800×600 pixels) is used for image display. The camera head is mounted on a yoke and can be moved in and out using a lead screw assembly. Control cables are used to tilt the camera head in the x and y direction to centre the image in the display. A laser pointer mounted on the camera provides orientation in the fuel pond. A spirit level located on the top of the camera indicates a horizontal position. The DCVD is easily mounted on a fuelling machine or on a walking bridge (Fig. 1). The integrated DCVD has significantly fewer parts than the previous version [2]. The complete system can be moved from the storage case to the bridge, powered up, the detector cooled to –60°C and have an image captured in less than five minutes. This is a significant improvement over the previous version, which took from 15 to 25 minutes to set up.

FIG. 1. The DCVD mounted on a walking bridge.

3. Data reduction process

The raw image files are read and processed using an automated script. For each image, four specific rod locations are identified manually to determine image scaling and rotation; the script interpolates the remaining rod and gap positions. It then calculates the mean, maximum and minimum values within a 3×3-pixel region surrounding the selected pixel. An approximate background value is obtained from the darkest pixel in the image histogram. Background-subtracted intensity values are used throughout this paper. Data processing results are displayed both graphically and numerically as shown in Fig. 2.

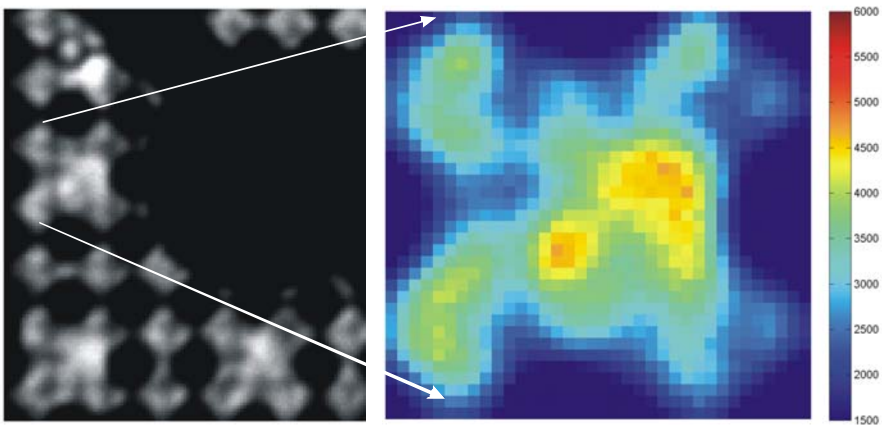


(a) Example of data output (b) Pixels selected for intensity analysis

FIG. 2. Data reduction output and pixel selection for quantitative analysis.

Fig. 2b shows a detailed view (covering 7×7 rods) for a typical assembly. Each interpolated 3×3-pixel region is identified with a black-framed white pixel (rods) or a white-framed black pixel (gaps). Data given for rods and water gaps are the minimum and maximum intensity values, respectively. As there is little variation from pixel to pixel within the 3×3 regions of interest for rods and water gaps, the choice of using the minimum or maximum value is straightforward. The selection of intensity values, however, for missing rod regions is slightly more complicated. The number of candidate pixels is reduced due to the "bolt hole" structure (dark circle around the very centre of the marked area in Fig. 2b) of the top plate that leaves about 12 pixels from which to choose the 3×3 region of interest. In this case the maximum pixel value is recorded.

For fuel assemblies with partial-length rods the situation is even more complex. A magnified, contrast-enhanced image of a missing rod and a partial-rod region is shown in Fig. 3a. The magnification in Fig. 3b is even higher. There is considerable variation in light intensity in the area of the partial-length rods. There are very few adjacent pixels that have similar intensity values and the differences between pixels can be large. The selection of a single "correct" pixel is problematic. Therefore, we have chosen to report the mean value for these regions. The variations in intensity are attributed to a number of factors such as the short water column above the partial rod, the perspective view of adjacent rods, and the (especially topmost) spacer grids.



(a) Contrast-enhanced image (b) False-colour intensity image

FIG. 3. Partial-length rod region of an Atrium 10B assembly.

4. Measurement results

4.1. Document convention

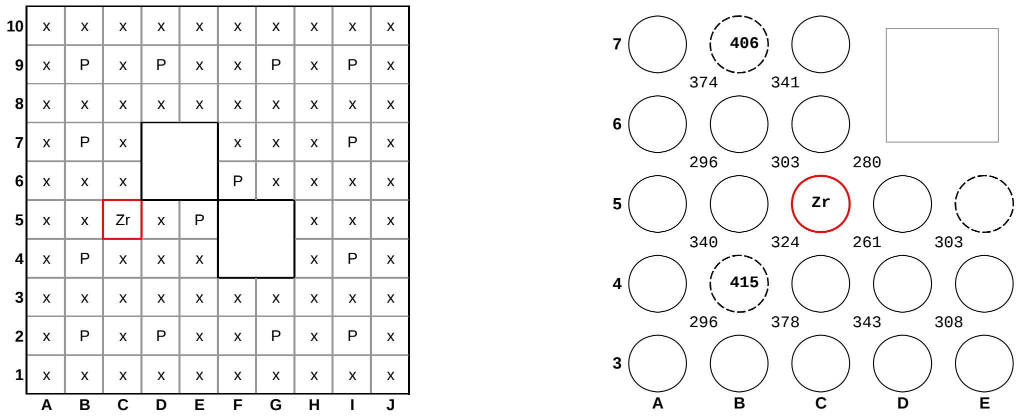
Image data within this report are generally presented in the following order: a visible light photograph where available, a DCVD grey-scale image and a DCVD false-colour image (cf. Fig. 6). Visible light photographs are presented to provide a context for the DCVD images.

Fuel rod matrices and the intensity data for selected regions of interest are presented in the format indicated in Fig. 4. Rows and columns are labelled by the letters and numbers shown and individual rods are identified using that convention, e.g. position A3. The rod type in the figures is indicated with the following code:

X	a fuel rod	**Zr**	a zircaloy rod
E	an empty position	**W**	a water rod
P	a partial-length fuel rod		

Rods of interest are identified using colour coding. Assembly identification numbers are given in the figure captions. For quantitative discussions, sub-regions of the assembly are shown in the format indicated in Fig. 4b. Intensity values, mostly for water gap areas, are shown between the rod positions. Intensities for partial-length or missing rods are shown within the dashed line rod outlines. Larger sub-regions containing no rods (or water rods) are indicated with a light rectangular outline. Intensity values are not reported for rods or gaps under the handle.

Because the DCVD images are digital, they are scaled to show optimum brightness and contrast. Care must be taken when comparing images. A long-cooled fuel image can look as bright as an image of a short-cooled fuel assembly. The only valid measure of glow intensity is the actual counts in the original data file.



(a) Rod matrix (b) Intensity

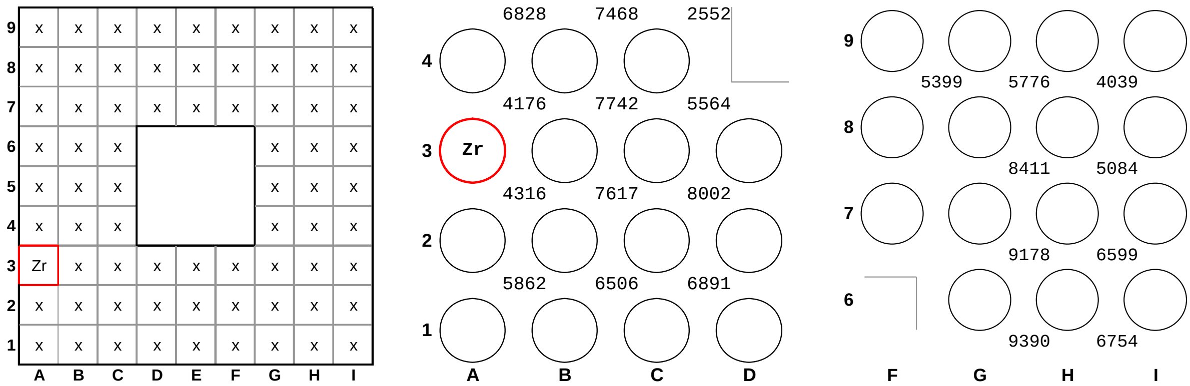
FIG. 4. Rod matrix and intensity example.

4.2. Fuel assemblies with Zircaloy rods

Zircaloy rods can be used in fuel assemblies to replace failed fuel rods or to act as a flux trap to modify the reactivity of the fuel. The irradiated Zircaloy rod does not contribute significantly to the total Cherenkov light because the gamma ray flux from the Zircaloy is low. Zircaloy (about 98 percent zirconium) has a very low neutron absorption cross-section and is mainly activated to ^{95}Zr, which has a half-life of only 64 d. The Cherenkov light intensity surrounding the Zircaloy rod should have a lower light intensity because the rods do not emit a significant level of gamma radiation compared to spent fuel rods.

Atrium 9, irradiated Zircaloy rod at A3

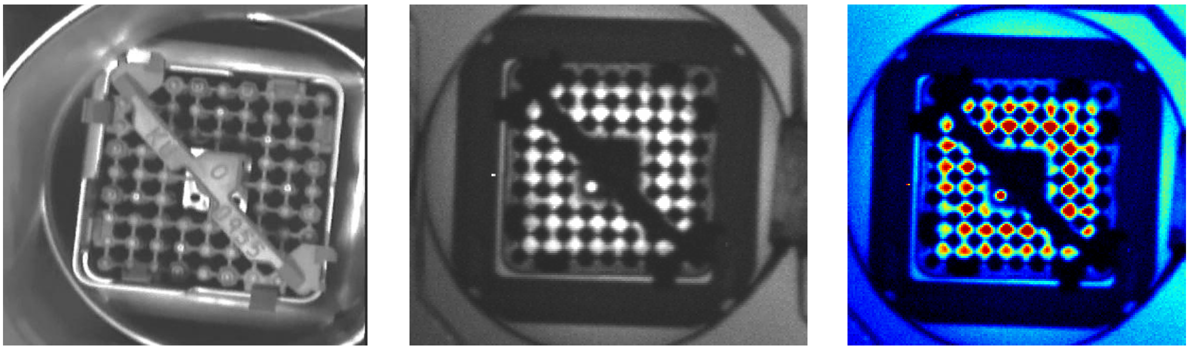
This assembly has a burnup of 38 100 MWd/t U and a cooling time of 2.4 years. The original fuel rod at A3 (Fig 5a) was removed because of failure due to debris wear fretting. A Zircaloy rod was substituted and the fuel assembly was returned to the reactor for irradiation to its design terminal burnup.



(a) Rod matrix (b) Intensities lower left (c) Intensities upper right

FIG. 5. Assembly KU0955 with a substituted zirconium rod at A3.

In the visible light photograph of this assembly (Fig. 6a), there appears to be a top plate tab between A2 and A3, on either side of the Zircaloy rod. This tab, as well as several others visible in the photograph, maintains separation between the fuel channel box and the top plate. Close examination of the Cherenkov light intensity in the grey-scale images shows that the light intensity is reduced around A2 and A3. The DCVD false-colour image shows this difference quite clearly. Intensity values around the Zircaloy rod (Fig. 5b) show a drop of 40 percent or more when compared to the intensities of the adjacent water gaps surrounded by fuel rods. A comparable normal fuel rod at I7 (Fig. 5c) shows a higher light intensity around I6 and I7 than around the Zircaloy rod.



(a) Visible light photograph (b) Grey-scale image (c) False-colour image

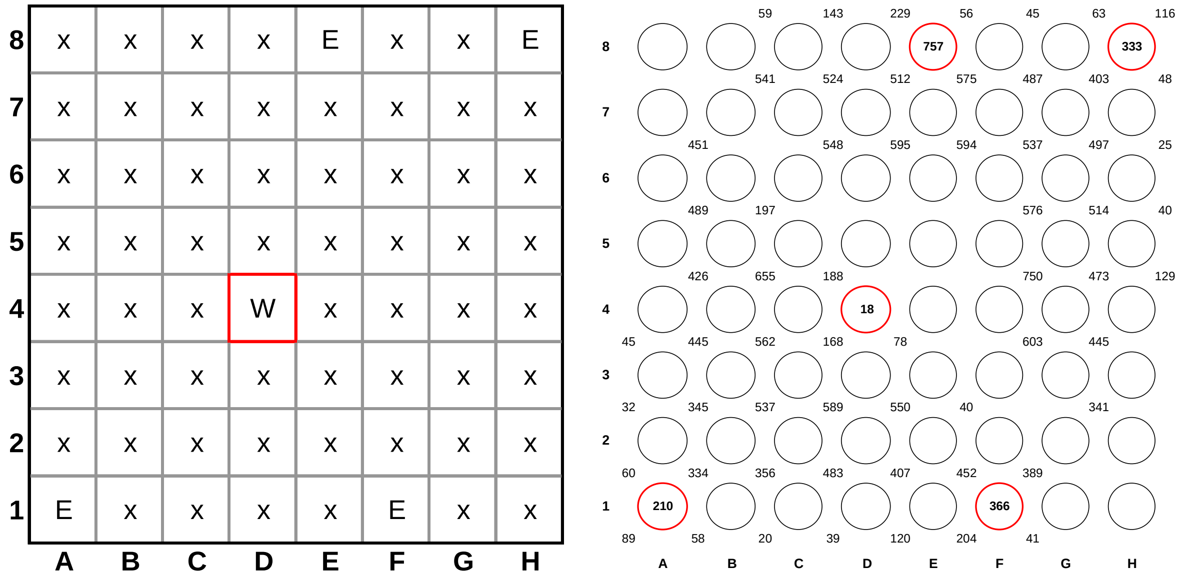
FIG. 6. Images of KU0955 with an irradiated Zircaloy rod at A3.

Another fuel assembly with a Zircaloy rod in the central area of a fuel assembly was also examined (images are not reproduced in this paper). The results showed that the intensity decreases in the water gaps surrounding the zirconium rod but not to the degree reported above.

4.3. Fuel assembly with a water rod and missing rods

The water rod at D4 in this assembly is used to increase the local reactivity to provide a more uniform burnup across the assembly. The DCVD images show a dark area around this rod (Fig. 8), which is confirmed by the intensity data (Fig. 7b). This is the first time a fuel assembly with a visible water rod has been imaged by a DCVD. The observed result is very promising for detecting water rods. However, more studies are required to confirm this characteristic.

Spent fuel assemblies can have fuel rods removed because of fuel failures or the requirement for post-irradiation examination. There are two missing fuel rods at the corner positions A1 and H8 and two more missing rods at E8 and F1 (Fig. 7a). In this case, the rods could not be unbolted from the top and bottom tie plates; the rods were cut out just below and above the respective plates. The end plugs of the removed fuel rods were left in the tie plates.



(a) Rod matrix (b) Intensity data

FIG. 7. Assembly FA6116 with one water rod and four empty positions.

Thus a small diameter rod remains where the fuel rods were removed; the normally expected round holes that identify missing rods are not observed. The missing rod positions can be identified in the DCVD images (Fig. 8) by their anomalous rod structures (smaller diameter) and confirmed by the higher light intensities (from Fig. 7b).

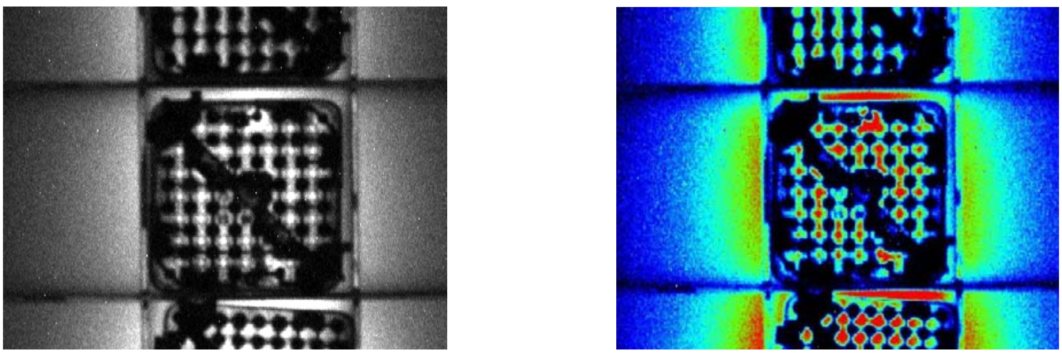

(a) Grey-scale image (b) False-colour image

FIG. 8. Images of assembly FA6116.

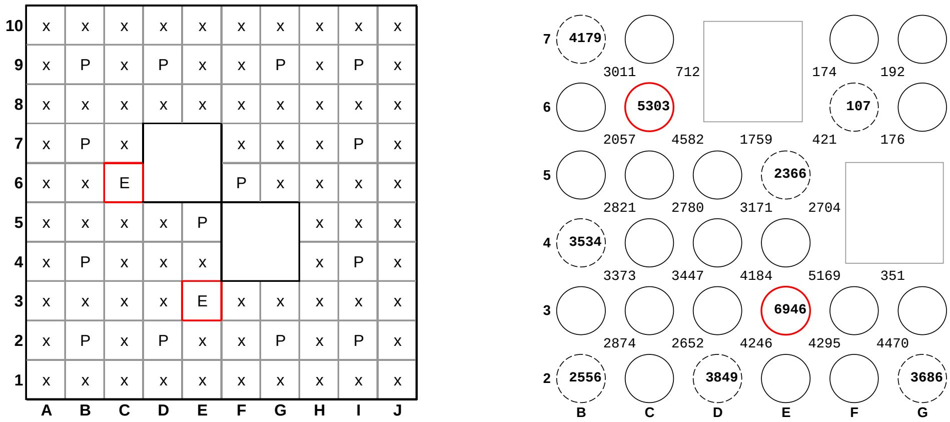
4.4. Fuel assemblies with missing and partial-length rods

Missing fuel rods from the GE12S and Atrium 10B fuel, which are manufactured with partial-length fuel rods, were examined. One objective of this study was to obtain the Cherenkov characteristics of missing rods and to compare them with the Cherenkov characteristics of partial-length fuel rods to determine their differences.

GE12S, missing rods at C6 and E3

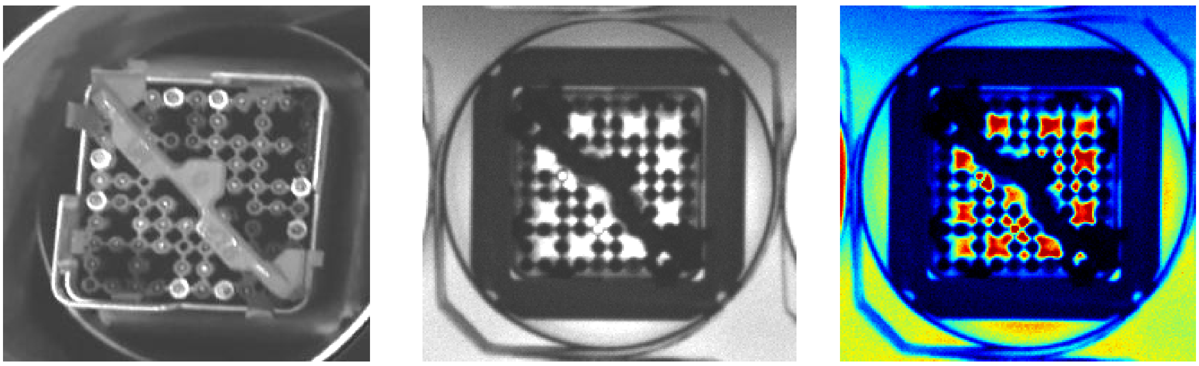
This assembly has a burnup of 42 200 MWd/t U and a cooling time of 3.6 years. Empty mounting holes are noted at C6 and E3 in the visible light photograph (Fig. 10a) at the missing rod locations

(Fig 9a). Similar features are noted at some other positions; however only from the Cherenkov images are those other positions clearly identifiable as containing fuel rods. In addition, replacement nuts that attach the top plate to the fuel rods are visible due to their shiny metallic surface. In both the grey-scale and false-colour images the missing rods are seen as interruptions in the rod pattern, i.e. the normally black rod image is replaced with the bright round disk in the area of the missing fuel rods. The light intensity in the area surrounding the missing rods also appears visually brighter than the light in the partial-length rod area.



(a) Rod matrix (b) Intensity data

FIG. 9. Assembly GN0278 with missing rods at C6 and D3.



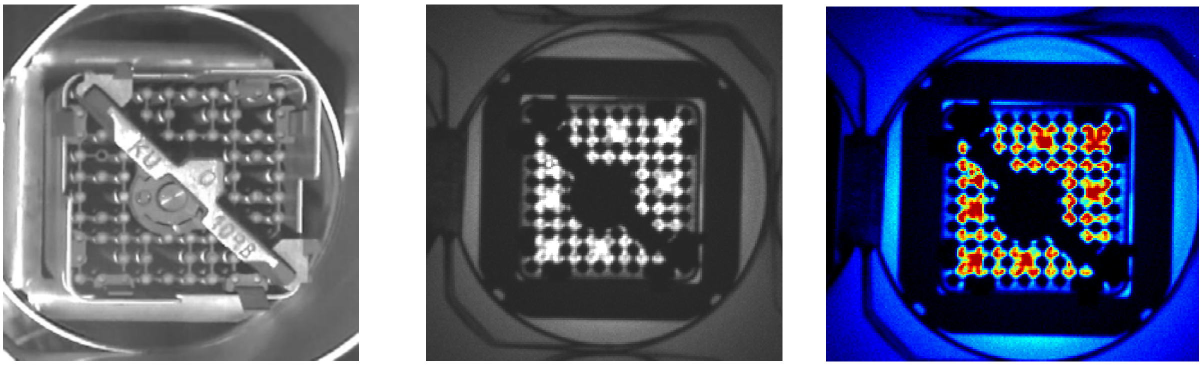
(a) Visible light photograph (b) Grey-scale image (c) False-colour image

FIG. 10. Images of assembly GN0278.

The intensities reported in Fig. 9b show that the Cherenkov light intensity above the missing fuel rods located in the inner area of the fuel assembly are 27 to 88 percent higher than the light above partial-length fuel rods.

Atrium 10B, missing rod at B7

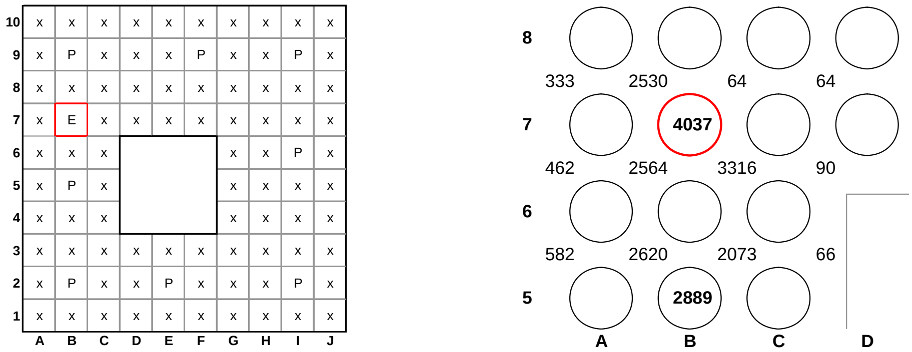
This assembly has a burnup of 39 100 MWd/t U and a cooling time of 1.7 years. The visible light photograph (Fig. 11a) shows an empty mounting hole in the top plate at B7 where the fuel rod should be attached.



(a) Visible light photograph (b) Grey-scale image (c) False-colour image

FIG. 11. Images of assembly KU1098.

The absence of the rod at B7 (Fig 12a) is clearly detected in the grey-scale image; the false-colour image may be slightly harder to interpret. Analysis of the pixel intensity data (Fig. 12b) at B7 shows that the light intensity in the area of the missing rod is approximately 40 percent higher than the partial rod at B5.



(a) Rod matrix (b) Intensity data

FIG. 12. Assembly KU1098 with a missing rod at B7.

5. Discussion

Fuel assemblies with substituted uranium rods that were subsequently irradiated and a single fuel assembly with substituted unirradiated uranium rods were also analyzed. The fuel assemblies with the irradiated, substituted uranium rods and the unirradiated, substituted rods could not be clearly identified in this preliminary analysis. Further studies are required to determine the ability to detect substituted uranium rods.

We have demonstrated that certain single rod defects can be detected, some quite easily, some with difficulty. It seems reasonable that multiple defects would be substantially easier to detect but assemblies of this type have not been available in the facilities visited. It would be useful to locate such assemblies in other facilities. We think that multiple defects amounting to perhaps 10 percent of the rods in an assembly would be relatively easy to detect using this method.

The investigation of fuel assemblies from other types of reactors is needed.

The present procedure for identifying partial defects relies on considerable post-processing capabilities. Future effort is planned to develop software to facilitate identification of partial defects.

6. Conclusion

The following conclusions can be made:

- Zircaloy rods in spent fuel can be detected by careful inspection of the DCVD image
- Quantitative analysis of the image can confirm uranium rod replacements with Zircaloy rods

- Water rods can be detected from the DCVD image and confirmed by quantitative analysis
- Missing fuel rods are readily identified
- Partial-length fuel rods have been characterized
- Substituted uranium rods are more difficult to detect and need further investigation

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Modelling of Cherenkov light emission from BWR nuclear fuel with missing or substituted rods

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Presented by K. Axell

Abstract. Computer simulations of Cherenkov glow from spent nuclear fuel were carried out. Spent nuclear fuel in storage ponds are verified with the help of the Cherenkov viewing device (CVD) and the Digital Cherenkov viewing device (DCVD). The instruments image the Cherenkov glow generated by gamma ray emissions from spent fuel into the water. An attempt to build a realistic digital model of the DCVD image containing partial-length, missing, and substituted rods was made to see if the effects of the deviations from normal can be predicted. It was concluded that partial-length or missing rods in the model was in good agreement with measured data, but replaced rods in the model showed a weaker attenuation of the Cherenkov glow than the observed DCVD images.

1. Introduction

The Cherenkov viewing device (CVD) is routinely used by the IAEA to verify spent nuclear fuel in storage ponds. A new instrument, called the digital Cherenkov viewing device (DCVD) is now available to the IAEA inspectors for difficult cases of older fuel or fuel with low burnup. Both instruments image the ultraviolet Cherenkov light (280-340 nm) generated by fission products in the fuel and the so called collimation effect is used for the fuel/non fuel decision. In an earlier study [3] the origin of the collimation effect and the contribution to the total Cherenkov light from different depths within the fuel was studied. The model showed that the whole fuel volume contributes to the Cherenkov image. The current study is a part of a larger study of the ability of the DVCD to detect individual (or groups of) missing or substituted rods in the fuel. Computer simulations of the Cherenkov light from fuel assemblies containing missing rods, partial-length rods, water rods, fresh fuel rods, or homogeneous zircaloy rods were done.

2. The computer model

The computer modeling was done in two steps.

- First, the lateral distribution of the Cherenkov emission was calculated at one or two representative levels in the fuel. This was done at the Department of Nuclear and Particle Physics at Uppsala University, using the Monte Carlo simulation code MCNP [1].
- Second, the radiation transfer from the whole volume of the fuel element to a DCVD camera was calculated. Here, the emission distributions from step one were used. The calculation was done with the optical ray tracing code Zemax [4].

Both calculations steps are time consuming. Each MCNP simulation (step 1) was based on 7.0×10^9 emitted gamma quanta, requiring about 550 h processor time, distributed on ten personal computer (PC) machines. Each Zemax run (step 2) took about 20 hours on one PC.

2.1 Calculation of the lateral emission distributions

The Cherenkov light emitted from nuclear fuel is caused by electrons traveling faster than the speed of light in the water surrounding the fuel. The main source of these high-speed electrons is from the Compton interaction of gamma rays emitted from the fuel with electrons in the surrounding water. Another source of highly energetic electrons is direct β^- emission. However, these electrons predominantly do not leave the fuel and their contribution to the Cherenkov light emission has not been included in this work. The flux distribution of electrons at energies above the threshold for Cherenkov-light production of 257 keV was obtained by simulations using the established Monte Carlo simulation code MCNP version 4C2 [1]. Only emission of gamma rays from the fuel has been simulated and the induced electron flux distribution from the Compton process was calculated. Based on the electron flux distribution, the emission of Cherenkov light has been modeled.

Simulated gamma-ray spectrum

In the simulations, the gamma-ray spectrum of a spent BWR fuel assembly with a discharge burnup of 36 GWd/tU, an initial enrichment of 2.65 % and a cooling time of 10 years was used. The spectrum is obtained using the ORIGEN-ARP fuel depletion code [2]. It is illustrated in Figure 1. It should be noted that 79 % of the gamma rays have an energy of 662 keV, emitted in the decay of ^{137}Cs. The simulated gamma-ray source distribution in the fuel material was uniform. No gamma-ray emissions from neighboring fuel assemblies are simulated.

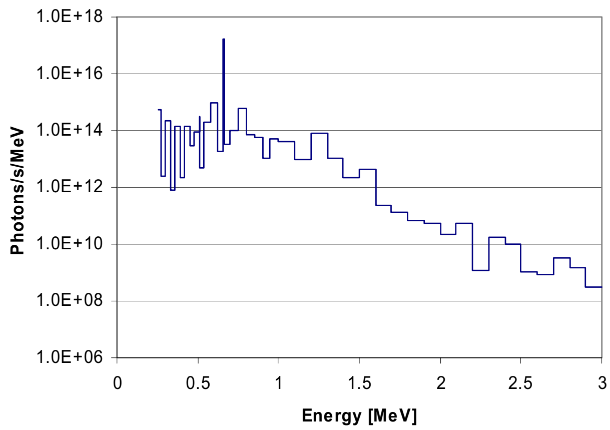


Figure 1. The gamma-ray spectrum used in the MCNP simulations.

Modeling of the Cherenkov-light emission

Cherenkov light occurs when charged particles move through a medium faster than the speed of light in that medium. In this work, the medium is water and the index of refraction is assumed to be 1.34. Accordingly, the electrons must have an energy above 257 keV for Cherenkov light to be emitted. The number of light quanta that are emitted along the track of an electron depends on its kinetic energy E_K. The number is proportional to:

$$\left.\frac{dN}{dx}\right|_{E_k} \propto 1-\frac{1}{\beta^2 n^2} \qquad \text{Eq. (1)}$$

where

$$\beta^2 = 1-\left(\frac{m_0 c^2}{E_{tot}}\right)^2 = 1-\left(\frac{511}{511+E_k}\right)^2$$

(E_k expressed in keV)

The gamma-ray emission from a 5-cm vertical fuel assembly segment was simulated using the MCNP simulation code. On top of this segment an MCNP tally layer is applied where only electrons in water are followed. In the tally, the track length of each electron is analyzed. The selected MCNP tally is divided into 10 energy bins and 64x64 respectively 74x74 geometric bins or pixels for the two fuel types under study, BWR8x8 respectively Atrium9. The geometry has been selected so that each fuel rod is covered by approximately 8x8 pixels centered over the rod. The pixel pattern covering one fuel rod is illustrated in Figure 2.

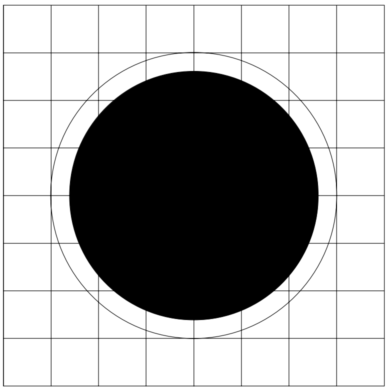


Figure 2. The pixel pattern covering one fuel rod. All rods are covered by an identical pattern.

For each pixel (i,j) and energy bin (E_k), MCNP gives the sum of the simulated electron track length, x(i,j,E_k). When modeling the emission of Cherenkov light, this number has been multiplied by the number of emitted light quanta per track length dN/dx obtained from Eq. (1).The light intensity N(i,j) in pixel (i,j) is given by the sum of the contribution from each energy bin:

$$N(i,j) = \sum_{E_k}\left[\left.\frac{dN}{dx}\right|_{E_k} \cdot x(i,j,E_k)\right] \propto \sum_{E_k}\left[\left(1-\frac{1}{\beta^2 n^2}\right)\cdot x(i,j,E_k)\right] \qquad \text{Eq. (2)}$$

where β^2 is the same as in Eq. 1. For each energy bin, the mean value of E_k in the bin is used.

2.2 Calculation of the radiation transfer

The optical ray tracing program typically traces light rays from a source to a detector. A light ray path can encounter several optical or mechanical objects. The program takes into account effects that these objects have on the light ray such as absorption, reflection, refraction, scatter, and energy loss. The model is built in a manner similar to how models are made in a 3D CAD program. The whole model can be built from standard objects that are included in the software. Examples of objects are lenses, mirrors, detectors, rods, tubes, plates and sources. All objects can be given different properties, such as, position, orientation, transmission value, scattering functions and so on. Figure 3 shows one of the fuel assemblies that was created (Atrium 9). It consists a 9x9 array of 4 meter long fuel tubes, a top plate, a handle, 6 spacers, and a camera unit (not shown), consisting of a lens and a detector. The aperture and focal length of the model lens is the same as the 250 mm lens normally used on the DCVD. The detector is divided into pixels, so an image can be created. The model also includes the storage pond (not shown). The top of the fuel is 9 meters below the water surface and the camera is 2 meters above the water surface. The light source is modeled as a small square plane surface that emits rays upwards in a narrow angle. During the simulation, this light source travels around in the fuel to about 100 000 different positions distributed evenly in the whole volume of the fuel element. From each position a number of rays are emitted according to the local emission strength, obtained as described in section 2.1. All light that reaches the detector is accumulated in the pixels until an image representing the whole volume is created. Most nuclear fuels have a burnup profile along the length axis of the fuel. This was taken into account with weighting factors that varied with the depth.

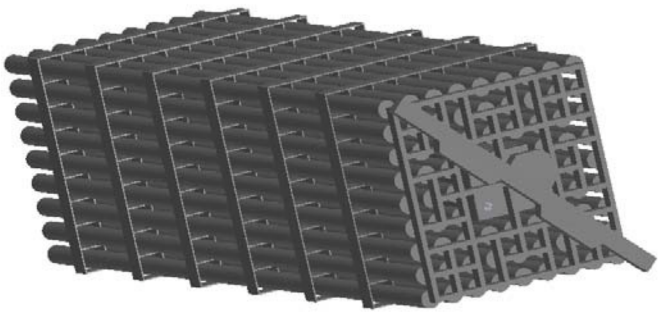


Figure 3. Atrium 9 fuel model.

The fuel model shown in Figure 3 is off center and the fuel length is compressed to make details more visible. In the spent fuel pond, the fuel is vertical instead of horizontal as shown in the figure. The camera (not shown) looks vertically down on the top of the fuel.

3 ABB (now Westinghouse Electric Sweden) 8x8 BWR fuel

The simulation of a simplified 8x8 type of fuel assembly manufactured by ABB was carried out. The parts included in the model were the 8x8 array of fuel rods and six spacers grids. All other parts were excluded to make the effects of substituted parts more visible. The cross-section of the fuel type is illustrated in Figure 4a. All rods have the same radius, 12.25 mm, and the same pitch, 16.3 mm. There is also one water rod in the central part of the assembly (position D4), i.e. a water-filled zircaloy tube of the same dimension as the fuel rods. Figure 4b shows the positions where the substitutions were done.

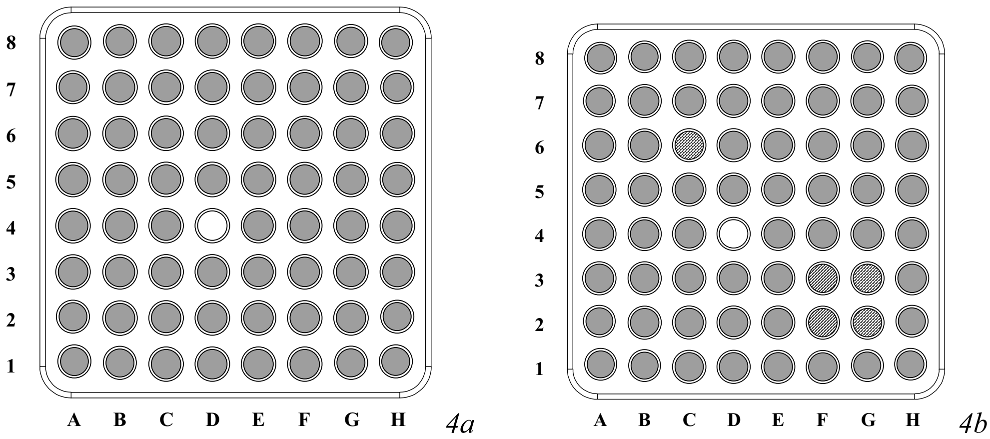
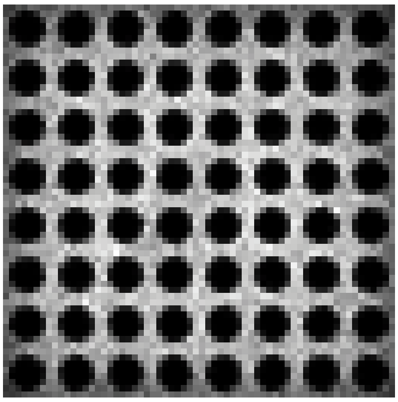
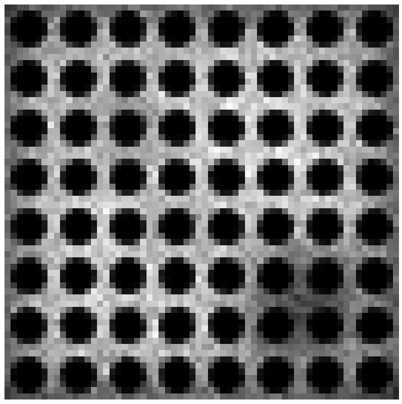


Figure 4. Cross-section of a BWR 8x8 fuel assembly.

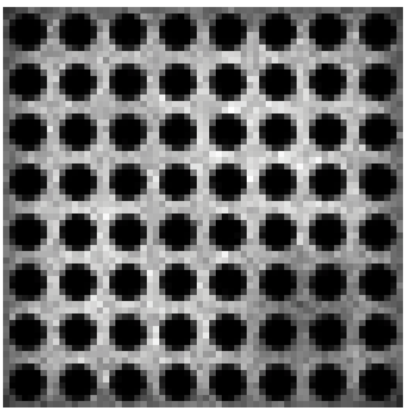
Four runs of simulation step 1 were made. One with the normal fuel and three with all the 5 rods marked in Figure 4b replaced with fresh fuel, water rods, or zircaloy rods. These rods do not contain any radiating material in the simulations. Therefore, it can be noted that the substitutions give rise to relatively strong attenuation of the gamma radiation from the surrounding rods. Because the fuel geometry does not change along the fuel, only one lateral emission distribution map is needed. Figure 5 below shows the emission maps for four different cases.



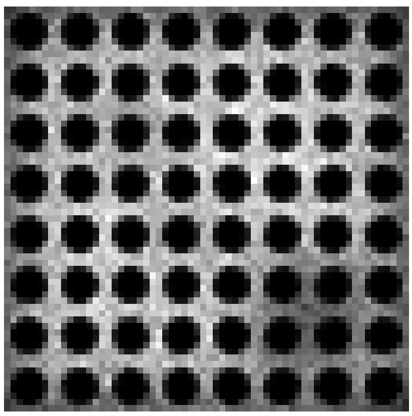
Normal Fuel



Five rods replaced by fresh fuel



Five rods replaced by water rods



Five rods replaced by Zircaloy rods

Figure 5. Lateral cross-section view.

The results show that the Cherenkov emission is weaker around the replaced rods, especially if they are fresh fuel rods. However, it seems that it is more difficult to detect the effects of a single replaced rod but the group of four can be clearly seen. Water rods give the smallest effect. Figure 6 shows the final model images after the radiation transfer has been made. Here, two runs were made for each substitution case: one with the replaced rods black and one with the replaced rods shiny as mirrors. The black color corresponds to irradiated and oxidized rods and the shiny ones correspond to fresh un-irradiated rods. The shiny rods were modeled as perfect mirrors for run time reasons. In reality, there would be both absorption and scattering from the rod surface. From Figure 6, it can be seen that the attenuation of the Cherenkov glow is fainter in the final image than in one of the lateral cross-sections. With black rods, the simulated Cherenkov images all show a dark area around the group of four replaced rods. The single replaced rod (black) is difficult to detect. Shiny rods balance the darkening effect around the replaced rods, so they are very hard to distinguish from the normal ones. In the fresh fuel case the darkening is still there, but is barely visible. With homogeneous zircaloy rods and water channels, the darkening is more or less balanced by the shine of the tubes.

The simulation shows that a small group of black substituted rods can be detected, but if they are shiny if can be more difficult. For the single rod situation the shiny replaced rod is more visible then a black replaced rod.

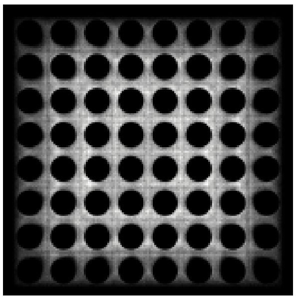
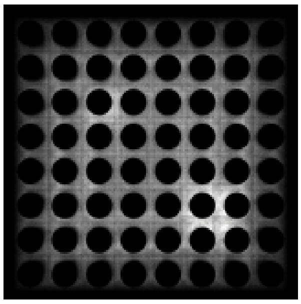
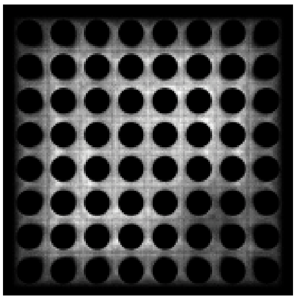
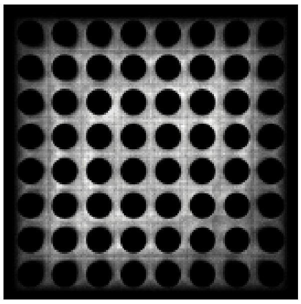
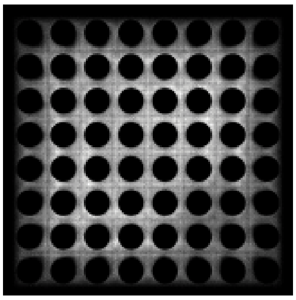
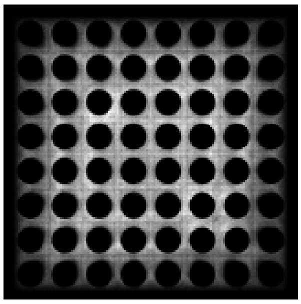
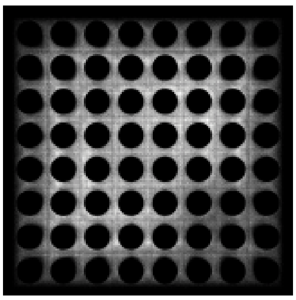
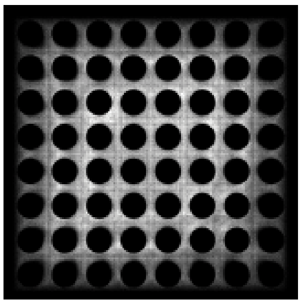
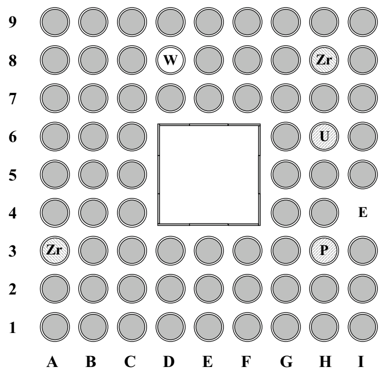
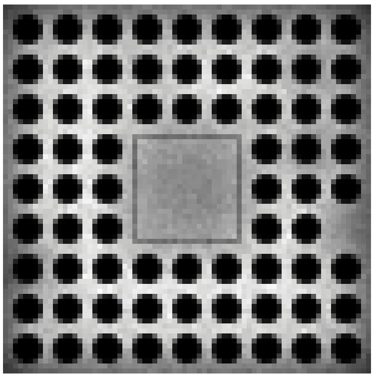
	Replaced rods black	Replaced rods shiny
Normal fuel		
Rods replaced with fresh fuel		
Rods replaced with water rods		
Rods replaced with Zircaloy rods		

Figure 6. Resulting images in the DCVD camera for the simulated cases of fuel rod replacements with black and shiny rods, respectively.

4 Framatome ANP Atrium 9 fuel

An Atrium 9 fuel assembly from Framatome was modeled. The fuel geometry is illustrated in Figure 7a. All rods have the same diameter, 11.00 mm, and the same pitch of 14.45 mm. In the centre of the fuel, there is a square shaped water channel with a side of 38.55 mm (outer) and a wall thickness of 0.725 mm. Figure 7a also shows the rod substitution positions. The coordinates for the substituted rods are: a water rod in (D8), homogeneous zircaloy rods in (A3) and (H8), an unirradiated uranium rod in (H6), a partial-length fuel rod in (H3) and an empty position in (I4). The partial length rod is 2.35 meters long (the top part missing). Accordingly, these rods do not contain any radiating material in the simulations. Figure 7b shows the lateral emission distribution for a cross section below the top of the partial length fuel rod and Figure 7c shows the same at a level above the partial length rod. The partial length rod (H3) is not present at this level.





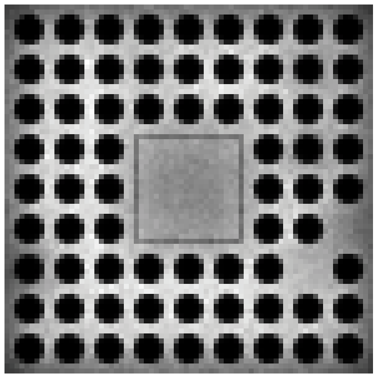
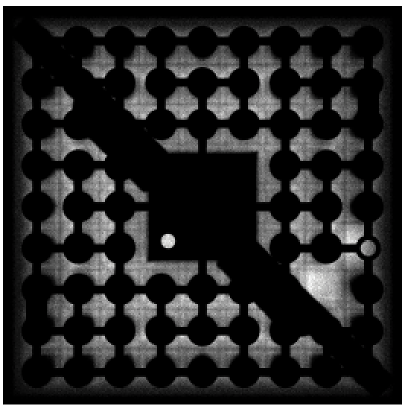


Fig 7a Fuel matrix. *Fig 7b Cross section (below).* *Fig 7c Cross section (above).*

The local emission strength decreases about 15% at the position above the partial-length rod and a bit more (22%) at the position of the missing rod, which may due to its location near the edge. The decrease in Cherenkov intensity around the substituted rods is very low (about 5%) and reaches only a few millimeters out from the rod. Position A3 is an exception. There, the intensity above and below A3 is about 20% lower than the intensity between the neighbors in the same column. The final model DCVD image with all rods black (irradiated) is shown in Figure 8a. Figure 8b shows the same, but here the zircaloy rods (A3, H8), and the uranium rod (H6) are shiny (un-irradiated).



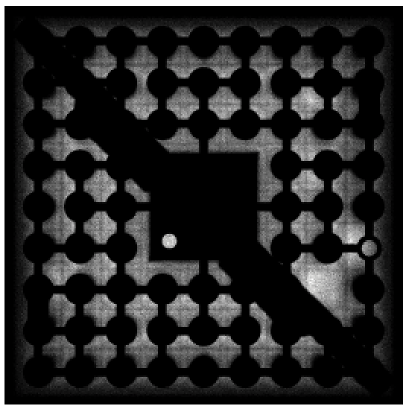


Figure 8a. Model image of Atrium 9, dark rods. *Figure 8b. Model image of Atrium 9, shiny rods.*

Two real DCVD images are shown in Figure 8c (Atrium 9) and in Figure 8d (Atrium 10B). The lower left half of the model fuel, that is, the half below the handle should be directly comparable to the DCVD image of an Atrium 9 fuel. This fuel has a substituted irradiated zircaloy rod at A3 identical to the model. The Atrium 10B has partial and missing rods. In the Atrium 9 case the model shows a small decrease in light intensity around the zircaloy rod at position A3 (10%) but is contrasted in the real image where there is a significant decrease in Cherenkov light. Clearly the model is under predicting this light decrease. In the missing rod case, the model shows a very similar strong local glow spot at a missing rod position, like in a real DCVD image (see position B7 in Fig. 8d). Partial-length rods also look similar in the model and a real image but the limited resolution in the real images makes it hard to compare details (see one of the partial rod positions in Fig. 8d).

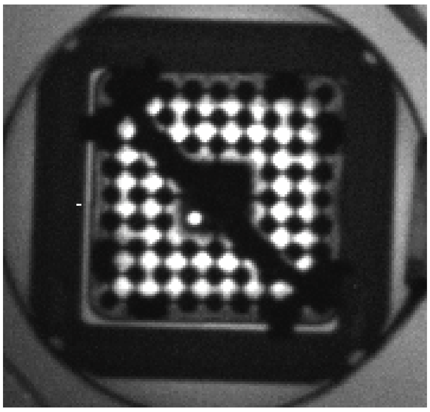


Figure 8c. DCVD image Atrium 9.

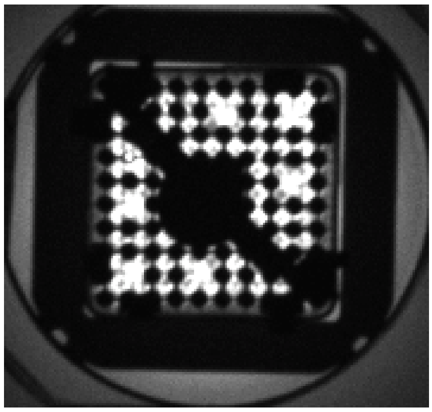


Figure 8d. DCVD image Atrium 10B.

Substituted rods seems to be a more difficult case. From the model images (Figures 6, 8a, and 8b) we can draw the conclusion that a single irradiated substituted rod is very difficult to see and an un-irradiated may be seen. But, in Figure 8c (real image) we can clearly detect the zircaloy rod at A3 visually. At this stage, we can not explain why the model does not show exactly the same attenuation for the substituted rods that is seen in the real images. One reason could be that the spacer grids have a more complicated form than in the model. An-other could be that the surface of the black rods scatter some light. In the model we have assumed 100% absorption, to bring down the computer run time. There was also a difference in fuel cooling time between the model images and the real images.

5 Conclusions

- The model can show what the position of a missing or partial length rod should resemble.
- According to the model, a small group of substituted irradiated rods can be detected, but a group of substituted un-irradiared rods are more difficult.
- The model shows weaker effects of single substituted rods that can be observed in DCVD images.

ACKNOWLEDGEMENTS

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[4] The optical design program Zemax EE is a trademark of Zemax development corporation.

Development of CANDU spent fuel verification system using optical fiber scintillator

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Presented by I.J. Park

Abstract. In CANDU, spent fuels discharge 16~24 bundles from the reactor core at everyday. Those are contained on the tray and the tray is stacked in the spent fuel. Currently, the Agency uses the CANDU Bundle Verifier for Stack (CBVS). It consists of a CZT gamma spectrometric probe which moves vertically along the space in between the columns of trays. Somewhat, spent fuel verification by non-destructive assay has been implemented for safeguards purpose using various radiation detectors such as a gas type detector, a semiconductor detector and so on. However, due to the severe circumstance of spent fuel storage such as high temperature, high radiation intensity and difficult to access area, the applicable radiation detectors and measurement techniques are very limited. An optical fiber scintillator has been known to have a good radiation hardness and physical properties for high temperature and humidity. In order to verify spent fuels stored in difficult to access area, KINAC designed and developed a prototype which was a spent fuel verification equipment using an optical fiber scintillator. The field test was performed at Wolsung NPP (Nuclear Power Plant) pond storage area. And this system will be made an entry for IAEA's verification equipment. For registration, KINAC would be followed the IAEA's QA procedure. At now, KINAC/IAEA developed the user, functional requirement and design specification for System, Hardware and Software separately. After finishing the procedure, it will be used for verification of spent fuel in lieu of CANDU Bundle Verifier for Baskets (CBVS).

1. Introduction

Spent fuels can be verified by measuring radiation attribute from fission products, activated structure components and transuranic elements that build up as a result of fission process. However, due to the intensive radiation and severe environment of spent fuel storage, the applicable radiation detectors and measurement techniques are very restricted [1]. In Wolsung NPP, CANDU spent fuels discharge 16~24 bundles from the reactor core at everyday. Those are contained on a tray and the tray is stacked in the spent fuel pond. The IAEA verifies the spent fuels during the PIV (Physical Inventory Verification) period using a NDA or underwater viewing devices. However these devices cannot verify all the spent fuels in the pond area due to the funnel structure at the bottom layers of a stack. Currently, the IAEA and National inspector verifies the spent fuels using SCAV (Spent fuel CANDU Verifier) by moving vertically along the space between the columns of trays. But, the detector size (abided by gamma shielding requirements) is very large and the detector cannot access to the spent fuel bundles due to the funnel structure. But this movement has given the burden and a potential danger to the facility, as well as a time consuming. So, KINAC developed a verification system based on Optical Fibers that performs gross gamma measurements supporting the re-verification of CANDU spent fuel bundles stored in ponds without moving the horizontal storage trays. A prototype of Optical Fiber System was tested at a CANDU plant. The IAEA requested from ROK Support Program to complete the development of the OFS system into an IAEA's qualified device and associated documentation supporting routine inspection. The Program has been started since Jan. 2006. At now, we have made the related doc. for IAEA's qualification. As follows, KINAC/IAEA has produced the user and functional requirement for Optical Fiber system. At now, we will ready for the design specification and the acceptance test plan. The paper described the distinctive feature of the OFS system and explained the test result of the OFS system on the ground and future plan. Sec. 2 shows the review of Requirement (user and functional requirement) of OFS system. Sec. 3 describes the configuration and distinguishing characteristic of OFS system. Sec. 4 shows the test result of OFS system. The last section is conclusion and future plan.

2. The review of User and Functional Requirement for OFS system

For qualification, we have developed the user and functional requrement. The requirement described the minimum function of OFS system. The function of OFS system likes as follows;

- The system shall be allowed to re-verify bundle inventory stored under water at CANDU reactor without the need for tray movement.
- The system shall be able to verify bundle inventory during routine inspection in lieu of CANDU Bundle Verifier for Stack(CBVS).
- The system shall be able to record radiation traces supporting attribute test of the irrad -iated bundles stored in horizontal trays at CANDU reactor.
- The system is intended to be resident at CANDU reactor.
- The operation of system shall be attended by an IAEA inspector.
- The use of system shall be for the IAEA and/or National Inspection only.
- The system shall confirm the number of bundle at CANDU spent fuel pond.
- The system shall prevent the design risk of contamination from the water of the pond.
- The system shall be able to perform the vertical scanning of the storage tray.
- The system shall be designed to allow the steady motion of the detector along its vertical path.
- The system shall be able to allow the inspector to drive conclusions on the spot.
- The system shall be designed to take into account of the maintenances and repairs.
- The system shall be designed to meet the safety requirements for electrical equipment.
- The system shall be designed to prevent any risk of blockage of the probe during motion along the bundle.
- The system shall be designed to prevent any risk of damage to the fuel bundles.
- The system shall be designed to be operated from the bridge over the pond in compliance with safety rule.
- The system shall be designed enough to be inserted in the vicinity of any bundle thus allowing the verification without moving the storage trays.
- The system shall be designed enough to length of detector for verifying any bundle layers.
- The system shall be designed easily to move the detector in X-Y plan within the assigned range.

- The system shall be designed to compact size and minimize weight.
- All data files shall be stored in formats compatible with industrial standard.
- The system shall be designed to have inspector's information to data acquisition and processing system.
- The system shall be designed to check the communication status.
- The system shall be designed to display the net signal which is allowing the inspector to confirm the presence of the bundles on the spot.
- The system shall be designed to have a button which is supporting to various functions.
- The Graphic User Interface (GUI) shall be designed to user-friendly, easy-to-handle.
- The system shall be designed to process the data in order to automatically count the number of detected bundle layers.
- The system shall be designed to a warning message if the number of detected bundles is different from the declared number at the measurement point.
- The system shall be designed to a warning message in case of detected malfunctioning of the system.
- The system may be designed to self-testing sequence.
- The system shall be designed to consider with commercially available component.
- The system shall be designed to consider with harsh environment(for example, temperature or humidity, etc)

3. The Configuration for OFS system

The OFS system consists of an optical fiber coupled to a light guide. A compensating light guide, two Photo Multiplier Tube(PMT) module, electronic equipment, a scanning system and data acquisition software as shown in Figure 1.

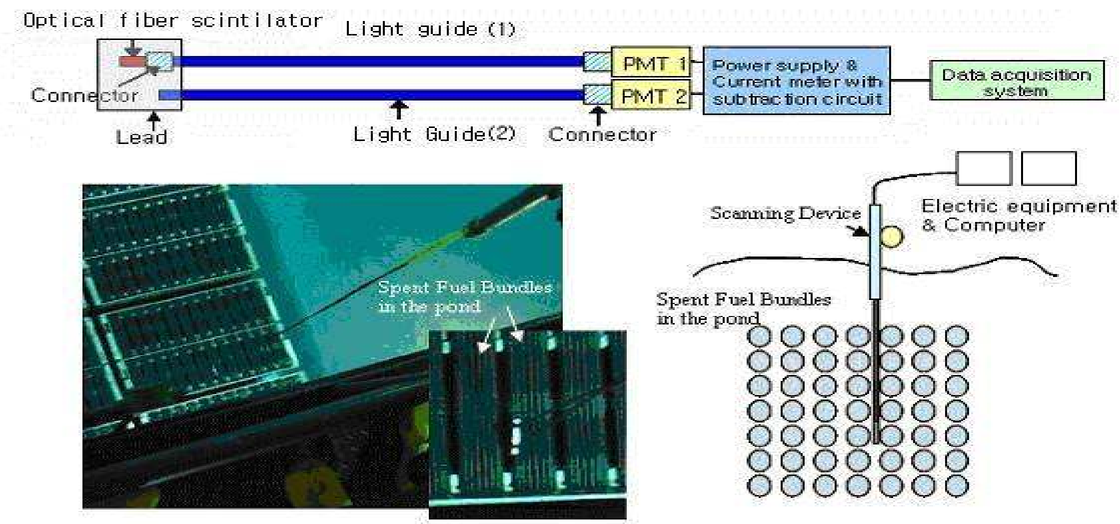


FIG. 1. A spent fuel verification system using optical fiber scintilator.

The detector head inserted into between bundles in tray, which is around 1.5cm gap. The OFS system measures the gross gamma intensity as a function of vertical position by scanning the storage stack without moving the tray. We initally used the CANDU Bundle Verifier for Stack (CBVS) scanning device to insert OFS into target point. However, the device is too heavy and its is different to handle due to the size of sensor[2]. And, according to requirement, it is needed to develop the actuation system which is easily and safely handled the OFS. The main concepts of functional and user requirement are three principles; compact, user friendly and easily handle to decommission for system. Table 1 summed up between principles and described function at OFS system. Figure 2. shows the picture of OFS's actuation system. The characteristcs of the developing system are ① easily manual x-y position handling device for adjusting the target point ② PC sliding Guide for connecting Notebook with current meter ③ no vibration in putting the detector into target position, what is called "damping function" used to stopper □ light weight (less than 40 Kg(totally)) using light meterial.

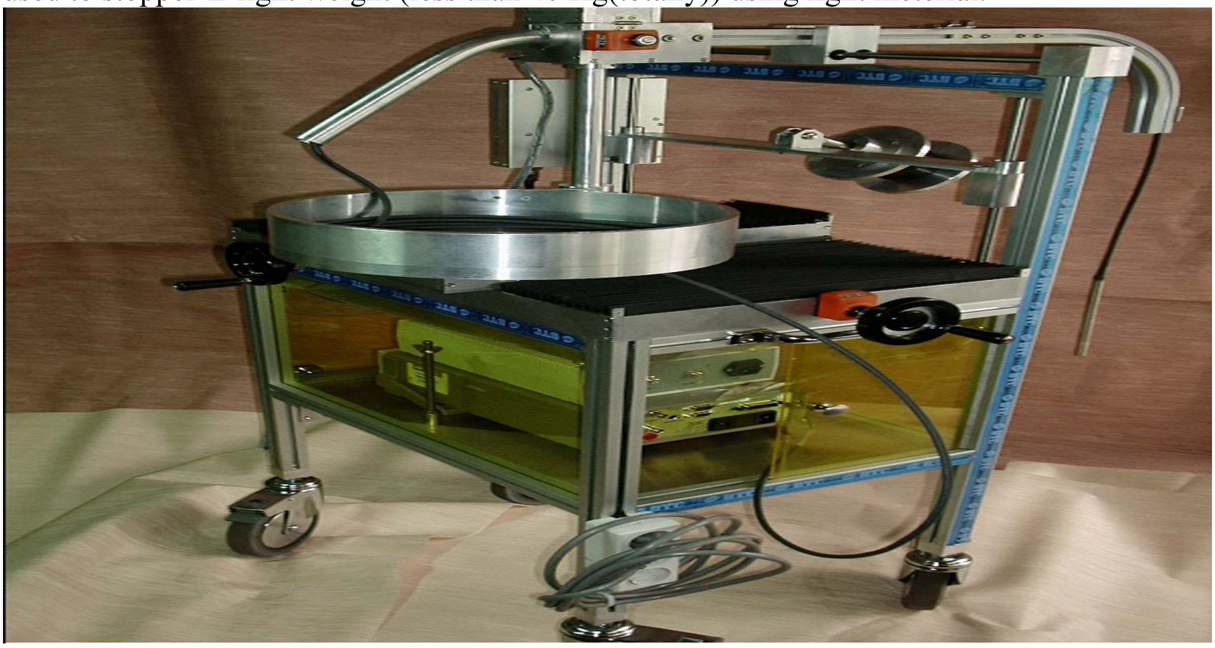


FIG. 2. The prototype of actuation system for OFS.

From Figure 2, conforming to requirements, the prototype system was developed.

Table 1. Comparsion the principles with the described function at OFS system.

Principles	Described function at OFS system
Compact	- Less than 40Kg(totally) - furnish a knob for handling easily, put a wheel for moving easily - stopper for no vibration
User friendly to handle	- a manual x-y position handling device for adjusting the target point - fixing structure at bridge for no vibration of system - easily graphic user interface(3 pages)
easily handle to decommission for system	- filtering system was used at the front of the OFS system.

Figure 3 shows the flowchart of OFS system and Graphic User Interface(GUI). To meet the above-mentioned requirements, the software system was designed to simple, compact and user friendly.

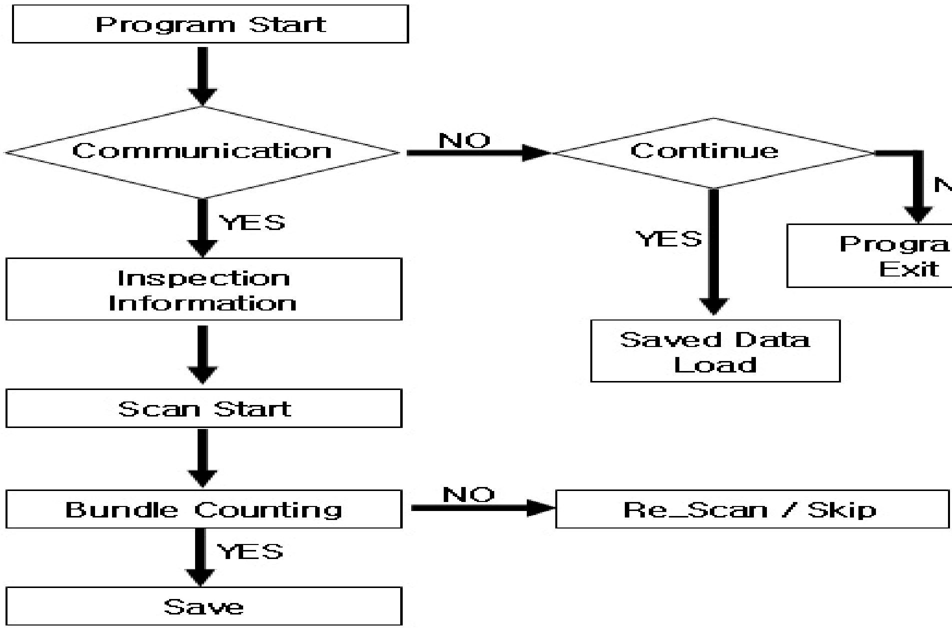


FIG. 3. Flowchart of OFS system.

In starting the program, it showed to check the communcation status. The communication check displayed between notebook and photodiodemeter. Figure 4. shows the configuration between notebook and photodiodemeter.

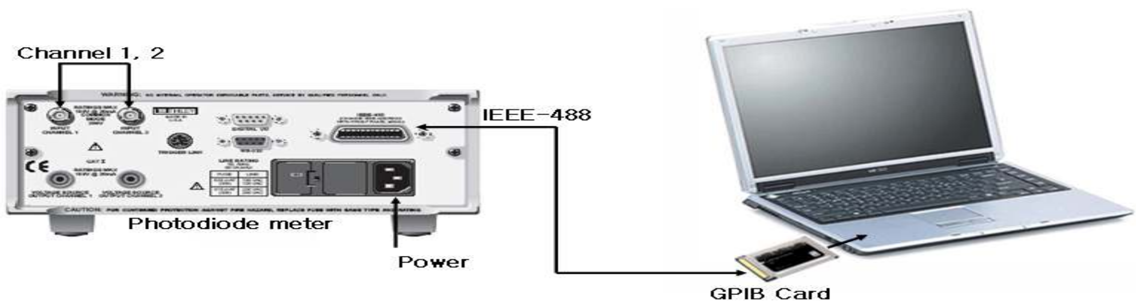
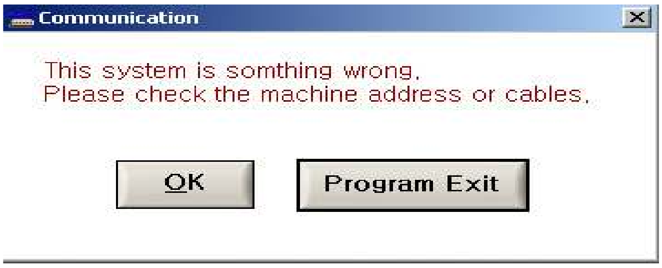


FIG. 4. The Configuration between notebook and photodiode-meter.

The initial communication status displayed the status of system with different color which the green is normal status and the red is abonormal status. And in case of abnormal, a warning message is displayed at the window. Figure 5 shows the communciation status. After checking the communciation status, the window of inspection information was displayed. The item of the window was configured with 13 items which included inspection number, facility name and code, MBA code, item ID and # of declared bundle layer etc. Figure 6 showed the window of inspection information. After that, the data acquisition and saving windows was displayed. The configuartion of window likes that; the upper side shows data acquisition part from sensor which displayed the delta value between signal I(include Scintillator) and II(not include Scintillator) and the lower side displayed the text value of upper side and the right side of lower shows the comparison between the mesured number of bundle layer and declared number of bundle layer. If the number is the same, then the ramp dispalyed green and success. But if fail, then the ramp displayed was red and fail. The inspector would be retried using rescan button. Figure 7 shows the window of data acquisition and saving.

(a)

(b)



(c)

FIG. 5. (a) normal case, (b) abnormal case,(c) information window at abnormal case.

FIG. 6. The window of Inspection Information.

For comparing the number of decleared bundle layer with the number of measured bundle layer, it was developed to bundle searching algorithm. The frame of bundle searching algorithm was that the bundle was determined when 2 samples seires data was increase and decrease in Figure 7.

4. The test result of OFS system

Using the developed Software, the OFS system was tested on the ground. The traget plant was Wolsung 3 unit. The traget was tested to five points.

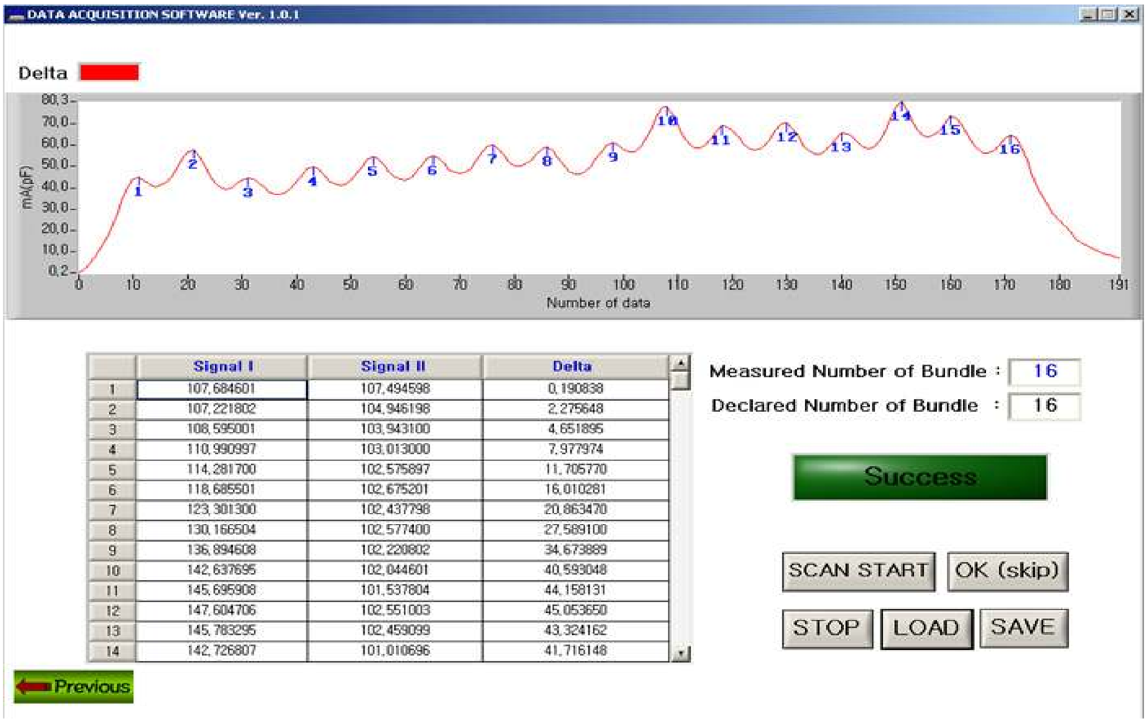


FIG. 7. The window of data acquisition and saving window.

Fig. 7 shows the upward direction scanning result at one point. Fig. 8 shows the both direction(up & down) scanning result at other point. Table 2 shows the test result from Data Acquisition Software using bundle searching algorithm.

Table 2. The test result from Data Acquisition Software using bundle searching algorithm.

Test points	Sensor Direction	# of decleared bundle layer	# of measured bundle layer	Success or Fail
1	Up	16	16	Success
2	Down & up	32	32	Success
3	Down & up	32	32	Success
4	Down & up	32	32	Success
5	Down	16	16	Success
5-1	Down & up	32	32	Success

5. Conclusion and Future plan

The IAEA verifies the spent fuels during the PIV(Physical Inventory Verification) period using a NDA or underwater viewing devices. However these devices cannot verify all the spent fuels in the pond area due to the funnel structure at the bottom layers of a stack. Currently, the IAEA and National inspector verifies the spent fuels using CANDU Bundle Verifier for Baskets (CBVB))by moving vertically along the space in between the columns of trays.

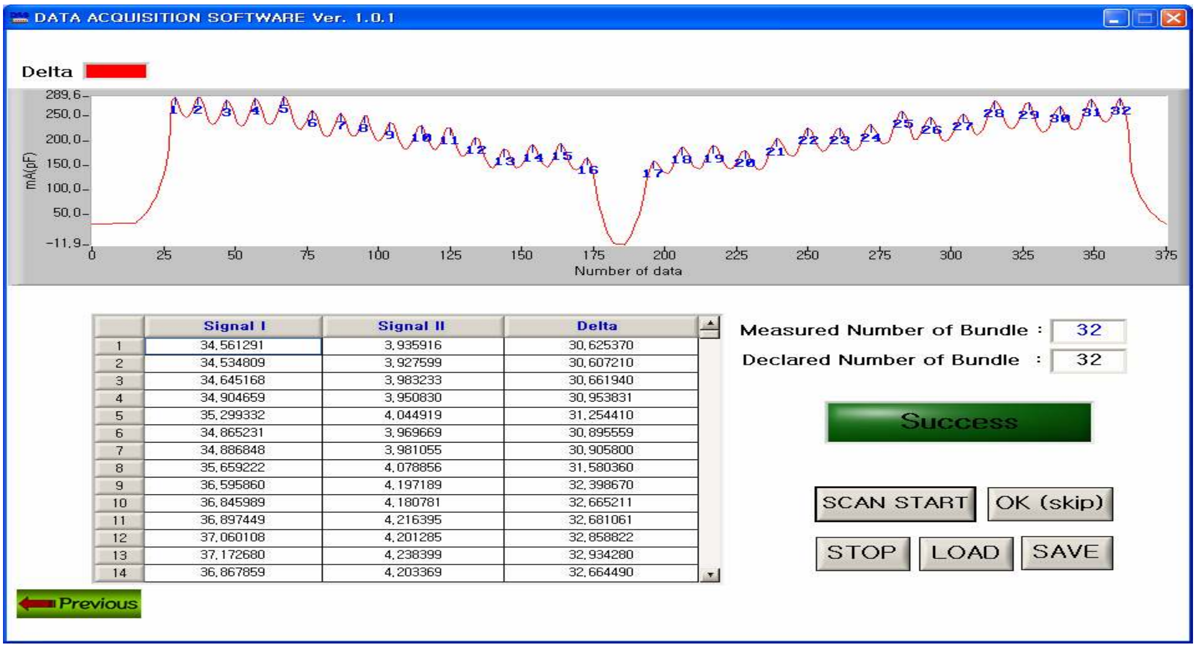


FIG. 8. The both side(up & down) of test data from one point.

But, the detector size (abided by gamma shielding requirements) is very large and the detector cannot access to the spent fuel bundles due to the funnel structure. If the C/S fails, the inspector requires to move the trays for re-verification one by one. But this movement has given the burden and a potential danger to the facility, as well as a time consuming.

KINAC developed a verification system based on Optical Fibers that performs gross gamma measurements supporting the re-verification of CANDU spent fuel bundles stored in ponds without moving the storage trays. The measurement result on the spot easily found out the clear peak as the spent fuel existed in each tray. The OFS system can be used to item counting tool for safeguard purpose.

If the OFS system will be applied to IAEA inspection, then it will be minimized the inspection effort and impact on CANDU power plant operation. And if the OFS system will be admitted to the qualified verification tool from IAEA, the OFS system should be able to use to CANDU operation countrys for verification equipment.

ACKNOWLEDGEMENTS

This work has been carried out under the Nuclear Research and Development program supported by MOST.

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Experience of implementing the additional protocol in the EU: The first years

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Abstract. The Additional Protocol (AP) is in force in all 25 Member States of the European Union (EU). The three APs related to the Safeguards Agreements INFCIRC/193, INFCIRC/263 and INFCIRC/290 entered into force on 30 April 2004 covering the fifteen states members of the EU at that time. These APs are all trilateral agreements between the IAEA, the Member States and the European Atomic Energy Community (Euratom), reflecting the role of Euratom has as regional safeguards authority for accountancy and control (RSAC) in the European Union. They are also the only trilateral APs currently in force in the world. At the time of the EU enlargement in 2004 the AP was in force in seven of the ten new Member States. These States have been reporting to the IAEA on the basis of their bilateral APs for several years. In the remaining three new Member States the AP was brought into force after their accession to the EU. The European Commission is a supra-national body, which in addition to its role as a RSAC, is also a nuclear operator and a component of Europe‘s nuclear research capacity. It therefore declares its own activities relevant to the AP as a separate entity and party to the AP. Thanks to its role as a focal point in implementing the AP in the European Union, the Commission has accumulated an ample experience in reporting, following up and in matters relating to Complementary Access. The paper reviews this experience and presents an outlook of the future challenges of the AP implementation in the EU.

1. Background

1.1. IAEA Safeguards in the EU

The traditional system of nuclear safeguards in the EU is established in the Euratom Treaty, which in its Articles 2(e) and 77(a) makes the European Commission responsible for ensuring by appropriate supervision that nuclear materials are not diverted from their intended uses. The Euratom Regulation[1], implementing this provision according to Article 79 sets out the requirements for the nuclear operators of the Member States to report to the Commission on nuclear materials. In order to fulfil its responsibilities, the European Commission has the powers to carry out inspections at nuclear installations.

Article 77(b) of the Treaty requests that the European Commission makes sure that obligations undertaken under Agreements with third States or international organisations are fulfilled. Here the Safeguards Agreement with the IAEA (INFCIRC/193) is of paramount importance. This Safeguards Agreement[2] is trilateral, with the States, the Community and the IAEA as parties. In fact, three Safeguards Agreements cover the Member States of the EU: one is common for all non nuclear weapon states (NNWS), while the two nuclear weapon states (NWS), France and the United Kingdom, are party to one Agreement each. Recognising the Commission’s role as the safeguards authority of the European Community (EURATOM), the Safeguards Agreements foresee that the IAEA has to make full use of the Community’s safeguards system.

These special features had to be taken into account when the AP was negotiated for the Member States of the European Union. The Model Protocol needed adaptation with regard to the attribution of tasks, some of which fell to the Commission’s field of responsibilities while others were considered as being outside the existing European legal framework. Careful reflection was necessary with regard to the attribution of the tasks, in order to avoid duplication of work.

1.2. Additional Protocol

1.2.1. Non nuclear weapon states

Two main orientations emerged in the deliberations in the EU Council. One Group of Member States was in favour of attributing the broadest range of responsibilities possible to the Commission. Other Member States, which had an established national system when they acceded to the EU, wanted to limit the Commission's role strictly to the area covered by the Euratom Treaty, i.e. the control of nuclear materials.

The outcome of the negotiations was a compromise. The tasks involving nuclear material and activities were attributed to the Commission, while the State was made responsible to declare directly non-nuclear material related tasks. Responsibility is shared in matters concerning nuclear sites and waste. Annex III to the AP, however, permits the States wishing to do so to entrust to the Commission the implementation of the AP on its behalf. Ten of the thirteen non-nuclear weapon states chose to use this so-called side-letter arrangement. The other three opted for the direct declaration to the IAEA of the parts of the AP reporting that is considered as State responsibility.

The Joint Research Centre (JRC) is part of the European Commission and operates in four different Member States. Taking into account this specificity, in particular the independence from the host State authorities in planning and performing research, the full responsibility for the implementation of the AP on behalf of the JRC is assumed by the Commission.

1.2.2. Nuclear weapon states

As an expression of their commitment to strengthening the international safeguards system, the nuclear weapon states of the EU agreed to place part of their nuclear sector under the supplementary controls created by the Model Protocol. Additional Protocols were thus negotiated for the two nuclear weapon states of the EU, France and the United Kingdom. The most notable difference compared to the non-nuclear weapon states' AP is that reporting is mainly required concerning activities conducted between them and non-nuclear weapon states. The responsibilities for implementation were divided between the Member State and the Community in a similar way as in the non-side letter States' AP; hence the Commission's competence is limited to nuclear materials only.

The three Additional Protocols were signed on 22 September 1998.

1.2.3. New Member States

On their accession to the EU the new Member States had all bilateral Safeguards Agreements with the IAEA and had signed APs to these Agreements. Pursuant to the Act of Accession, the new Member States have to fold in their Safeguards Agreements into the Community one. So far two – Estonia and Slovakia – have accomplished this by the entry into force of the trilateral Euratom/IAEA/EU Member States Agreement and its AP on 1 December 2005. In the other eight new Member States the AP to the bilateral Safeguards Agreement is in force, but they have yet to fold in to the common trilateral Agreement. Slovenia is the first of the new Member States having bilateral Agreements and APs in force that completed the national procedures for the change. The European Commission's statement to the IAEA that will confirm its preparedness to fulfil the Safeguards Agreement's obligations also for this country is expected on 1 September.

2. Getting Ready for Implementation

2.1. Legal Framework

Before the APs could enter into force, adequate legal basis for their implementation was needed. New legal measures were required both at the European and national levels. The Euratom Regulation was revised, mainly in order to accommodate the new reporting requirements concerning nuclear sites and certain types of waste. The revised Euratom Regulation provides the legal basis for the collection of data as regards AP reporting requirements related to nuclear material and falling under the Commission‘s responsibility or the joint responsibility of the Commission and the Member State.

For their part, the Member States needed to make their national legislation conform to the new commitments regarding the collection of information about some activities related to the nuclear fuel cycle. The powers that the Additional Protocol gives to the IAEA for verification also required adaptations.

Detailed arrangements for the implementation of the Protocol in each Member State were prepared on the basis of a generic model paper, which had been elaborated by the Commission in consultation with all Member States. The text covers information flows for reporting, verification and follow-up, as well as the contact points for different purposes. On the basis of the model paper a specific implementation paper was prepared for each Member State, describing all the arrangements between each responsible National Authority and the Commission’s service entrusted with the implementation of the AP. It clarifies the role of all parties in a coherent way within the EU.

The process before the AP could enter into force turned out to be lengthy: it took almost six years from the signature to the entry into force on 30 April 2004.

2.2. Preparations for Reporting

During these years the Commission prepared the implementation of the AP in working groups and by conducting meetings with nuclear operators, Member States and the IAEA. In technical discussions with the IAEA, the Commission developed an approach to take into account the specificity of the European nuclear industry. Specific issues arising from the AP, such as the status of facilities under decommissioning or the detailed conditions for the exemption from the IAEA safeguards, were addressed. In order to test the practical arrangements for reporting and verification, two field trials were performed before the implementation started.

2.2.1. Site definition and site declarations

Preparations were based on the site definition as set out in Article 18 of the AP. The aim was to get the actors in the Member States ready so that reporting requirements could be fulfilled in a correct and timely manner. The number of sites was to be kept reasonable, in order to avoid excess burden to all parties. Buildings adjacent to nuclear facilities but unrelated to the nuclear activity did not need to be included in the site. Where appropriate, information about such buildings was provided by listing their identities and use without, however, declaring them as a part of the site.

The Additional Protocol does not require declarations concerning facilities and LOFs considered decommissioned. However, in order to avoid ambiguity and unnecessary questions, clarification was sought from operators and common verification visits started well before the entry into force of the AP. Due to the large number of such installations and the limited resources available to the two Safeguards Authorities, the status of all locations could not be confirmed before the initial declaration was submitted. Information on the locations yet to be verified was provided outside the formal AP declaration.

2.2.2. Exemptions

With regard to holders of nuclear material seeking exemption from the IAEA Safeguards, the situation is similar. No site declaration has been made for locations that are candidates for exemption. Instead a note listing these locations and referring to the exemption request has been attached to the State declaration.

The common European MBA regrouping a large number of very small holders of nuclear material, the CAM, turned out to be a special case among the MBAs to be exempted. Although it is otherwise similar to the national LOFs or MBAs that several States have exempted under their Safeguards Agreement, it has the unique feature that it comprises holders from all Member States of EU-13. This seems to have led to some reluctance on the Agency side to give positive reply to the exemption request. However, working together we have devised a procedure which should satisfy the needs of the IAEA without creating a big burden to many small users of nuclear material whose activities are not linked to the nuclear fuel cycle.

Another issue yet to be resolved is the interpretation that the IAEA gives to the quantitative criterion (Article 37 of the Safeguards Agreement), which foresees an exemption for a total quantity of 1 SQ as upper limit. The narrow interpretation of the IAEA that this upper limit would apply for the whole Agreement seems unrealistic with the increasing number of States in the EU. Besides, in the context of the Additional Protocol the location of all nuclear material exempted from Safeguards under Article 37 is reported annually pursuant to Article 2a(vii) and the Agency has access to these locations on 24-hour notice.

3. Reporting experience

The three AP s entered into force on 30 April 2004. The first reporting deadlines thus fell on 29 August 2004 (export declarations) and on 28 October 2004 (initial declarations). Despite the extensive preparations undertaken before the entry into force of the AP, intense work continued until the last moments before the deadline. Although with some difficulty, the initial reporting was made on time. The first annual update was submitted on 15 May 2005 and the second on 15 May 2006. The initial declarations for the first two new Member States having acceded to the common Safeguards Agreement – Estonia and Slovakia – were submitted to the IAEA on 29 May 2006.

3.1. Some Statistics

In the initial declaration for the NNWS (EU-13), 118 sites were reported. The total number of line entries in the initial declarations was 3969, of which 3701 were in the site declarations. The declarations concerning research and development activities reported under Articles 2a(i) and 2b(i) contained about 100 line entries. Mines and concentration plants (almost all 'permanently closed down') were reported in 35 lines.

The export declarations under Article 2a(ix) usually contain 250-350 lines per quarter. By far the greatest number of them concerns one multinational company which operates in the borderless Europe, transferring equipment between its locations in the different States.

In July and September 2005 we received in all 359 of requests for amplification and clarification under Article 2c. They were handled in line with the implementation arrangements with each Member State, seeking to provide response within reasonable time while ensuring sufficient harmonisation in reporting across Member States and coherence with respect to the requirements set out in the AP. Some of the requests were quite generic, may have contributed to response that was less specific than the Agency perhaps hoped to receive. We chose to proceed in two steps. First, a reply was provided under Article 2c. As a second step replies were included in the annual update of the AP declarations where appropriate.

While the second annual update of most site declarations consisted in minor modifications made following the IAEA‘s requests under Article 2c, some were reworked more thoroughly. One site disappeared altogether because the facility forming its core had been decommissioned in 2005. At the same time three new sites were declared for installations which had initially been considered decommissioned but were now introduced as sites on the IAEA’s request.

3.2. Reporting Tools and Data Transmission

The Commission has developed a reporting tool **CAPE** (**C**ommission **A**dditional **P**rotocol **E**ditor) for all reporting entities to use when preparing their declarations. The declarations (submission files) are extracted from the local CAPE database in an XML format and they can be visualised using a standard internet browser. Each submission file, consisting of one or more declarations, contains in one package the declaration(s) and all attached files, such as the site map, linked to the declaration.

The submission of data to Euratom is mostly done by e-mail, using the Commission's closed web forum for document exchange (CIRCA) or by sending an electronic data carrier, CD or diskette by post. About 80% of all transmissions of AP declarations were received in an electronic format, most of them through electronic transmission channels. The number of paper submission by letter or fax is relatively small although its share has risen due to the fact that declarations with 'no change' are relatively often received by fax or letter.

After introducing all the data and attachments to be included in the declarations into the central CAPE database, the declarations are regrouped into submission packages – one per State – and sent to Vienna. This is done by e-mail or using the CIRCA forum, depending on the size of the files to be sent. The declarations are currently converted into the Protocol Reporter format before submission, but this additional step is soon to disappear, as it will become possible for the Agency to load the CAPE submission file into its database.

Although CAPE became available to the users only shortly before the reporting deadline for the initial declaration, it was used in the majority of declarations. About 85% of all initial site declarations were received in CAPE format. Three Member States resort to the Agency's Protocol Reporter, while a few entities use other tools such as Excel or Word.

From the point of view of a unit handling a large number of declarations originating from a large number or declaring entities, it is important that the format of reporting is standardized. This helps avoid ambiguity as to the content of the declaration and thereby speeds up processing. Although the situation is already relatively good in this respect, some room for improvement remains.

4. Complementary Access

Different from the classic Nuclear Material Accountancy based Safeguards, where the Commission's role is an active one, in the framework of the AP, our role is to observe, explain and be a link in the communication chain.

For the side-letter states the Commission is the focal point of communication concerning the CA. It receives notice of all CAs and transmits the information to the operator and the National Safeguards Authority. Euratom inspectors accompany the IAEA during the CAs. Communications after the visit, including possible questions or requests for further clarification, arrive to the Commission, which prepares answers in cooperation with the operator and/or State. With regard to the non-side letter states, the Commission receives advance notices and is involved in the areas under its own or shared responsibility, i.e. locations with nuclear material.

We contribute by our presence to the equal and just implementation of the Additional Protocol throughout the countries of the European Union. We are in a position to explain to our Member State actors their obligations and their rights in this context and we can facilitate the work of the IAEA in the EU.

4.1. Some Statistics

The IAEA made the first Complementary Access on 21 December 2004, and the verification activities gained momentum beginning from the second half of March 2005. By 24 August 2006, 53 Complementary Accesses had taken place in EU-13: 33 at 24-hour notice and 20 at 2-hour notice (in the framework of a safeguards inspection). As regards the split between the side-letter states and non-side-letter states, the former account for 42 of the 53 CAs. Euratom inspectors were present during all Accesses. A representative of the national safeguards authority attended almost all Accesses in the non-side-letter states and some, although fewer, in the side-letter states.

All Complementary Accesses have so far been made to locations using nuclear material. A rough classification yields a quick image of the types of places accessed so far (Figure 1). Not surprisingly, installations with research related activities have received the largest number of visits. With regard to ‘manufacturing’, one could add to the 2 Complementary Accesses the 9 that took place on sites with enrichment plants where most of the accessed buildings were related to equipment manufacturing.

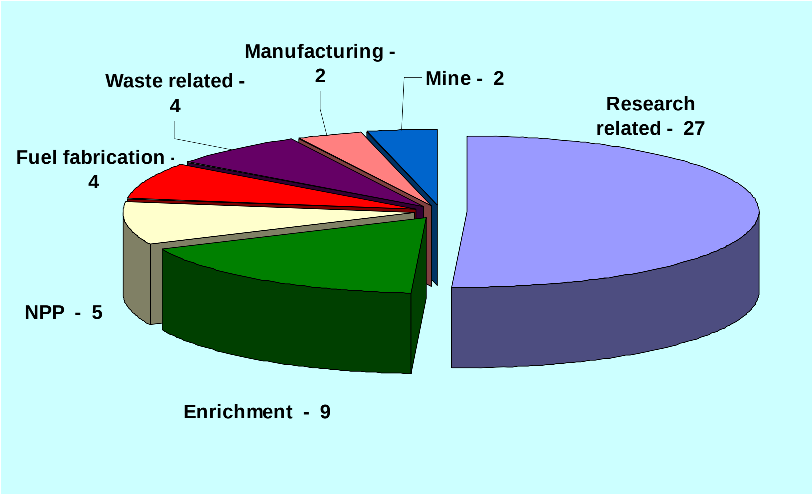

FIG. 1. Complementary Access in EU-13 by type of installation.

Figure 2 shows the percentage share of CAs made in the NNWS of the EU by 10 July 2006 by Member State compared to the percentage share of the sites in each Member State.

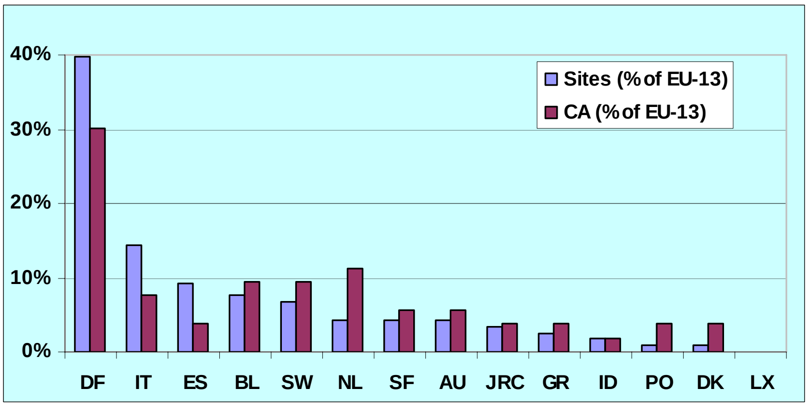

FIG. 2. Sites and Complementary Access by State in percentage terms.

4.2. Activities

Apart from visual observation and measurement of radiation levels, environmental sampling has been widely used: in 44 installations with over 100 samples collected in all. Typically one or two samples were collected per CA. We expect that the Agency will inform us about the results not only in case of apparent inconsistency between the results and the declared activities but also where no inconsistency is found. In this way the operators can rid themselves of the reference samples that are no longer needed.

Photography is commonly used as a beneficial means of documenting activities carried out during Complementary Access. Photography has been used during 43 CAs. The special procedures applied in enrichment plants taken into account, photography was used in all except one CA. Most Member States are reluctant to allow extended use of photography during Complementary Access for security reasons. There is an agreed compromise procedure that puts a certain control in the hands of the operators who have the right to check the pictures taken before they are released to Vienna. The operator can decide which pictures are released and whether copies are given in an electronic format or on paper. The pictures that the operator considers too sensitive to be released are kept on site under seal. In this case, however, the benefit drawn from photography as a means of note-taking is reduced. In practice operators have asked that pictures remain on site or be deleted only in very few cases.

4.3. Past Activities

Another issue that has led to discussions with Member States and the IAEA have been the Agency's requests for clarification relating to very early days of nuclear activities. Doubts have been expressed concerning the usefulness referring to the fifties or the sixties of the previous century. In addition it appears very difficult to find today adequate sources of information. Difficulties could arise in particular when such questions are put without prior notice during Complementary Access and the answers obtained from the persons available at that moment are linked as the operator's statement to environmental samples.

4.4. Planning and execution

The planning and the execution of Complementary access visits create sometimes problems to Euratom, notably the ones announced 24 hours in advance. If in some active areas of the fuel cycle it seems reasonable to avoid by the short advance notice concealment of undeclared activities, it doesn't seem meaningful to use the same practice in particular idle locations like a closed or even decommissioned mine.

5. Conclusion

The Commission was successful in putting into force and implementing the Additional Protocol in the European Union (EU15). Both the content and the timeliness of the reports met IAEA requirements. We trust that our work has facilitated the Agency's task in implementing the Additional Protocol in the European Union. Despite some critical views that have been expressed we feel that the cooperation has been good.

The outstanding issues and challenges for the future can be summarised as follows:

(1) Facilitate the resolution of issues related to clarification and amplification requests of the IAEA in the framework of the AP.
(2) Finalise the process of adherence of the new Member States to the Euratom/ IAEA/ Member States trilateral Safeguards Agreement and its AP.
(3) Work towards a unique implementation mode of the AP in all EU Non-Nuclear-Weapon-States and contribute to an equal implementation in all of these States.

After several years of successful implementation of the AP in a Member State, the IAEA may arrive to a positive Safeguards Conclusion for this State, i.e. that the received declarations are correct and complete. Then the IAEA will proceed in introducing "Integrated Safeguards", a combined system optimising the use of the measures available both in the Safeguards Agreement and in its Additional Protocol.

Hence, another important task for the Commission is to define together with the IAEA the modalities of the future co-operation under an "Integrated Safeguards" regime.

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[2] Agreement between the Kingdom of Belgium, the Kingdom of Denmark, the Federal Republic of Germany, Ireland, the Italian Republic, the Grand Duchy of Luxembourg, the Kingdom of the Netherlands, the European Atomic Energy Community and the International Atomic Energy Agency in implementation of Article III (1) and (4) of the Treaty on the non-proliferation of nuclear weapons (78/164/Euratom). OJ L 51, 22.2.1978, p. 1.

Implementation of INFCIRC/193 and its Protocol Additional in the Slovak Republic

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Abstract. The paper shows the development of safeguards implementation in Slovakia during past years. The former Czechoslovakia was one of the first countries, which has implemented the safeguards system. After dissolution of Czechoslovakia and after accession to the European Union (EU) our safeguards system went through several changes in order to meet all new requirements, all new challenges. Especially the implementation of INFCIRC/193 and its Additional Protocol (AP) has been very challenging.

1. Introduction

The former Czechoslovakia acceded to the Non-Proliferation Treaty on 1 July 1968. Safeguards has been accepted as set forth in the Safeguards Agreement between the Government of the Czechoslovak Socialist Republic and the IAEA published in the IAEA INFCIRC/153 which was signed on 1 March 1972 and entered into force on 3 March 1973. Based on requirements of the Safeguard Agreement the State System of Accounting for and Control of nuclear material has been established and the Czechoslovak Atomic Energy Commission was authorized to play the role of a regulator in this area.

After dissolution of Czechoslovakia the Slovak Republic succeeded to the Safeguards Agreement. As a regulator the Nuclear Regulatory Authority of the Slovak Republic (UJD) has been constituted.

A full legal frame of safeguards was completed in the Act No. 130 Coll. on Peaceful Use of Nuclear Energy issued in 1998. Subsequently in 1999 the UJD defined in Decree No. 198 details and requirements on accounting system.

After EU accession EU legislation became valid in the Slovak Republic. In order to meet requirements described in Euratom Treaty and Regulation No. 3227/76, which has been replaced in 2005 by European Commission (EC) Regulation No. 302/2005 on the application of EURATOM safeguards, it became necessary to adapt national legislation. On 1 December 2004 a new Atomic Law No. 541/2004 Coll. on Peaceful Use of Nuclear Energy entered into force. Following new Atomic Law and EU safeguards legislation Decree No. 54/2006 Coll. on the record-keeping system and control of nuclear material as well as notification of selected activities was issued. Although the responsibility for meeting liabilities towards the INFCIRC/193 rests with the EC, Slovakia has maintained its own nuclear material record keeping system.

In the frame of strengthening the IAEA safeguards an implementation of the AP became actual. As a first step an impact on legislation has been assessed. The AP was signed by the government of the Slovak Republic in September 1999. However, the AP could not be ratified, as the laws being in effect in the Slovak Republic in 1999 did not make it possible to meet all requirements resulting from the AP. To enable the ratification of the AP, first of all it was necessary to amend the Atomic Law and associated regulations.

On 30 April 2004 the AP to the INFCIRC/193 entered into force for old EU member states.

The National Parliament of the Slovak Republic agreed with INFCIRC/193 and its AP on 27 May 2004 and on 21 June 2004 the president of the Slovak Republic has signed the letter of accession of Slovakia to the safeguards agreement INFCIRC/193 including the relevant AP.

On 1 December 2005 the safeguards agreement INFCIRC/193 including the relevant AP entered into force. Starting from 1 December 2005 the UJD has begun with the implementation of INFCIRC/193 and its Protocol Additional.

2. INFCIRC/193 and its Additional Protocol implementation

2.1. INFCIRC/193

On 1 May 2004, the date of entry of the Slovak Republic into EU, both Slovakia and EC were not fully prepared for the new reporting system according to the EC regulations. On the one hand we had to amend the Atomic Law; on the other hand the EC had to prepare itself for the distributed data processing. Therefore we agreed to continue sending reports according to the former valid Safeguards Agreement INFCIRC/173 to the IAEA, and at the same time to prepare and send reports to the EC according to the new Safeguards Agreement INFCIRC/193 and EC Regulation No. 3227/76, later replaced by EC Regulation No. 302/2005. Such situation lasts for 19 months until 1 December 2005, when the trilateral Safeguards Agreement INFCIRC/193 and its Additional Protocol entered into force. Slovakia and Estonia have become first countries from new EU member states which have applied INFCIRC/193 and its AP.

During this transient period Slovakia and EC had to solve a lot of problems. We had to create new reporting system regarding EC legislation and new safeguards agreement and until that time we had to report according to the old safeguards agreement to the IAEA. In addition, the responsibility for sending accountancy reports to the EC rests with operators, not with state. UJD had to support the operators in reports preparation. The EC has prepared new software, which is being used not only for accountancy report preparation, but also for sending encrypted reports to the EC Safeguard Office in Luxembourg. This software is called ACCESS. The EC has provided the UJD and operators with working stations; however the installation of ACCESS software, testing and activation of whole system was complicated. The working stations with ACCESS software were delivered in February 2005 and it has taken about one year till we could prepare and send first exact report. In order to overcome problems and speed up the implementation of INFCIRC/193 the UJD and EC have organized several meetings either on technical or legislative level.

After solving all the problems, since the beginning of 2006 the reporting system is working well. The Nuclear Power Plant operators prepare and send the accountancy reports both to the EC and UJD. According to the agreement between UJD and EC the UJD reports for small holders. As it was mentioned before, the UJD maintains national SSAC.

2.2. Additional Protocol

The situation with implementation of AP was more difficult as implementation of INFCIRC/193. We have had experience with safeguards, so we were only modifying our existing system, but we did not have experience with AP.

As mentioned before Slovakia has ratified the AP in May 2004 and we have declared our preparedness to implement the AP by the end of 2004. The following procedure took almost one year, thus the AP entered into force on 1 December 2005. The UJD has informed the EC, that we wish implement the AP as a non-side letter state, and that the site representative for whole Slovak Republic will be UJD.

For the preparation of the submissions we are using Protocol Reporter, the submissions are encrypted using PGP code. In the following table the reports sent either to the EC or the IAEA is shown:

Number	Article	Reference (site code and rep. for 2a iii)	Submitted by	Submission of
1	2a(ix)	Q4/05 exports	UJD	Feb-06
2	2a(ix)	Q1/06 exports	UJD	May-06
3	2a(i)	State R&D	UJD	May-06
4	2a(iii)	SSXBOHU (site rep.)	EC	May-06
5	2a(iii)	SSXMOCH (site rep.)	EC	May-06
6	2a(iv)	Equipment	UJD	May-06
7	2a(v)	Mines	EC	May-06
8	2a(vi)	Source material	EC	May-06
9	2a(vii)	Exemptions	EC	May-06
10	2a(x)	10-yr plan	UJD	May-06
11	2b(i)	Private R&D	UJD	May-06
12	NOTE	Pending exemption requests	EC	May-06
13	2a(ix)	Q2/06 exports	UJD	Aug-06

Table 1. The AP submissions

3. Supporting activities

As an instrument supporting non-proliferation of nuclear weapons a control of export/import of nuclear material, nuclear related and dual-use material, equipment and technologies is being used. The government of the Slovak Republic like many other governments' uses for this purpose licensing system.

Basic principles for control of export/import of selected goods were defined in the Act No. 26/2002. The act clearly describes requirement of exporters/importers and also responsibility and powers of the regulator which is the Ministry of Economy of SR issuing license. In like manner as for safeguards, after accession to the EU, Regulation 1334/2000 setting up a Community regime for the control of exports of dual-use items and technology entered into force.

The role of the UJD in the export/import control system is anchored in the Act No. 541/2004 Coll. The act gives the UJD power to issue permission for export/import of nuclear material, nuclear-related or dual-use material, equipment and technologies a list of which is included in Regulation 1334/2000. In issuing the permission the UJD strictly follows recommendations and requirements of the Regulation 1334/2000 and NSG guidelines published in the IAEA INFCIRC/254 and guidelines of the Zangger Committee published in the IAEA INFCIRC/209.

The government of the Slovak Republic plays active role within activities of the NSG and the Zangger Committee aimed in strengthening non-proliferation regimes. Delegation of Slovakia is actively participating on activities of the NSG consultative group, which performs support to NSG plenary meetings.

4. Conclusion

The transition from INFCIRC/173 to INFCIRC/193 and its AP we went through has required much work. Some procedures were used for the first time, and we have had a good opportunity to go through all steps of INFCIRC/193 and its AP implementation. We could also strengthen the co-operation with the EC Safeguards Office. We are going to start the preparatory work for the integrated safeguards implementation.

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[2] Regulation No. 54/2006 Coll. on the record-keeping system and control of nuclear material as well as notification of selected activities
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[4] Regulation No. 302/2005 on the application of Euratom safeguards

Additional protocol experience in Romania*

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Abstract. National Commission for Nuclear Activities Control (CNCAN) is the national regulatory body with regulation, authorization and control responsibilities. CNCAN has the right and obligation to ensure that safeguards are applied, in accordance with the terms of the safeguards agreement, on all source or special fissionable material in all peaceful nuclear activities within the State, under its jurisdiction or carried out under its control anywhere, for the exclusive purpose of verifying that such material is not diverted to nuclear weapons or other nuclear explosive devices. CNCAN has built a strong primary and secondary legislation in order to have a strong legal framework to fulfill the NPT, Safeguards Agreement and Additional Protocol requirements.

In respect of the non-proliferation issues CNCAN has as a major goal to strengthen the effectiveness and to improve the efficiency of the safeguards system. Also closer co-operation between the IAEA and CNCAN as coordinator of the national system of accounting for and control of nuclear material has been developed by organizing international and national seminars on the implementation of safeguards and the additional protocol. After the entry into force of the Additional Protocol, CNCAN prepared appropriate declarations and answers to the relevant IAEA questions in order to obtain a drawn conclusion of the absence of undeclared nuclear material and nuclear activities within Romania territory. The IAEA evaluated in Romania not only the results of its nuclear material related activities under the Safeguards Agreement but also the results of its broader, more qualitative, evaluation and verification activities under the Additional Protocol. CNCAN assured that the IAEA inspectors have complementary access according to the Additional Protocol as requested in accordance with the provisions of the Safeguards Agreement and the Additional Protocol and cooperated in resolving in a timely manner, any questions or inconsistencies identified by the IAEA during its verification and evaluation activities. During 2001-2006 the IAEA performed in Romania 21 complementary accesses and took 21 samples.

Training for the implementation of the Additional Protocol started in 2001 with a national seminar organized by the IAEA and CNCAN. In 2003 IAEA and CNCAN organized a regional seminar on the Additional Protocol implementation during 2004 - 2005 national seminars.

* Only an abstract is presented here, as the full paper was not available.

Experience and challenges on safeguards practices and approaches for BAEC 3 MW TRIGA Mk-II research reactor and other establishment of Bangladesh

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Abstract. Bangladesh is deeply committed to nuclear non-proliferation signing and accesses to different unilateral protocols, agreements and treaties like Nuclear Non-Proliferation Treaty (NPT), Comprehensive Test Ban Treaty (CTBT), Safeguards Agreements, Nuclear Cooperation Agreements with the USA and other countries etc. are the manifestations of such commitment. The first of such agreements, the NPT was signed in August 1979. NPT is a national commitment that the signatory country will not engage in activities related on nuclear detonations. Subsequently a bilateral agreement entitled "Safeguards Agreements" was signed with the International Atomic Energy Agency (IAEA) in June 1982. This provides for international verification of facilities and balancing of nuclear materials. 3MW TRIGA Mark-II Research Reactor Facility of Bangladesh Atomic Energy Commission (BAEC) is inspected physically by the IAEA Safeguards Inspectors on an annual basis. For this purpose, a subsidiary arrangement was made with the IAEA, which had defined the scope of such verification. Bangladesh has also signed a Nuclear Cooperation Agreement with the USA on September 17, 1981, which facilitated export of nuclear technology from USA to Bangladesh. Bangladesh also signed another bilateral agreement entitled "Protocol Additional to the Safeguards Agreements" with the IAEA in March 30, 2001. The main purpose of this agreement is to provide the IAEA with information on "so called dual-purpose materials and facilities" including the front end and back end of the nuclear fuel cycle. Its enforcement required filing of an initial declaration, draft of which has been sent to the Ministry of Science, Information and Communication Technology (MOSICT) for approval and transmittal to IAEA through the office of the permanent mission in Geneva. This protocol till to-date is the highest level of verification under the non-proliferation regime. The commitment of Bangladesh to non-proliferation is adequately reflected in this document. IAEA would decide on the modalities, frequency and scope of conducting its verification program on the basis of the declaration. It may be pointed out that the IAEA Inspectors are as a rule designated subject to their acceptance by the Government. After intensive preparatory works, Bangladesh signed the CTBT in October 1996. The treaty was ratified in March 2000. This treaty provides for an additional commitment of the signatory country that it would not conduct any nuclear detonation in the future. It may be pointed out that Bangladesh is one of the 34 countries of an exclusive list relevant to the treaty. It was envisaged that the treaty would not come into force until and unless each of these states signs the treaty. This is a manifestation of Bangladesh's potential nuclear capabilities and by signing this Bangladesh has adequately and convincingly reiterated its commitment to the international community about its peaceful intentions. This paper describes Bangladesh's experience to implement IAEA safeguards including various legal and institutional agreements made for the rectification and smooth implementation of NPT, safeguards and Additional Protocol.

1. Introduction

The safeguards system comprises measures by which the IAEA Secretariat independently verifies the declarations made by States about their nuclear materials and activities. These measures are implemented under the various types of agreements and protocols. The technical objectives of safeguards are the timely detection of diversion of significant quantities of nuclear material from peaceful uses to the manufacture of nuclear weapons or other nuclear explosive devices or for purposes unknown; and the deterrence of such diversion by the risk of early detection [2]. Atomic Energy Research Establishment (AERE) is the largest R&D establishment of the BAEC. It is about 40km away from the Dhaka city. The 3 MW TRIGA Mark-II research reactor of BAEC has been operating since September 14, 1986. The reactor is used for radioisotope production (^{131}I, ^{99m}Tc,^{46}Sc), various R&D activities, manpower training and education [3].

As a facility and material balance area, AERE facility and other than AERE facility have been designated by the Agency (IAEA) as BDA-, BD-A and BDZ- respectively. AERE facilities includes BDA- and material balance area BD-A. On the other hand, BDZ- presently includes Bangladesh Cancer Research Institute and Dhaka Medical College Hospital in Dhaka city. The facilities at AERE which are of concern from safeguards point of view are the 3MW TRIGA Mk-II research reactor and the radioisotope production laboratory. The reactor facility has fuel elements and fission chambers where nuclear materials are used. Where as the radioisotope production laboratory has isotope transfer cask made from depleted uranium. Accounting of nuclear materials is carried out by maintaining and routinely updating several records as recommended by IAEA and standards prescribed/adopted by the Nuclear Safety and Radiation Control Regulations of BAEC.

Nuclear fuel was first imported into the country in 1985. Fuels were loaded into the reactor core on 13 September 1986. IAEA safeguards inspectors visited the facility for the first time in 1986. Since then the facility is inspected regularly on annual basis by two or three inspectors at a time. The inventories of nuclear materials at AERE include several kilograms of 19.7% enriched uranium in the form of TRIGA fuels, several kilograms of depleted uranium in the form of shielding for the radioisotope transfer cask and a few grams of 99.3% enriched uranium in the form of fission chambers. A few of the fuels are still fresh and they have been stored in the fresh fuel storage room. Rest of the fuels is loaded into the reactor core. The facility has not yet generated any spent fuel. According to the safety analysis report (SAR), the initial core loading is capable of producing about 1278 megawatt-days of energy. But as of now, the figure for the total cumulative burn-up stands only at around 432 megawatt-days with the total operating hours of about 5496 hrs. However it is expected that after completion of the ongoing reactor system-upgrading program, utilization of the reactor will be increased manifolds and as result the burn up figure will rise sharply.

BAEC with its limited resources is always trying hard to strengthen the safeguards and physical protection programs around its reactor and associated facilities. Presently AERE does not have any facility to measure the actual amount of nuclear materials in its possession. For ascertaining the amount of the inventory, AERE needs to depend solely on the information provided by the manufacturer/supplier of the product containing the nuclear material.

2. Material Accounting and Reporting

The facilities at AERE which are of concern from safeguards point of view are the 3MW TRIGA Mk-II research reactor and the radioisotope production laboratory. The reactor facility has fuel elements and fission chambers where nuclear materials are used. Where as the radioisotope production laboratory has isotope transfer cask made from depleted uranium.

Accounting of nuclear materials is carried out by maintaining and routinely updating several records as recommended by IAEA and standards prescribed/adopted by the Nuclear Safety and Radiation Control Regulations of BAEC. The reports that are maintained at the research reactor facility mainly include the followings:

1. Fuel elements history file,
2. Fuel element transfer sheet,
3. KMP General ledger for nuclear materials,

4. KMP Sub-ledger for nuclear materials,
5. Physical inventory listing (PIL),
6. Materials balance record (MBR),
7. Inventory change report (ICR),
8. Fuel burn-up record along with data on operating hours, full power operating hours, etc.
9. Verification of inventories of declared nuclear material and under certain types of agreements, of non- nuclear material and equipment, and of inventory changes at the facility.

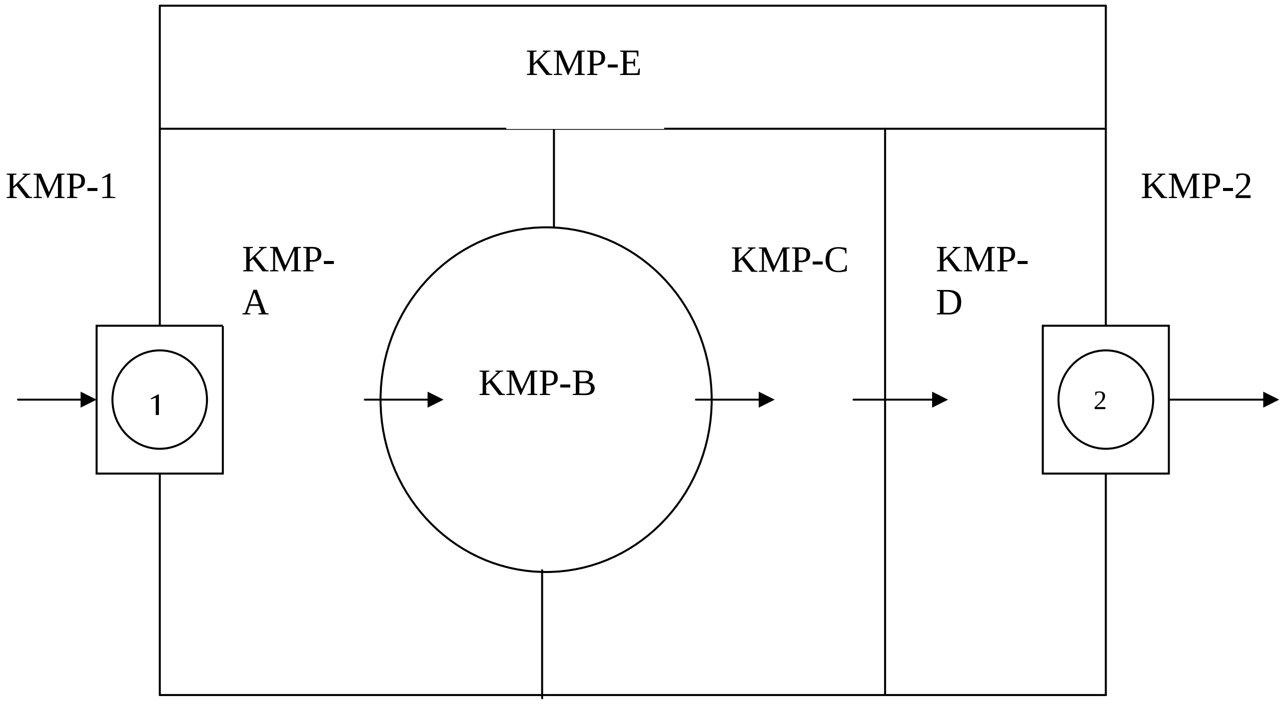

Figure 1: Flowchart for Safeguards Inspection

<u>Legend:</u>
KMP-1: Fresh fuel receipt, KMP-2: Spent fuel storage, KMP-A: Fresh fuel storage, KMP-B: Fuel in core, KMP-C: Spare fuel storage in reactor tank, KMP-D: Spent fuel storage pits, KMP-E: Other location

Figure 1 shows the schematic flow chart for nuclear material of TRIGA reactor facility. KMP-1: BAEC TRIGA Research Reactor Facility (TRF) had received fresh fuel only one time (which are actually received from General Atomics, USA the fuel/reactor supplier) in 1985. KMP-2: BAEC TRF did not generate any spent fuel till now. There is no permanent spent fuel storage facility in TRF. KMP-A: A fresh fuel storage room, adjacent to the east side of the reactor hall, provided for the storage of fresh fuel elements. Fresh fuel elements are kept in fuel storage drums and these drums are storage in the fresh fuel storage room. KMP-B: At first fresh fuel loaded in the reactor core in 1986 and one fresh fuel loaded in the core in 1992 because one fuel dropped from fuel handling tool during fuel inspection. KMP-C: there is 3 fuel storage racks, each capable of holding 10 fuel elements, are located under water along the walls of the reactor tank to provide temporary storage of fuel moderator or graphite dummy elements. KMP-D: There is 3 storage pits, located in the reactor room floor, have been provided for temporary storage of fuel elements as well as radioactive samples. The storage pits were designed to hold 19 TRIGA LEU fuel elements in each pit.

AERE has a committee called "Nuclear Safety and Safeguards Committee". The chairman of this committee is the Director General (DG), AERE. The purpose of the committee is to audit, review, evaluate, recommend and approve all programs, actions and issues involving inventory accounting and safeguards of all nuclear materials including spent fuels stored and/or used in AERE as per the subsidiary arrangements between the Government of the People's Republic of Bangladesh and the IAEA under NPT treaty. All these reports and declarations are submitted to the IAEA through the International Affairs Division (IAD) of BAEC.

3. Status of IAEA Safeguards Inspection

Bangladesh is committed for peaceful use of nuclear energy. As a part of its commitment it has signed and rectified almost all the international treaties, agreements, protocols, in connection with the nuclear non-proliferation. It has also signed relevant international convention related to nuclear and radiation safety. Bangladesh will continue all its efforts in order to strengthen the nuclear non-proliferation regime so as to achieve a peaceful world for the generations to come. To achieve the goal, it will welcome all sorts of cooperation and help from various International Agencies and technically developed nations.

Bangladesh is deeply committed to nuclear non-proliferation. Signing and accesses to different unilateral protocols, agreements and treaties like NPT, CTBT, Safeguards Agreements, Nuclear Cooperation Agreements with the USA and other countries etc. are the manifestations of such commitment. This has helped establish its impeachable credentials in the use atomic energy solely for peaceful purposes. Bangladesh signed the Nuclear Non-proliferation Treaty (NPT) on September 1979. In connection with the NPT, it signed Safeguards Agreement with the International Atomic Energy Agency (IAEA) on 11 July 1982. Bangladesh also signed the Protocol Additional to the Safeguards Agreements at the end of year 2000. Subsequent to this, necessary subsidiary arrangements were made with IAEA in order to facilitate inspection of the TRIGA research reactor and associated facilities of the Atomic Energy Research Establishment (AERE) at Savar, Dhaka by the IAEA safeguards inspectors. AERE is the largest R&D establishment of the Bangladesh Atomic Energy Commission (BAEC). It is the only establishment in the country where nuclear materials are being used. As a facility and material balance area, AERE facility and other facility have been designated by the Agency (IAEA) as BDA-, BD-A and BDZ- respectively. A detail site map of AERE (shown in Figure 2) is developed with the help of GPS (Global Positioning System) and submitted to IAEA.

Nuclear fuel was first imported into the country in 1985. Fuels were loaded into the reactor core on 13 September 1986. IAEA safeguards inspectors visited the facility for the first time in 1986. Since then the facility is inspected regularly on annual basis by two or three inspectors at a time.

4. Status of Material Inventories

Nuclear material accounting records of all nuclear material on inventory and inventory changes are maintained by operator for each facility under safeguards. The facility nuclear material accounting data and also safeguards-relevant design information, are transmitted

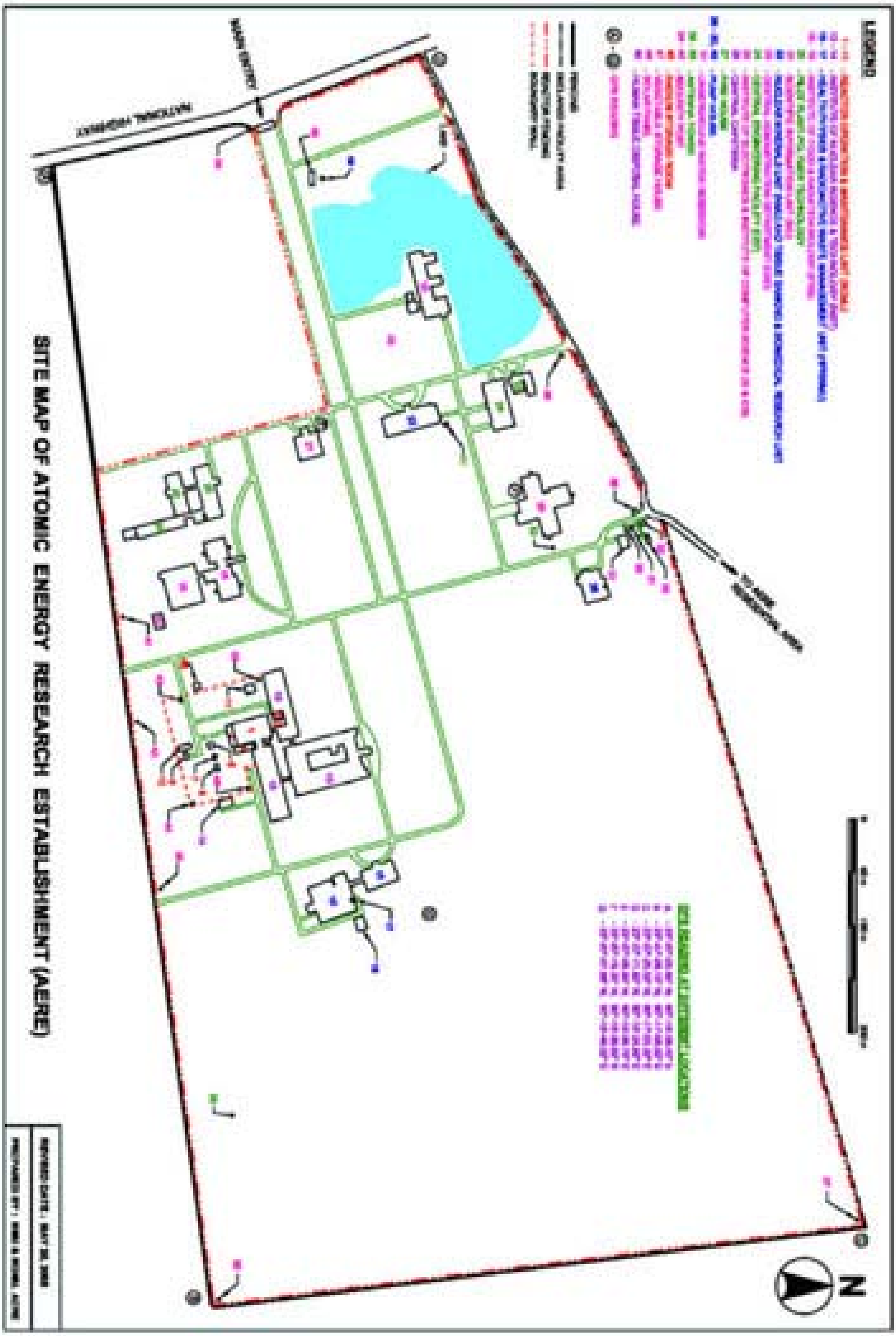
LEGEND
SITE MAP OF ATOMIC ENERGY RESEARCH ESTABLISHMENT (AERE)
N

Figure 2.

through the State authorities to the Agency [2]. The inventories of nuclear materials at AERE include several kilograms of 19.7% enriched uranium in the form of TRIGA fuels, several kilograms of depleted uranium in the form of shielding for the radioisotope transfer cask and a few grams of 99.3% enriched uranium in the form of fission chambers. A few of the fuels are still fresh and they have been stored in the fresh fuel storage room. Rest of the fuels are loaded into the reactor core. The facility has not yet generated any spent fuel. According to the safety analysis report (SAR), the initial core loading is capable of producing about 1278 megawatt-days of energy. But as of now, the total cumulative burn-up stands only at around 432 megawatt-days with the total operating hours of about 5496 hrs. However it is expected that after completion of the ongoing reactor system-upgrading program, utilization of the reactor will be increased manifolds and as result the burn up figure will rise sharply.

5. Strengthening Safeguards and Physical Protection Program

BAEC with its limited resources is always trying hard to strengthen the safeguards and physical protection programs around its reactor and associated facilities. As a part of this strengthening program, close circuit TV (CCTV) system has been installed at the reactor facility. US DOE also has provided physical protection devices such as balanced magnetic switch, infrared motion detector, electronic access control system etc and successfully installed in the reactor facility. Similar physical protection devices were also installed at central waste processing and storage facility (CWPSF) and ^{60}Co source room of the Institute of Food and radiation Biology (IFRB) of AERE.

6. Conclusion

Bangladesh is committed for peaceful use of nuclear energy. As a part of its commitment it has signed and rectified almost all the international treaties, agreements, protocols, in connection with the nuclear non-proliferation. It has also signed relevant international convention related to nuclear and radiation safety. Bangladesh will continue all its efforts in order to strengthen the nuclear non-proliferation regime so as to achieve a peaceful world for the generations to come. To achieve the goal, it will welcome all sorts of cooperation and help from various International Agencies and technically developed nations.

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Implementation of safeguards commitments: A Cuban experience

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Abstract. Cuba has been involved in the implementation of IAEA's safeguards for around 25 years.

From the very beginning of its nuclear power program, 66-type safeguards agreements for each of its nuclear facilities were subscribed between Cuba and the Agency since 1980. Later, in 1999 and following the establishment of measures for strengthening effectiveness and improving efficiency of the IAEA's safeguards system by the international community, Cuba signed an additional protocol to its safeguards agreements in force at that time.

During this period (1980-2004) Cuba established a State System for Accounting for and Control of Nuclear Materials (SSAC) along with its legal support. For almost 14 years the IAEA carried out inspections aimed at verifying the compliance with its safeguards commitments by Cuba.

Cuba made a further commitment towards a global and regional nuclear non proliferation, peace and international security, when on September 14th, 2002 decided to become a member of both the Treaty for the Proscription of Nuclear Weapons in Latin America and the Caribbean, and the Treaty on the Non-Proliferation of Nuclear Weapons.

In September 2003, Cuba decided to sign, simultaneously with the Agency, a Safeguards Agreement and its Additional Protocol, which entered into force on June 3rd, 2004.

The strengthening of its SSAC and its legal support along with the negotiation and instrumentation of its new safeguards commitments has characterized the present stage. Since 2004 the IAEA has been involved in the verification of the correctness and completeness of the information provided by Cuba as part of its obligations.

1. Introduction

An important agreement was signed by Cuba and the former Soviet Union in April 1976. Such agreement paved the way to develop a nuclear program which main purpose was the electricity generation. As a token of the pacific character this program bore, from 1980, Cuba subscribed along with the IAEA 66-type safeguards agreements to each of its nuclear facilities which were: a nuclear power plant [1], a research nuclear reactor [2] and a zero-nuclear power reactor [3].

The safeguards became institutionalized at the beginning of the Cuban nuclear program. Among the system of regulatory measures to be enforced under the Decree-Law No. 56 of the Council of State "For the regulation of the pacific use of nuclear energy" issued the 25th of May 1982, the State System of Accounting for and Control of Nuclear Materials (SSAC) stands out.

During this first stage of the IAEA safeguards applications the negotiations and entry into force of the subsidiary arrangements, information design, import notifications, and the verification in situ of the enforcement of the commitments assumed by virtue of those specific safeguards during fourteen years stand out from a technical point of view.

Fig. 1. IAEA inspectors performing a verification activity.

During this first stage it is important to highlight, at a regional level, the Cuban statement made during the First Ibero-American Summit held in Guadalajara, Mexico, July, 1991 that stated its willingness to be part of the Treaty for the Proscription of Nuclear Weapons in Latin America and the Caribbean (Tlatelolco Treaty) after the incorporation of the remaining countries to such treaty. Cuba on March 25th, 1995, signed the Treaty and its three amendments.

In 1995, after almost eighteen months of establishing negotiations with the Agency, which being previously aware of the non-reception of any of the elements that may be subject to safeguards and after Cuba had announced its decision of definitely cancelling the project, the IAEA Board of Governors had the termination of the Agreement on safeguards relative to research nuclear reactor approved [4].

Far beyond our regional context and after the completion of a negotiation process by the international community in June, 1997 that was focused on the strengthening of the effectiveness and efficiency of the safeguards system, on October 19th, 1998, Cuba signed and additional protocol to its safeguards agreements in force becoming the first and unique non-signing country of the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) that adopted such an action.

2. Current commitments

Cuba took a transcendental step towards the global and regional nuclear non-proliferation, peace, and international security when it announced its decision to ratify the Tlatelolco Treaty and adhere to the NPT on September 14th, 2002, and implementing the decision during the posterior months of October and November of the same year. Cuba, once more, reaffirms its political will and commitment with the fostering, strengthening and consolidation of multilateralism and international treaties.

Cuba simultaneously acceded to sign a comprehensive safeguards agreement [5], and its additional protocol [6] with the Agency, which came into force as of June 3rd, 2004.

The present stage has been characterized by the SSAC strengthening and its legal support, the negotiation and instrumentation of the enforcement of the new commitments assumed.

The IAEA, during these two years, has been involved in the verification of the completeness and correctness of the information provided by Cuba.

2.1 National legal base

Upon the signing of the Tlatelolco Treaty by Cuba in March, 1995, the update of the legal support of the SSAC resulted in the adoption of the Decree No. 208 of the Executive Committee of the Council of Ministers “On the State System of Accounting for and Control of Nuclear Materials” of June 10th, 1996 and Resolution No. 62/96 of the Ministry of Sciences, Technology and the Environment (CITMA in Spanish) “Regulation for the Accounting for and Control of Nuclear Materials”, of July 12th, 1996. Its structure and scope were designed, among other objectives, to meet international commitments of comprehensive safeguards.

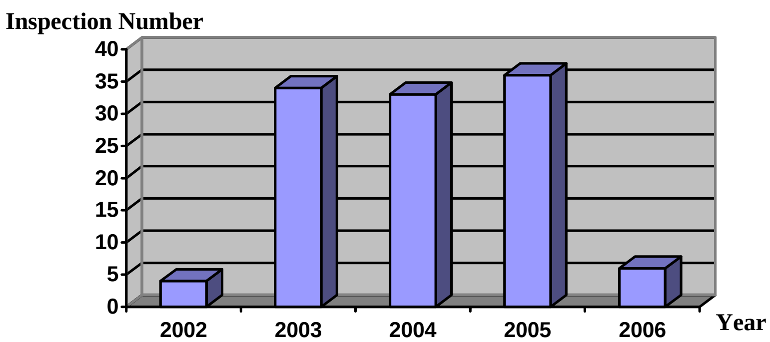
Currently, the update of the Regulation is in the process of being approved so as to broaden its scope to the obligations assumed by the Additional Protocol.

2.2 Impact of comprehensive safeguards agreement

The establishment of the national inventory of nuclear materials was the result of a survey made internally by each ministry following the CITMA instructions and training, the ministerial sworn statements and the process of verification in situ.

During this stage the number of annual domestic safeguards inspections carried out up to that moment increased almost eight times more than the previously scheduled. On one hand, this was motivated by the initial taking and verification of the national inventory of nuclear materials, and on the other hand, by the commitments related to the additional protocol.

Table 1. Performance of the domestic safeguards inspections carried out during the last five years.



According to the safeguards assumed, Cuba has adopted a system of control that is shown in Fig. 2.

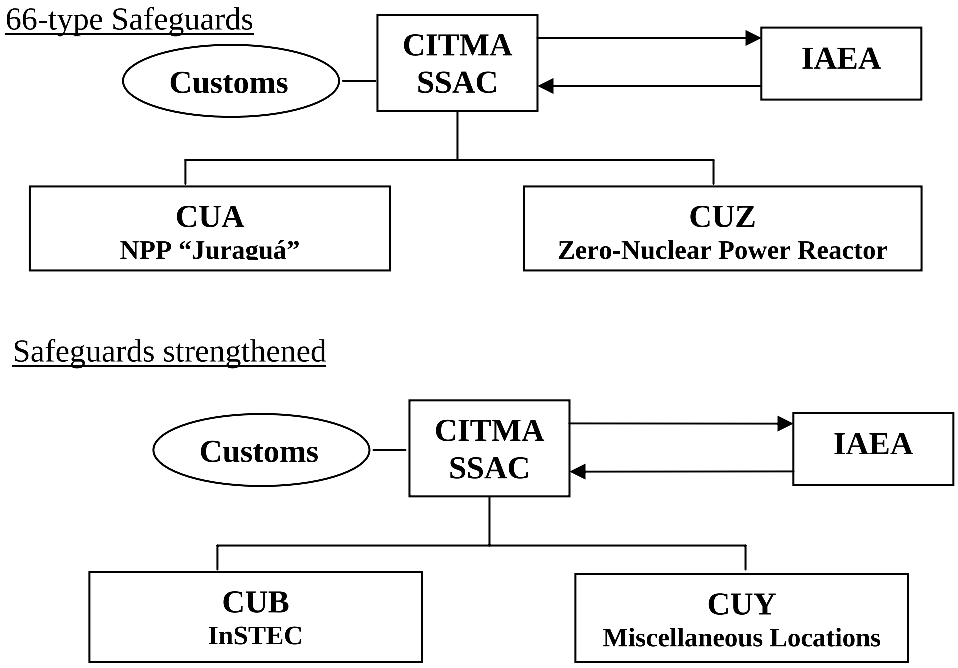


Fig. 2. Organization chart of the system of control of safeguards implemented.

2.3 On the IAEA-related activities

Cuba used the IAEA safeguards forms R.04/ b, c and d [7] to report the initial inventory list of nuclear materials to the Agency. In addition, an executive report was submitted including the total quantities, by items, of nuclear material with pictures included.

Up to the present the following reports and declarations have been submitted to the Agency:

Table 2. Summary of the reports and declarations submitted to the IAEA.

	Quantity
Reports by virtue of the Safeguards Agreement	
• Initial Physical Inventory List (IPIL).	3
• Inventory Change Report (ICR).	3
• Physical Inventory List (PIL).	9
• Material Balance Report (MBR)	2
TOTAL	**17**
Declarations by virtue of the Additional Protocol	
• Initial Declaration.	1
• Quarterly Declarations.	9
• Annual Declaration update.	2
• Other declarations.	1
TOTAL	**13**

The Protocol Reporter electronic support designed by the Agency has been used satisfactory for the preparation of the Additional Protocol’s declarations.

As part of the verification process for the completeness and correctness of the information provided by Cuba, two Agency's safeguards inspections have been carried out to the country. During these inspections the IAEA implemented measures contained in the strengthening of the safeguards such as:

- Environmental sample takings
- Complementary access following a 24 hour advance notification

In addition to the aforementioned also the Agency requested information from Cuba in order to cross check the data provided by third parties. Cuba successfully submitted it in due time.

Fig. 3. Examples of the activities developed within the framework of the new safeguards agreements.

2.4 Commitments with the Agency for Nuclear Weapons Proscription in Latin America and the Caribbean (OPANAL in Spanish)-related activities for

Cuba has reported its corresponding declarations as to the non-occurrence of any activity banned by provisions stated in the Treaty for a six-month period complying with what is established in Article 14, item 1, of the Tlatelolco Treaty.

As a recognition that Cuba prioritize the matters related to nuclear disarmament, and to reaffirm its political will towards an active contribution concerning the mere application of the international legally binding instruments, the XVIII Ordinary Period of Sessions of the OPANAL General Summit was held from the 5^{ht} to the 6^{th} of November, 2003 in Havana.

3. Conclusions

Throughout these 25 years of implementing the IAEA safeguards, the Agency has been able to confirm and acknowledge the Cuba's compliance with its international commitments. In addition to that, the verification activities have also been performed in a cooperative and transparent manner.

The nuclear materials that Cuba has are being basically used for medical applications, teaching purposes, and in industries focusing on peaceful non-nuclear applications. Currently, the Cuban nuclear program is not based on the development of nuclear power plants. However, Cuba is making use of the wide peaceful applications of nuclear techniques in favour of its population and its economics development.

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[3] INFCIRC/311, of October 7th, 1983. Agreement between Cuba and the Agency for the application of safeguards in connection with the supply of a Zero-Power Nuclear Reactor from the Hungarian People's Republic.

[4] GOV/INF/763 of March 14th, 1995. Termination of the Agreement of September 25th, 1980 between Cuba and the Agency for the application of safeguards in connection with the supply of a Nuclear Research Reactor from the Union of Soviet Socialist Republics.

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[6] INFCIRC/633/Add.1, of September 18th, 2003. Additional Protocol to the Agreement between the Republic of Cuba and the International Atomic Energy Agency for the application of safeguards in connection with the Treaty for the Prohibition of Nuclear Weapons in Latin America and the Caribbean and the Treaty of the Non-Proliferation of Nuclear Weapons.

[7] Forms R.04/b "Facility Summary", R.04/c "State Summary" y R.04/d "Explanatory Note".

Coordination improvement on safeguards application between ABACC and IAEA

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Abstract. The cooperation within the framework of the Quadripartite Agreement is an on going activity and the success of a regional and an international safeguards system working together relies on the good coordination between the parties. Taking into account the requirements for both efficient and effective safeguards, as well as the requirement for both Agencies (ABACC and IAEA) to minimize inspector resources and intrusiveness while applying safeguards and the need for each Agency to be able to reach its own independent conclusions, the coordination on safeguards application between ABACC and IAEA becomes complex and sometimes difficult.

This paper presents a description of the main co-ordination activities achieved between ABACC and the IAEA in the framework of the Quadripartite Agreement, INFCIRC/435, during the last four years. Introduction of new policies, new safeguards approaches and activities at sensitive installations, optimization of inspection effort, establishment of inspection procedures for non-routine activities, implementation and use of safeguards equipment on a joint use basis, coordination of equipment supply and maintenance, and implementation of new techniques in the field are considered. The paper presents an overview on the planning, implementation and analysis of results from the inspection activities and describes how joint inspection activities have contributed towards the safeguards system. Some views are also presented on further improvements and what is foreseen in the near future within the framework of integrated safeguards.

INTRODUCTION

The Quadripartite Agreement states that ABACC and the IAEA shall apply nuclear safeguards in a cooperative manner and states that both agencies shall avoid unnecessary duplication of safeguards activities. In order to obtain the maximum of efficiency and effectiveness, using the minimum effort and assuring independent conclusions from each organization, coordination between ABACC and the IAEA while applying safeguards plays a major role.

In the last 4 years a significant effort has been made by both Agencies to improve Coordination. Many accomplishes have been reached in some areas, among which we can list:

a- Documentation and communication framework area:
  - Guidelines for inspection coordination between ABACC and IAEA;
  - Procedures for secure communication between ABACC and IAEA;
  - Procedures for Nuclear Material Reporting from the States to the Agencies;
  - Procedures for Common Use of Equipment;
  - Procedures for specific inspections (sensitive installations).

b- Concept and evaluation area:
  - Participation in special groups concerned with particular installations;
  - Development of safeguards approaches and procedures for specific sensitive installations;
  - Implementation of new policies;
  - Reclassification of installations by type;
  - Domestic transfer verification approach;
  - Facility Attachment negotiations.

c- Operations area:
  - Planning of Inspections;
  - Optimization of PDI with emphasis on effectiveness and inspection resources;
  - The application of Joint Use of Equipment during inspections.

d- Technical and operational support area:
  - Planning of equipment acquisition between the Agencies;
  - Comparison of DA analysis results;
  - Data analysis from NDA results applied to error calculation for equipment used at facilities;
  - Joint training of inspectors on joint use of equipment and procedures for inspections;
  - Joint training on specific inspection approaches (sensitive installations).

Most of the items listed above are “on going” as long as the Agencies apply safeguards based on the Quadripartite Agreement. New items that may contribute for improving the safeguards system are always considered to become active items in the coordination agenda.

RECENT IMPROVEMENTS

Although a lot of improvement has been made, the application of safeguards in the framework of the Quadripartite Agreement still has room for further cooperation which will improve the efficiency and effectiveness. Recently, more specifically in the last three years, many issues were discussed to improve coordination. Some points are presented below.

A - Introduction of new policies and new safeguards approaches.

In order to strengthen the safeguards systems, new or revised policies from the IAEA, and some from ABACC, have been developed and implemented. Even though the new measures fulfill the Legal framework of the Quadripartite Agreement, a measure may trigger a large impact on the Operator and State Parties.
ABACC and IAEA have conducted discussions on these new measures on an efficient and relatively fast mode. In some cases, discussions with the State Parties are necessary.

B - Establishment of the Guidelines for Joint Inspection.

One of the main objectives behind the Joint Inspection concept is that all inspection activities are performed only once by a joint IAEA and ABACC team. This means that for every activity to be performed by inspectors from both organizations it should be executed together and the results should be shared. Note that the number of inspectors per agency in each activity will be dependent on the job complexity. Part of this concept has already been implemented, such as in the use of surveillance systems and some containment equipment in Common Use.

ABACC and the IAEA have been negotiating and preparing the framework for such implementation. Among that, the necessary steps and actions are the following:

- The **development and agreement** of well defined **Procedures for Inspection** activities. For all relevant installations to be inspected a detailed procedure for each type of inspection and activity is established in advance in order to permit that the activities to be performed in the field be executed only once and fulfill the ABACC and IAEA requirements. These procedures will constitute a set of documents called **Guidelines for Joint Inspection.**

- **A full implementation of an Agreement for Common Use of Equipment**, which means to have almost all equipment in the field being shared by both Agencies and to have the inspectorates personnel trained. In addition, the procedures for calibration and service for such equipment shall be agreed;

- **A Common Accountancy Inspection Procedure** that allows both agencies to perform the same activity in the field and transfer data to their own different databases. This system for accountancy auditing has been implemented between IAEA and ABACC and a common database is created before each inspection to be loaded in a Common Audit Software. The outcome results are saved in a format that allows both organizations to load the results on their own systems ;

Again, all actions are taken with the aim of introducing savings where possible without loosing the ability for each organization to reach its own conclusion on the results of the inspections.

C - Optimization of inspection effort in all activities.

The main objective is to perform each inspection activity, fulfilling the criteria of both agencies with a minimum effort. To achieve such goals the following actions are taken:

- Both agencies agree on the number of inspectors per organization in each inspection (PDI) and necessary frequency of inspections. Both are defined on the basis of the analysis of the facilities or LOFs, considering the type of process in the installation, the inventory of nuclear material, and the complexity and number of activities to be performed in each inspection,
- Coordination of inspections schedule taking into consideration the agreed PDI, which allows the split of a mission team during the mission;
- Coordination on unannounced inspections considering points such as triggering of the inspection and communication procedures. This will permit that both agencies may take part in almost all UI inspections avoiding an excessive inspection effort, even fulfilling the criteria of both organizations.

All actions are taken without loosing the ability of each organization to reach its own conclusion on the results of the inspections. However improvements on the saving of inspection effort can still be made.

C - Establishment of the equipment supplying and maintenance.

This is one of the coordination improvements between ABACC and the IAEA already in force since the beginning. However, the large number of tasks involved and complexity of the technical coordination require permanent discussions and updated procedures.

Usually, during the coordination meetings both organizations agree on a long term framework -- with the span of around four years --, on which equipment will be necessary to be supplied, based on the commissioning of new installations, necessary replacements and new technology instruments that may replace old ones with better performance. After that, the organizations agree on which equipment each agency is going to supply.

Following the action list, the organization in charge makes a more detailed schedule for each equipment to manage its procurement, acquisition, installation and test. In the test phase both organizations have the right to participate and the Joint Use Procedure is discussed and implemented.

With regard to maintenance, the organization in charge of the acquisition is also responsible for the maintenance in order to keep total control on the equipment performance.

Even though sharing the costs of equipment evenly is not a fixed rule, the balance in the cost decision is followed between the two organizations.

D - Implementation of new technology techniques

As long as new equipment and services applying new technologies are available for safeguards application, the IAEA and ABACC make all the effort to use these new devices. In this field we can highlight the following improvements:

- The application of digital technology in safeguards equipment replacing analogic systems, mainly in surveillance, containment and non-destructive verification systems. This new equipment usually speeds up field activities and allows an easy exchange of data and analysis of results. The back office storage duty is also highly facilitated.
- **The implementation as much as possible of electronic media** on interchange of data dealing with information, communication and conclusions which flow between Agencies and in some cases communication with the States Parties. This requires the setting up of systems and procedures among the all parties;
- During the last four years the use of Swipe Sampling Technique has increased. This powerful technique has strengthened the safeguards applied to sensitive installations, like enrichment and reprocessing, and also allows that the safeguards approach and effort be optimized. Nevertheless, the result analysis and understanding requires from IAEA and ABACC a special coordination –where, when and how to get samples - and customized field procedures.

NEAR FUTURE IMPROVEMENTS TO BE DEALT WITH

As previously mentioned, improvements in coordination shall be a permanent goal to be pursued by both organizations. Nevertheless, we may state some comments on what we expect to be the main effort in the near future.

First - As described in the item **Joint Inspection Framework,** full implementation continues to be the more immediate and more rewarding to accomplish. As soon as all the points to implement this framework are agreed and put into force, the following benefits may be envisaged:
- safeguards will be applied much more efficiently considering that that the activities will not be duplicated;
- technical confidence between the inspectors from both organizations will increase. This will, of course, provide space for more optimization, acting as a feed-back correction of any inefficiency.

Second - **SNRI –** Short Notice Random Inspections is in an implementation phase. Even though it is being introduced at Fuel Fabrication Plants and Conversion Plants, its application is foreseen on other types of installations, such power reactors and enrichment facilities.

The coordination on these types of inspections can be impacted due to many factors, such as when the inspection is to be triggered, who is going to trigger and which activities should be done at each inspection.

Third - The effective application of **DIV** activities. IAEA and ABACC recognize that an effective DIV will improve the safeguards. Both organizations are working together to have an effective DIV procedure to be applied in field. Some specific installations will have specific procedures and the activities to be executed should be within the framework of the Quadripartite Agreement.

Fourth - The approval and qualification to use new technology systems, such as remote monitoring and verification systems with data transmission. These applications usually require coordination between the agencies, on which data and system will be shared, and also the agreement and special requirements from the State in which it is applied.

Finally, in case of the Additional Protocol coming into force, some challenging points arise which require further definition such as:

- the exact role of ABACC on the Additional Protocol. This matter is currently under review, specifically between Brazil, Argentina and ABACC.
- the flow of information among the parties;
- the new activity that each party has to perform, being in the office or in the field, alone or together;
- the way safeguards results are to be presented;
- new common procedures for the IAEA and ABACC

Coordination on the application of the Additional Protocol will be a major undertaking, with many new issues on implementation and operational functioning to be addressed, which will surely have influence in the traditional or current coordination measures that both organizations have set up in the last years. Nevertheless, the advance in the relationship that the organizations have today not only facilitates the overcoming of these undertakings but also offers many tools to conduct negotiations and settings of the protocol.

Finally, the preparation of ABACC safeguards system to have some assessment by the IAEA auditing should also be considered as a near future task to improve coordination.

CONCLUSIONS

Coordination between ABACC and the IAEA is an integral part of the Quadripartite Agreement. Since the beginning of safeguards implementation under this agreement, both organizations have put a lot of effort to implement this.

In the last three years, significant advance was made in the coordination between the organizations. The data obtained from safeguards application shows that the safeguards are being applied in a more efficient way and manner, keeping high standards of efficiency.

Furthermore, coordination improvement has allowed that both increase the knowledge among their safeguards systems. This helps to build confidence between the IAEA and ABACC, by promoting a regenerative feedback that makes coordination and safeguards application more effective.

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Migration of safeguards instrumentation: Challenges and opportunities

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Abstract. Development of new safeguards instrumentation for use by the International Atomic Energy Agency (IAEA) is spurred by a number of factors: Technical issues, including the emergence of new technologies and the inevitable obsolescence of system components, as well as policy-related issues including changes in safeguards policies, new approaches to the safeguarding of excess material, and the introduction of new types of facilities. To cope with these issues, development projects are required that migrate between current safeguards systems and state-of-the-art technology. These projects usually involve IAEA staff as well as Member States and their Support Programs (MSSP), the private sector, and research and development institutions. One such development project currently underway is the migration between the IAEA's DCM-14 based surveillance system to the Next Generation Surveillance System (NGSS). The challenges and opportunities of the NGSS project include those factors that prompted the migration, the development of user requirements, and the selection of appropriate development partners and funding resources. This paper will outline the challenges, issues, opportunities, and synergies that relate to safeguards instrumentation migration, using the NGSS project as an example. The problems necessitating new development will be outlined, as will the management of these problems from project initiation until new instrumentation is available for replacement. The generation of user requirements will be explained in detail. Focus will be given to how to best manage this process in order to produce requirements that will be useful both during the development project and after the migration is completed. The challenges and opportunities in the selection and procurement process will then be highlighted, followed by the structure of the actual development project. Finally, a summary of key steps that can help alleviate migration problems and the pressure to quickly complete development projects will conclude the paper.

INTRODUCTION

To verify that Non-proliferation Treaty (NPT) signatory parties comply with their safeguards agreements, the International Atomic Energy Agency (IAEA) relies on a strict inspection regime with facility visits on a 'minimum time needed to divert' basis. IAEA inspectors are supported by instrumentation installed at safeguarded facilities to gather information to establish Continuity of Knowledge on a Member State's treaty commitment observance.

The design of safeguards instrumentation is determined by a variety of technical and political factors including changes in IAEA user requirements and safeguards policies, emergence of new technologies, and component obsolescence. These force the IAEA to migrate from one generation of instrumentation towards another including a new safeguards solution. The according development efforts involve IAEA staff members, support from Member States Support Programs (MSSPs), research and development bodies, and commercial partners.

The migration between the aging Digital Camera Module 14 (DCM-14) system (the standard surveillance tool of the IAEA since 1998) and the Next Generation Surveillance System (NGSS) is a major ongoing development effort. Using NGSS as an example, this paper will outline the issues impacting migration. A summary of the opportunities that can be realized to mitigate some migration problems and alleviate the time pressure to complete development projects will conclude the paper.

MIGRATION FACTORS

The factors that spur IAEA safeguards instrumentation development efforts can be subdivided into technical and non-technical categories, where non-technical issues comprise policy-related, institutional, and legal aspects. Both categories equally impact the need for new equipment and the user requirements that drive the new design. The overall goal of technical and institutional developments is to increase the effectiveness and efficiency of the safeguards regime in response to emerging global challenges. The following section will outline migration factors in greater detail.

Technical Factors

Instrumentation fielded by the IAEA must be designed according to application specific requirements, in order to withstand a variety of challenges, all of which are critical to the mission of providing reliable safeguards relevant information. First of all, the IAEA needs absolute assurance that the generated data are authentic and have not been tampered with. From a data security point of view, all instrumentation installed at safeguarded facilities has to be considered as installed in a hostile environment where potential proliferators could expend State resources to conceal prohibited nuclear weapons activities by defeating the system. To achieve this, all instrumentation is placed in secure tamper indicating enclosures and all data are authenticated.

For reasons of cost effectiveness a safeguards system, as far as possible, consists of commercial off-the-shelf (COTS) components complemented by custom-designed parts, in order to meet safeguards specific needs. The same complementary relationship is found in software products; executable programs must be adapted for safeguards specific applications or written as customized programs.

Safeguards instrumentation must be built to withstand a variety of harsh environmental conditions when installed on a global basis. The time for the inspector to retrieve safeguards relevant data must be minimized. IAEA inspectors must be familiar with a variety of safeguards systems, meaning that interaction with the equipment has to be easy and standardized.

Lastly, the lifecycle management of safeguards instrumentation is a particularly challenging issue, as identical pieces of equipment must be available over a long period of time, in order to allow replacement of defective units without delay. The highly specific application-related features force the IAEA to undergo extensive testing and vulnerability assessments of equipment, before it can be authorized for safeguards use. Any changes to a design after these assessments require resource-intensive repetition of the approval process.

Non-technical Factors

In order to provide the global non-proliferation community with sufficient assurance that treaty signatory states are in compliance with their commitments, international safeguards must be a very flexible verification tool. The IAEA responds to new challenges by applying both new policies and new instrumentation. Most recently, the IAEA implemented the Additional Protocol after the discovery of a covert nuclear weapons program in Iraq. The Additional Protocol expands the IAEA's rights and obligations to search for undeclared facilities and to draw state-level safeguards conclusions. Another example in development is the IAEA proposal to implement multi- or international fuel cycle models to limit the spread of sensitive technologies to already existing fuel provider states.

Any change in the IAEA's authority is immediately reflected in the tools safeguards inspectors can use to fulfill their duties. In performing Complementary Access in the frame of the Additional Protocol, inspectors may find themselves outside declared facilities and away from traditional equipment support. Portable instrumentation will help the inspector perform on-site measurements and to gather data. New inspection tools such as satellite imagery, nuclear forensics, wide-area monitoring, or environmental sampling are being applied. However, in addition to offering opportunities, new technologies require the IAEA to evaluate a broad spectrum of technologies. For the case of multi-national fuel cycle models, instrumentation needs might arise that promote the transparency of fuel providing facilities to provide sufficient assurance of fuel supply for recipient countries to sign up to such models.

The IAEA may be prompted by new facility types which differ from existing materials handling installations to review its suite of safeguards instruments. For example, traditional fuel measurement systems are likely to prove inappropriate when applied to a Pebble Bed Modular Reactor. The design of new instruments would be complicated by the lack of prototype facilities at the time of evaluation.

The Case of NGSS

The need to replace the DCM-14 based surveillance family arose mainly from technical factors. The system was fielded with good reliability in 1998, when the IAEA decided to select it as its first and exclusive digital image surveillance system with an obligation on the part of the manufacturer to assure a ten-years supply. While the DCM-14 is still the Agency’s workhorse for safeguards applications, new safeguards requirements and technological advances, e.g., in microelectronics, data processing and storage power, as well as in data communications, triggered the idea of a new, state-of-the-art system.

Parts obsolescence is another motivation to replace the DCM-14. Although development projects use lifetime requirements of instruments as guidelines, the electronics market eventually discontinues or replaces components. To cope with this problem, industrial or military standard components are used wherever possible. The consumer market is rarely an alternative, as components seldom meet IAEA environmental standards. In addition, consumer components typically quickly disappear from the market. The production of the PCMCIA card socket implemented in the DCM-14 was discontinued several years ago, as were several other components. The manufacturer, in order to meet the requested ten-year lifetime guarantee of the DCM-14, had to use last-time-buy opportunities. In order to have a new system available before the end of 2008 and taking a five years development period into account, it was necessary to initiate a new development in 2003.

USER REQUIREMENTS

After the need for new equipment has been identified, stands the question of exactly what functions the instrumentation is required to perform. Answering this can be very complicated, especially when several parties are interested in using the equipment. Funding availability might restrict the set of available solution choices and must be kept in mind when outlining the operability requirements of the equipment. This is further complicated, as safeguards should be as little as possible intrusive to plant operation.

Overall, the IAEA has a firm understanding of its instrumentation needs and how to best apply instrumentation to implement policy agreements on a technological level. Certain areas (such as environmental conditions and radiation exposure) are defined to the extent possible, other application scenarios fall into minutely defined categories. Defining user requirements for a development project of the scope of NGSS, however, proved to be a very complicated, cumbersome process.

Several parties within the IAEA provided input related to the user requirements. On the part of the Divisions of Operations (under which IAEA inspectors are grouped), input mainly consisted of request for improved usability of the instrumentation, stronger standardization of the surveillance instrumentation, additional operational features, and the request that the new user interface should be similar to the existing user interface, at least on the software side. Inspector input varied based on personal experiences with currently fielded instrumentation. In some cases, input varied widely over the same parameters. Some inspector requests were mutually exclusive. Other input came from the Divisions of Concepts and Planning, Information Technologies, and even from Procurement within the Department of Management.

The IAEA Division of Technical Support, (specifically the Section for Installed Systems, under which surveillance is located) faced the task of integrating user input with technical requirements into a single document. This included reviewing the requirements for the DCM-14 based surveillance system to see what was still applicable for NGSS and evaluating different monitoring scenarios where surveillance is used today and where it could potentially be used in the future, in order to find the smallest common denominator requirements. The Division of Technical Support also had to investigate the requirements necessary to integrate surveillance with other safeguards instrumentation (e.g., seals and non-destructive assay) and to plan for the manufacture and fielding of the finished instrument. This last task required particular care, as new safeguards instruments must be fielded with minimal impact on the inspection routine.

In order to analyze and assess all the input as objectively as possible, the IAEA tasked an outside consultant to gather, sort, and rate the varied opinions with regard to NGSS. The process was lengthy, but the use of a third party functioning as an impartial communication pool helped immensely in laying out the basic skeleton of the NGSS design.

Generating the user requirements offered opportunities as well as challenges to the IAEA. Migrating to a next generation system allowed the IAEA to make a clean cut with a legacy system and to explore how other market segments involved in treaty verification approach their mission. The IAEA was also able to explore the handling of product lifecycles within military and aerospace programs as well as other opportunities outside safeguards. The IAEA held an NGSS user requirements workshop, where experts from the international non-proliferation and safeguards community, military and space programs, R&D institutions, and the private industry gathered to broaden the view of what can be achieved in safeguards by using state-of-the-art technologies. In the end, the IAEA completed a comprehensive user requirements document outlining all design features needed for NGSS.

COOPERATION AND FUNDING

Having developed such an intimate understanding of its user requirements, it would appear, at first glance, that the IAEA would have been the best candidate to undertake the development of the NGSS. However, the IAEA has neither the personnel nor the resources required to accomplish such tasks. The IAEA must contract these projects to commercial entities and research and development institutions. Funding for such contracts is provided by MSSPs, some of them specifically designed to provide the IAEA with extra-budgetary funding for the development and procurement of safeguards instrumentation. The development of safeguards specific hardware and software is time-consuming and expensive. Possible development and manufacturing partners are also limited, as will be shown in the following sections.

Large Companies

Large commercial entities are appealing for both developing and manufacturing electronic instrumentation. In-house capabilities in research and development, tight quality control and documentation requirements, internal auditing, and financial security make large companies attractive partners for the IAEA. However, large companies focus on markets that are magnitudes larger than the safeguards market in production volumes. In-house processes are focused on automated large scale production, learning effects, and economies of scale.

While large companies might be interested in supporting the research and development of safeguards instruments, they are not interested in the small safeguards market, which has high production costs and quality standards as well as low order volumes which preclude automated production. On the other hand, large companies might overestimate the potential for safeguards equipment for other markets, enter into the development project, and then withdraw when the actual market potential becomes apparent. Such involvement can cause significant project delays and can drain both IAEA and MSSP resources.

Mid-sized Companies

Mid-sized companies are generally better partners for the development and manufacture of safeguards specific instruments. The total anticipated market volume is more appealing to them and negotiations and user requirements finalization can take place on a more intimate level. A few mid-sized companies have become reliable partners with the IAEA over the past decades, building the backbone of the supply of safeguards instrumentation.

Mid-sized companies also have their problems: they must focus on a certain field of expertise to remain competitive, meaning that a company might misunderstand the requirements of safeguards instrumentation and assume it would be a valuable addition to their portfolio. Once a company realizes that the desired product differs significantly from their area of expertise and has little applicability in their core capabilities, they might decide not to pursue the effort further. At this point, the IAEA might already have invested significant effort, in order to attract the company. Mid-sized companies supporting safeguards must be committed to consider safeguards as one of their core capabilities.

Small Companies

Small companies also seem to be preferable partners to the IAEA, particularly from a project management point of view. Small partners can concentrate on the IAEA and put all their resources towards safeguards developments. Small partners are also capable of building close relationships with their partners.

On the other hand, small companies depend greatly on their niche market. Previous experience is not necessarily transferable to other markets. Should the niche market change, a small company may be forced to withdraw from safeguards, since the market usually can only support one supplier per safeguards system. Small companies usually have few other customers and depend heavily on a continuous stream of orders to maintain operation. The IAEA usually orders equipment in the frame of Basic Supply Agreements (BSA), making it difficult for a small company to survive just on irregular IAEA purchase orders.

Research and Development Institutions

Government funded research and development (R&D) institutions also seem to be highly appealing IAEA partners, as they regularly expand knowledge and science to new levels. Such facilities have abundant resources and expertise, and few other entities grasp the IAEA's mission and requirements so fully. Nevertheless, the IAEA's cooperation with research laboratories has been overshadowed by difficulties.

While understanding technical issues is important, it is only part of the development process. Maintainability, operability, ruggedness, and cost-effectiveness in the parts used are other factors considered in designing safeguards solutions. While R&D facilities excel at providing scientifically sound solutions to a technical issue, they lack expertise in small scale production and in installing, operating, and maintaining equipment in the field. Technologies must transfer to industrial partners for commercialization after reaching prototype level at an R&D institution. This is the stage where problems arise.

Commercial partners have difficulties adapting prototype technology to a commercially acceptable product with a streamlined manufacturing process, a transition complicated by strict safeguards requirements. Operating with a lack of commercial pressure, R&D institutions tend to pursue the most scientifically sound solution, which may not be the one easiest to build, maintain, or support. Once a technology has been developed to the prototype stage, the scientists involved are usually forced to work on other projects and are unavailable to assist in commercialization.

Funding Parties

Several MSSPs provide extra-budgetary assistance to the IAEA. Other than the regular contributions all IAEA Member States pass to the Agency every year, these extra-budgetary resources are subject to the specific funding regulations of each individual MSSP. For NGSS, both the US and the German MSSPs have agreed to provide such assistance to the IAEA.

While such assistance is essential to relieve the IAEA's strained budget, there are a few issues impacting the IAEA's ability to use MSSP funds. Since part of the MSSP mission is to support the economy of its country of origin, many MSSPs are interested in having domestic institutions develop new safeguards instrumentation as well as in domestically manufacturing the systems for the IAEA. MSSPs promote domestic developers, secure funding for development and position them as manufacturers for later production.

PROCUREMENT

The United Nations (UN) procurement regulations also present a challenge to IAEA efforts. Development projects exceeding a certain expected cost volume must be issued as open tenders, inviting all into an open bid process. While this does attract comparable offers at competitive prices, this is not necessarily beneficial in safeguards. The unique experience, expertise, and reliability of specific companies and research institutions with safeguards and IAEA user requirements are essential to providing a project schedule and delivery plan at a fair price.

Under current UN procurement rules it is possible for a new company with a general interest in the safeguards market will underestimate the difficult process required to meet IAEA user requirements, bid low, and win a tender. There is a risk that the cheaper alternative will become expensive in the long term, as a newly awarded partner slowly acquires the experience which an established partner would have already calculated into its initial offer. In the worst case scenario, an established partner without a safeguards contract dissolves, while a new partner withdraws after failing in its efforts. Clear selection criteria, a detailed statement of work, and workable user requirements mitigate this problem.

THE NGSS PROJECT

The IAEA took the above issues into account by evaluating the received proposals, i.e., not only the price but also factors such as safeguards experience, company positioning, and established MSSP partnerships. Eleven companies responded with proposals to the IAEA's NGSS-related request for proposal. After an extensive review process, a consortium of two commercial partners, supported by two MSSPs, was selected by the IAEA. Both private entities have experience with the safeguards market, have partnered with each other before, and have a strong relationship with both MSSPs and the IAEA.

To allow the IAEA close control of the project and to allow easy MSSP funding, the NGSS contract with the developers was signed as a multi-phase contract. At the end of each phase, deliverables are presented and evaluated by the IAEA and the MSSPs. Only upon acceptance, the next phase can be initiated and funding requested. The IAEA can pass the whole project through its procurement process once and then order consecutive phases off that contract as with a BSA. For the funding bodies, funding does not have to be committed beyond the current project phase, and the whole project cost does not have to be assigned at the beginning of the project.

This contracting vehicle, however, has its drawbacks. After the completion of phase one, getting the funding in place quickly, so that the purchase order for the next phase could be issued, proved to be more complicated than anticipated, causing a delay, although the project itself was running according to schedule.

CONCLUSIONS

Technical and non-technical factors have strong influence on the migration of safeguards instrumentation that strives to increase the effectiveness and efficiency of international safeguards. New technologies provide new approaches to current IAEA safeguards and can strengthen the IAEA's treaty verification efforts. However, the IAEA must take into account its unique user requirements, its dependence on Member States' support, its limitations in partner selection, and potential funding complications.

Considering the lifecycle of the instrumentation as a whole is the greatest challenge. The best developer might not be the best manufacturer, and the best technical solution might not be the solution that is easiest to build and maintain. In order to address such issues, the IAEA has approached NGSS with a comprehensive lifecycle plan. NGSS is being developed under a joint effort based on user requirements that affect more than just development – they outline the lifecycle of the system.

The key to success is to draw upon all expertise and resources available in a comprehensive manner. NGSS has shown that the international safeguards community responds when the IAEA calls for assistance to provide expertise, recommendations, and funding, and it can serve as a role model for similar approaches in the future. Even though the process is not perfect and will need to be adapted for different projects, it has shown that relying on proven resources and expertise mitigates development pressure and allows a smooth instrumentation transition.

Activities of the ESARDA Working Group for Techniques and Standards for Non-Destructive Analysis

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Presented by A.-L. Weber

Abstract. ESARDA (European Safeguards Research and Development Association) is an association of European organizations involved in the R&D of Nuclear Safeguards mainly appointed to assist the Safeguard community with advanced progress in safeguards, enhancing the efficiency of systems and measures, as well as investigating how new techniques can be developed and implemented. Among the various activities of the association, Working Groups are established to promote and undertake collaborative R&D and information exchange activities in various fields.
This paper will describe the objectives, composition and activities of the Working Group for techniques and standards for Non-Destructive Analysis (ESARDA NDA-WG) and will give an overview of the major (ongoing or recently accomplished) projects. These projects can be categorised in four main technical fields: general function of NDA instrumentation, NDA techniques for Safeguards (γ spectrometry and neutron counting), NDA techniques for waste sentencing and modelling of NDA instruments.

1. Introduction

ESARDA is the European Safeguards Research and Development Association, gathering Safeguards Authorities, Operators of Nuclear Facilities, Research Centres and Universities to exchange information, co-ordinate and execute joint R&D programmes in the field of Nuclear Safeguards. Currently the members of the association are: AREVA (France), ATI (Austria), BNFL (UK), CEA (France), DTI (UK), EDF (France), ENEA (Italy), European Commission (representing the European Atomic Energy Community EURATOM), FZJ (Germany), HAEA (Hungary), IRSN (France), IKI (Hungary), SCK-CEN (Belgium), SKI (Sweden), STUK (Finland), UKAEA (UK), VATESI (Lithuania) and WKK (Germany). Representatives from other organisations that are not member parties (amongst them the IAEA, ABACC, and INMM) regularly take part in the activities of the association. Among the various activities of the association, Working Groups are established to promote and undertake collaborative R&D and information exchange activities in various technical fields.

The Working Group for techniques and standards for Non-Destructive Analysis (ESARDA NDA-WG) was established with the mission to provide the Safeguards Community with expert advice on Non Destructive Analysis methods, procedures, reference materials and on the performance of NDA methods, as stated in its Terms of

Reference. It acts as a forum for the exchange of information on Non Destructive Analysis methods, gathering national, regional and international control authorities, together with plant operators and R&D laboratories. This paper will describe the function, the composition and the activities of the NDA Working Group, by giving an overview of the major ongoing or recently completed projects.

2. Terms of reference

The NDA WG gives its support to the Safeguard community on Non Destructive Analysis methods by carrying out a series of tasks :

1. Maintain a list of NDA methods and instruments currently used or under development for accountancy and verification purposes.
2. Determine the reliability of NDA methods where possible with inter-comparison exercises of safeguards measurements.
3. Advise EURATOM and IAEA on the implementation of new and improved methods and advise on areas where R and D is needed.
4. Promote and coordinate R and D programs to fulfill safeguards and nuclear material management needs.
5. Promote the systematic and correct use of Reference Materials.
6. Assess and disseminate Performance Values for Uncertainties in NDA methods of nuclear material measurement.
7. Participate in the review, and promote the use of, International Target Values for uncertainties in measurements of nuclear material.
8. Consider sampling errors and sampling problems and their significance for NDA results.
9. Promote the use of internationally agreed definitions and terminology in the reporting of measurement results.
10. Assist in the development of new NDA methods in support of new safeguards requirements.
11. Promote cooperation with other working groups and the inspection authorities.
12. Collaborate with other working groups to develop comprehensive and integrated tools to fulfill new safeguards requirements.

The above tasks are undertaken by the NDA Working Group generally meeting twice per year in Europe: in spring around the ESARDA annual event and in autumn at the working place of one of the members.

3. Participants of the ESARDA NDA-WG

The NDA WG is composed of specialists from nuclear plant operators, NDA equipments suppliers, R&D laboratories and regulatory authorities. Some of the ESARDA members have representatives participating regularly to the WG activities: AREVA (through CANBERRA), BNFL (through BIL Solutions ltd), CEA, European Commission (JRC and TREN), IRSN, IKI, SCK-CEN, SKI and UKAEA. The group receives as well an active contribution also from observers (participants from non-member organisations): IAEA, ABACC and US national laboratories (Los Alamos and Lawrence Livermore).

4. Projects and achievements of the ESARDA NDA-WG

The WG activities undertaken these last years can be categorised according to four main technical fields:

- General functions of NDA instrumentation
- NDA techniques for safeguards (gamma spectrometry and neutron counting)
- NDA techniques for waste sentencing
- Modelling of NDA instruments (mainly Monte Carlo techniques)

4.1. NDA instrumentation

In the field of general NDA instrumentation, the NDA WG has been active for many years in the assessment and the updating of a comprehensive list of performances values for NDA techniques currently used for the assay of nuclear materials encountered in the fuel cycle for Safeguards purposes, according to the tasks 6 and 7 of the terms of reference. The successive evaluations are the result of specific international round robin exercises (carried out with specifically designed reference materials with the aim of assessing NDA performances), field measurements (field inspections activities) and tailored laboratory measurements, evaluated and discussed in the Esarda NDA WG.

The WG has participated to the redaction of the document on “International Target Values 2000 for Measurement Uncertainties in Safeguarding Nuclear Materials” [1]. This document lists for all the available analytical techniques (including DA and NDA) the desired measurement uncertainties that should be achieved in order to fulfil the Safeguards objectives.

Then the real performances of NDA instruments have been reviewed in the document “Performance Values for NDA techniques applied to safeguards” [2]. This document catalogues all the NDA techniques applied to Safeguards of nuclear materials, provides an extensive description of the principles, of the equipment and of the procedures and finally for each technique/application combination performs an accurate assessment of all the uncertainty components compiling tables of the real performances achievable with the NDA techniques.

Another performance value document for NDA techniques applied to waste sentencing is in progress. The document will contain a categorisation of the typical waste and waste packaging methods used in the nuclear plants, review the NDA techniques available for waste sentencing and finally compile the performance value with main focus on the fissile material determination. The performance values will include the detection limit and accuracy (total measurement uncertainty), to allow intercomparisons to be made, and the most appropriate measurement technique to be selected.

The use of unattended and remotely-operated instrumentation is becoming more and more used in nuclear safeguards in order to reduce the on-site inspection effort. Following a request of the IAEA and in collaboration with the Containment and Surveillance Working Group (ESARDA C/S-WG), a joint document on “Guidelines for developing Unattended Remote Monitoring and Measurement Systems” has been issued [3]. The scope of this document was provide a list of technical specifications and requirements that unattended equipment must (or in some cases simply should) fulfil in order to be acceptable for field deployment.

Finally a collection of NDA instrument characteristics is going to be compiled in a data-base, according to the 1st task of the terms of reference. As a first step all the NDA instruments applied for spent fuel have been catalogued together with their main technical characteristics. The data have been filled within an MSAccess database. Currently the structure of the database must be optimised in function of the interrogation needs and appropriate query tools must be developed. When this will be completed the database will be extended to all types of NDA instruments.

4.2 Gamma spectrometry

In the field of gamma spectrometry several inter-comparisons have been organised in order to assess the capabilities of this technique. The last of these was the Pu-2000 exercise dedicated to the determination of the plutonium isotopic composition. The main purpose was to test the performances of recent X and γ spectrometry methods developed for determining Pu isotopic composition over a wide range of abundances and to investigate possible sources of error. 20 plutonium-bearing reference samples have been prepared by the IRMM in Geel and measured by 8 laboratories using 18 different techniques (detector, acquisition chain and analysis software as MGA, MGA++ and FRAM). The results have been analysed and published [4].

The analysis work has led to a range of uncertainties that can be expected using those methods: from 3% to 20% for ^{238}Pu/Pu mass ratio when the ratio varies from 1.5% to 0.005%, 0.2 to 2% for the ^{239}Pu/Pu mass ratio (90 to 60%), about 4% for ^{240}Pu/Pu, about 3% for ^{241}Pu/Pu and about 3.5% for ^{241}Am/Am. As 10^4 results were to consider, only general conclusions have been drawn. A more elaborated statistical analysis of the leading parameters was then performed grouping data by plutonium type (low, medium and high burnup), revealing some bias for medium and high burnup plutonium and also some differences in the results obtained by different versions of the codes used. At that time, the NDA WG has decided to launch a complete performance evaluation of the plutonium isotopic codes, in cooperation with the developers, using the last versions of the codes, based on a set of ideal spectra firstly (taken in lab with good statistics on well-characterized reference samples), then on a selected number of spectra from the Pu2000 library. The measurement of real high burn-up plutonium, being more and more common in the European facilities, will be investigated by JRC/ITU, and the results will be reported within the NDA WG.

A compilation of uranium and plutonium spectra acquired during the Pu-2000 exercise and a previous one dedicated to uranium enrichment has been collected in the “ESARDA U/Pu Spectra Library” available on the web [5] for anyone who wants to use them to assess the performance of spectra analysis codes. The U and Pu reference spectra were provided by LNHB using two GeHp detectors (planar + coaxial) and a CZT detector. Other U ref spectra from LLNL were added with $^{235}U/U$ mass ratio from 0.018 to 99.1%, measured with a planar detector.

The benchmarking activity has recently found an important milestone in the organisation of the Workshop on "Gamma Evaluation Codes for Plutonium and Uranium Isotope Abundance Measurements by High-Resolution Gamma Spectrometry: Current Status and Future Challenges" held in Karlsruhe in November 2005. The workshop gathered 44 specialists from 12 countries including software developers, detector manufacturers, users from national laboratories and safeguards inspectorates (IAEA and Euratom). The current state-of-the-art of gamma spectrometry has been extensively evaluated and recommendations have been issued on harmonisation and version control, nuclear data standardisation, future requirement for unattended measurement and new materials from future fuel cycles, procedure optimisation. All the presentations and the minutes of the round table will be published in the proceedings of the Workshop [6].

4.3 Neutron counting and Monte Carlo modelling

A large interest is devoted to the application of Monte Carlo techniques to the numerical simulation of NDA instruments in general and neutron counters in particular, widely used in safeguards for the verification of fuel pins and assemblies, the measurement of the fissile content of scrap residues from reprocessing activities and the assay of individual fuel pellets for process control. The use of MC modelling is becoming increasingly widespread as a tool for reducing the reliance upon experiment for calibration of neutron coincidence counting systems.

Three benchmark exercises have been carried out in the last years in order to assess the capabilities of Monte Carlo to reproduce the experimental data:

- The first one dealt with the comparison of interpretational models used for the prediction of the real coincidence rates from a reference PWR fuel assembly measured with an active neutron collar when using the MCNP code [8] The participants used the same nuclear data for MCNP runs, with a fixed, predefined geometry model, but different interpretational models. The results of comparisons with experiment demonstrated that predictions could generally be made to an accuracy of 5 – 10%. However, due to uncertainties in the accuracy of the nuclear data constants used, it was difficult to evaluate the factors which determine the fundamental limitations to the level of absolute agreement that can be expected between measurement and calculations.
- The second one was then launched to analyse the influence of the main basic physical parameters (influence of fission spectrum, thermal treatment, cross section dataset, geometry model approximations) for Monte Carlo codes in common use on a simple case [7]: a point californium source placed at a fixed distance from a slab detector with interposed layers of moderator (polyethylene) and absorber (cadmium). The exercise has demonstrated that Monte Carlo modelling can be used to predict the Totals counting rate for a simple neutron counting geometry to a typical level of agreement of about 5% for typical lightly moderated geometries. The spread of results according to different nuclear data

and modelling styles is also of the order of a few % for lightly moderated geometries, but increases up to 10% for heavily moderated geometries. The experiments and comprehensive uncertainty analysis have provided a useful insight into the fundamental limitations to the level of agreement that can be accepted between measurement and calculation. The physics uncertainties concerning largely the physics and design of the 3He detectors lead to a minimum uncertainty of about 2% in the efficiency for this geometry. Furthermore, if one considers the typical uncertainties in the bulk density of polyethylene, an additional component of between 1 and 2.5% can be expected, increasing the overall uncertainty to up to 3%.

- The most recent one intended to model a passive multiplicity counter. This last benchmark was split in two parts. A full simulation of neutron generation, transport and detection, coupled with the simulation of electronics had the purpose to compare coincidence counting simulation tools, such as MCNPX and MCNP-PTA. A second phase was aimed to compare only the pulse train analysis models, also studying dead-time effects, and all the participants analysed the same set of pulse trains. An AWCC in fast mode was chosen, considering a set of 13 sources: random sources, pure spontaneous (^{252}Cf and Pu metal) sources, Pu oxide samples and mixed random source/PuO_2 samples. The results of this exercise have been described in a final report [9]. The exercise has confirmed that the point model works well for a typical well counter with reasonable samples. A special trick had to be used to make the point model work with the case of (α,n) neutrons having a significantly different detection efficiency from spontaneous fission neutrons. Both MCNPX and MCNP-PTA are reliable tools for the calculation of coincidence counter performance. The exercise also showed a consensus among the main labs concerning the method of calculating singles, doubles and triples from pulse trains. The results for the high counting rate and high (α,n) cases have large statistical uncertainties, preventing very precise conclusions being drawn about dead-time effects and correction methods.

  A follow-up of this benchmark is currently considered. The idea is to repeat the exercise with an experimental pulse train acquired in LIST mode, instead of a simulated one. The goal is to compare the available software for LIST mode data analysis in view of possible future developments of neutron counting towards the abolition of shift register analysers and direct acquisition and processing of pulse trains by a PC.

Under specific request of the IAEA, the NDA-WG is also redacting a “Good Practice Guide in the use of Numerical Simulation in NDA”. The objective is to set up a system of behaviour rules to be followed by anybody who is using computational modelling applied to NDA techniques, comprising both technical and non-technical considerations. Technical considerations will include the nuclear data used, the validity of the physics treatments and interpretational models, benchmarking the code under representative conditions, and the use of specific codes according to recognised procedures. Non-technical factors will include Quality Assurance, training and competency of the modelling practitioner. The guide will cover generic families of codes and application areas: Monte Carlo, gamma-attenuation modelling codes and burnup / depletion codes), but be limited to applications having a direct impact on an NDA system result, through calibration or data interpretation. The correct implementation of the good practises will constitute a sort of quality certification that will allow to help in the acceptance of modelling results in measurement techniques and evaluation procedures.

4.4 Waste sentencing

Apart from the above mentioned document on performance values, the working group has sponsored the preparation of reference standard for measurements of nuclear material in waste drums. 16 waste drums have been produced having different size (100 and 200 litre drums) and different matrix (homogeneous/heterogeneous with plastic, metal or mixtures). These drums are provided with insertion tubes where different reference sources can be located. A set of 37 pins containing plutonium can be arranged in order to load the drums with masses ranging from 15 mg up to 10 g and simulate concentrated and distributed sources. Waste drums and sources are currently stored at SCK in Mol.

The working group is currently organising an inter-comparison exercise of measurement techniques applied to the characterisation of plutonium in waste drums. The main goal of the proficiency test is the comparison of neutron counting systems especially passive neutron coincidence counting (determination of coincident pulse pairs) and multiplicity counting (determination of coincident pulse pairs and pulse triplets). The outcome of these systems is a ^{240}Pu equivalent mass, from which isotope mass can be computed if the isotopics are known. Hence the determination of the isotope vector (using gamma spectrometry) on the waste drums could be considered as well as part of the exercise.

5. Conclusions

The Esarda NDA WG will continue assessing the performances of NDA techniques in use for safeguard purposes as well as of new emerging methods, organising intercomparison exercises as round robin exercises where labs make comparative NDA measurements on a set of samples, or intercomparisons for data analysis codes.

The implementation of the additional protocol and the new Euratom approach could require developments such as newly designed instruments, integrating advanced technologies, more automated and remotely controlled. These aspects should be covered within the WG. A special attention need to be paid to spent fuel verifications. Regarding the detection of undeclared materials, a need has been expressed by the IAEA for developing rapid and sensitive screening techniques for environmental sampling. A joint meeting between both DA and NDA WGs was organised in May 2006 in order to initiate a common work.

The strengthening of policy towards the nuclear threat is requiring new concerns for the NDA WG in the area of fighting the illicit trafficking. The NDA WG is currently monitoring the international scenario and technical developments, as a few members are involved.

Future developments for the generation 4 reactors should also be of interest for the NDA WG, with the need to adapt current methods to the new fuel. Safeguards need to be developed with the design, and one of the members is involved in the generation 4 project.

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Secure data communication for safeguards implementation

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Abstract. Secure, reliable and cost-effective communication is necessary for effective IAEA safeguards implementation. The transmission of safeguards data from nuclear facilities to IAEA headquarters and regional offices (referred to as remote monitoring) initially demonstrated the need for secure, cost-effective data communication that satisfies the confidentiality requirements of Member States and provides data authenticity that allows the IAEA to draw soundly based, independent safeguards conclusions.

The implementation of integrated safeguards further demonstrates the need for information obtained in the course of an inspection, complementary access or design information verification, to be analyzed and corroborated with other information available at IAEA headquarters or regional offices while the inspector is still on-site. This requires the capability to connect the inspector, anywhere in the field, to the IAEA data network with secure voice, video, or data transfer capabilities. Such a data network would allow inspectors to 'push' information to headquarters, thereby providing IAEA managers with the ability to make evaluation in near real time, as well as the ability to 'pull' information from IAEA databases for evaluations in the field.

In order to expand the IAEA's present capabilities in the area of secure data communication, in 2005 the Safeguards Division of Technical Support (SGTS) initiated cooperation with the European Space Agency (ESA) on satellite communications. This cooperation included testing an ESA provided satellite terminal at IAEA headquarters and at a nuclear facility. Two feasibility studies were carried out to address end-to-end communication services based on existing orbital and ground-based infrastructures. The goal was to determine how the deployment of advanced satellite communication and services could meet future IAEA safeguards needs. The paper presents the results of these studies, including a summary of costs, benefits and possible implementation options. It also assesses how satellite communications associated with other headquarters' infrastructure can improve IAEA capabilities for safeguards implementation.

1. Introduction

The IAEA Department of Safeguards has been exploring remote transmission of data collected by surveillance and unattended monitoring systems installed at nuclear facilities since the late 1990s. Remote monitoring is the term used to define the transmission of these systems' state-of-health and safeguards data from nuclear facilities worldwide. Currently, there are approximately 120 remote systems linked to IAEA headquarters or the regional offices in Toronto and Tokyo. The IAEA employs whatever communication channels are the most economical and reliable for the given facility. This includes analogue telephone (PSTN), integrated services digital network (ISDN), broadband services such as data subscriber line (xDSL) and satellite. The Department downloads approximately 2 gigabytes of surveillance and sensor data per day over these various lines.

Communication channels are a critical component of remote monitoring, since their cost and reliability determine the effectiveness and efficiency of the monitoring system. Satellite communication was one of the first channels to be assessed when remote monitoring was initially considered for safeguards applications. Satellite communication is by its nature attractive because it can be implemented anywhere and is therefore independent of the communication infrastructure in a given country. Unfortunately, in the mid 1990s when satellite communication was first assessed in field tests, carried out in Switzerland, the cost of a satellite channel was very high and only became economical when other communication channels were not available.

Communication channels are chosen on a case-by-case basis, after analysis of the availability and cost of the communication infrastructure at the site where remote monitoring is to be implemented. This limited the installation of remote monitoring to facilities where the communications infrastructure was well developed. Remote monitoring has demonstrated a significant saving of inspection days over the last few years, and therefore a more standard strategy for efficient implementation is required. For example, there are planned connections to future facilities which are located in geographical areas where the communication's infrastructure is under development and terrestrial communication is not always possible. At these facilities, satellite communications would be used to fill the gaps in capacity where other communications channels are unsuitable for worldwide coverage.

In 2005, the Safeguards Division of Technical Support (SGTS) initiated cooperation with the European Space Agency (ESA) to analyse the future of satellite communication for remote monitoring and other safeguards applications. This cooperation led to two feasibility studies [1][2][3] that assessed front-to-back communication services that could meet near- and medium-term safeguards requirements. To facilitate the study, SGTS forecasted the number of sites that should be connected, along with the type of services and subsequent bandwidths that would be required over the next 5-10 years. The requirements specified the quantity of data to be transferred from field systems, including live video-streaming from surveillance cameras; identified expanded remote monitoring capabilities such as real time remote control like pan, tilt and zoom for surveillance cameras; and added secure voice and video communication between the inspectors and IAEA headquarters. Video conferencing places the greatest demand on bandwidth. Therefore, a major requirement for this proposed network is to support one video conferencing per geographical region in five regions (see below) during local business hours; the bandwidth required for two simultaneous video conferencing sessions is currently cost prohibitive.

Additionally, the IAEA also needs secure mobile satellite communications. Such a capability will play an important role in future verification activities by allowing the inspector to transfer data from the field, or to access data at headquarters securely from anywhere in the world. The IAEA and the ESA concurred that data security would be the IAEA's responsibility since a robust hardware-based virtual private network (VPN) system was already in use and its vulnerability had been assessed.

2. Feasibility studies

Following successful testing of satellite technology at IAEA headquarters and at a nuclear facility, the IAEA and the ESA decided to conduct feasibility studies for end-to-end assessments of the relevance of satellite communication for future safeguards needs. These studies were contracted to two companies so that the results could be compared. The firms were e2EServices Ltd [1][2] and Paradigm Services Ltd [3]; both firms are based in the United Kingdom. Safeguards requirements were initially provided in a document entitled "A Secure Global Communications Network for IAEA Safeguards and International Emergency Center (IEC) Applications". The studies included the IEC, which also requires a global satellite-based network.

This document also included two lists of facilities: (a) facilities where SGTS currently has data communications, and (b) facilities where it is projected to have communications within five years. The requirements were defined and discussed in a series of meetings between expert consultants, SGTS, and the ESA. Meetings with consultants were held separately to avoid the firms conferring with each other. The studies were completed and executive summaries were finalized and presented at an IAEA workshop held in Vienna on 18 September 2006.

The studies divided satellite coverage into five regions based on a list of current and proposed facilities. The regions are as follows:

(a) Europe, Middle East, North Africa (EMEA);

(b) North America (NA);

(c) North East Asia (NEA);

(d) South America and Africa (SAA); and

(e) Australasia and Far East (Pacific Rim: PRIM).

The facilities were classified into three categories as follows:

(a) Small SG sites (SGS-S) which correspond to typical light water reactor (LWR) sites generating about 10 Mbytes of monitoring data per day;

(b) Medium SG sites (SGS-M) which correspond to larger facilities with multi-camera surveillance systems generating about 200 Mbytes of monitoring data per day; and

(c) Large SG sites (SGS-L) which correspond to large and complex facilities with radiation and process monitoring systems generating several Gbytes of data per day.

Data were further categorized by the type of capability SGTS considered was required; this is summarized in Table 1.

Table 1.

ID	Application	SGS-S	SGS-M	SGS-L
1	Off line file transfer	yes	yes	yes
2	Live video	[future]	[future]	yes
3	Live sound	[future]	[future]	yes
4	Interactive data (Remote control)	yes	yes	yes
5	Interactive data (Internet access)	yes*	yes*	yes
6	Voice	yes*	yes*	yes
7	Video Conference	yes*	yes*	yes
8	Real time file transfer	yes*	yes*	yes
9	Remote e-mail	yes*	yes*	yes

* SGS-S and SGS-M have these requirements, but only when teams are at the sites.

A market survey identified two available satellite technologies that best met the IAEA requirements: for a fixed system, digital video broadcast return channel by satellite (DVB-RCS) was identified; and for a mobile system, broadband global area network (BGAN) INMARSAT was identified.

DVB-RCS has been developed based on recent advances in the consumer market for satellite television. It was originally intended to provide broadband Internet access to remote locations and to take advantage of the DVB television standard for transmitting a high data rate stream of Internet protocol (IP) packets to users equipped with small satellite terminals. DVR-RCS can be made to look like a terrestrial xDSL (digital subscriber line), one of the prominent technologies used by the IAEA for remote monitoring. DVB-RCS is an open standard (non-proprietary) with several vendors offering services. This is important if different vendors were selected for different geographical regions. DVB-RCS packages typically cost about $3000, including an antenna and the ability to share bandwidth. This is important as several systems in the same region could share a 256 kilobyte/sec service, which would allow for more cost-effective, full-scale deployment.

For mobile services, both studies recommended the BGAN which is a portable satellite receiver used in the maritime industry. BGAN has a laptop-sized antenna that can support up to 495 kilobytes/sec, but dedicated connectivity can be costly (over $20 per minute). When used with the SGTS standard VPN, BGAN would be ideal for inspectors who need secure communications with IAEA headquarters while deployed in locations where the local communications infrastructure is lacking.

The following service providers were contacted regarding technical assistance and cost estimates. All of the companies listed below responded favourably for future consideration: Datasat, Equant, Intelsat, New Skies, Satlynx (SES), Skylogic (Eutelsat), Stratos, Telenor, and Telespazio.

2.1. System performance

Figure 1 shows a proposed satellite communication implementation during a 24-hour period at a typical nuclear facility. Local business hours are reserved for the inspector to use secure voice, video or video conferencing sessions, and/or remote connection to the Safeguards LAN. The off-hours would be used to transfer remote monitoring data via scheduled jobs that would be automated.

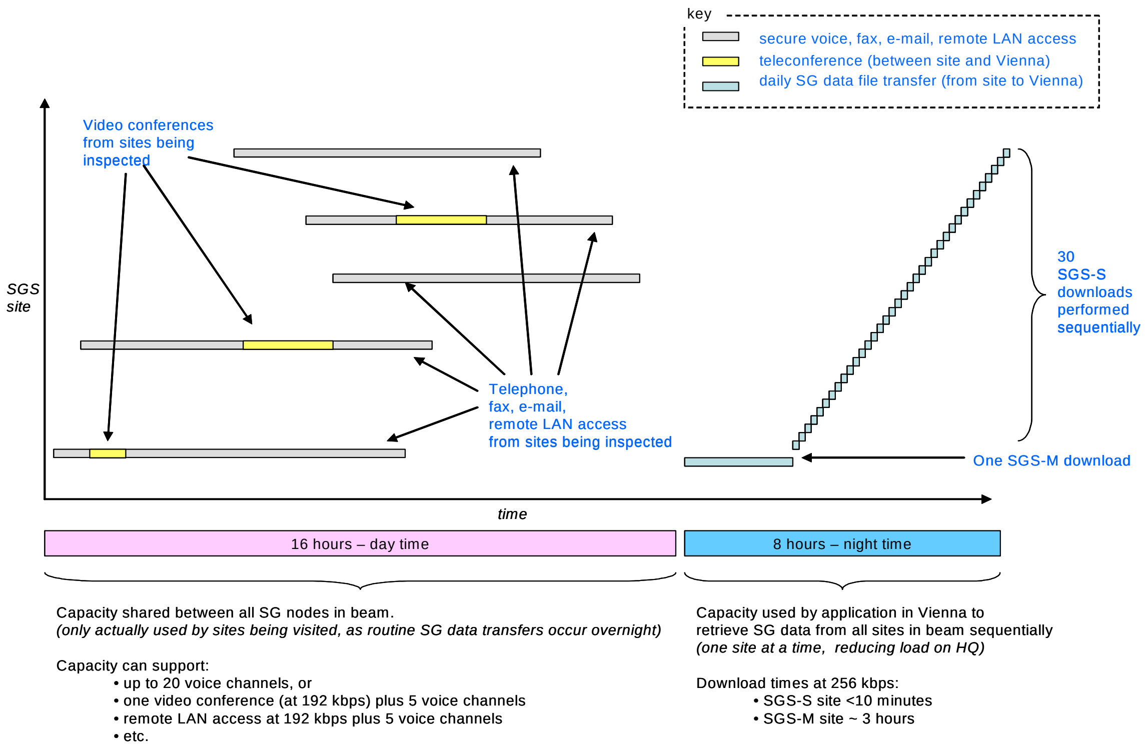

FIG. 1.

It should be noted that video conferencing sessions are user datagram protocol (UDP) based applications and are not affected by satellite latency such as the transmission control protocol (TCP) applications of the file transfer protocol (FTP).

2.2. Core network

The core network, depicted in Figure 2, consists of C & Ku-Band satellite systems in the five regions that are linked to several land-based hubs operated by a commercial provider. These hubs are connected to IAEA headquarters or regional offices by leased lines. The so-called ‘last mile’ is a dedicated connection with the Internet, providing a backup in case of failure. A phone line can also be used as a backup from headquarters to the sites.

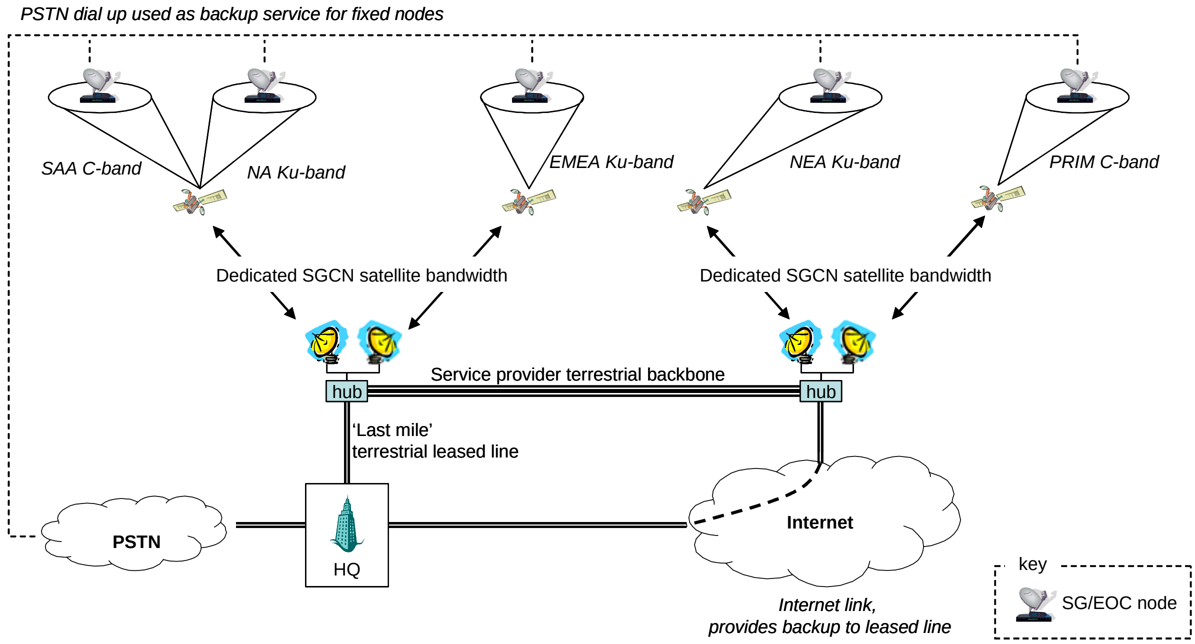

FIG. 2.

The required bandwidths for the regions over the next year or so are shown in Table 2. The IAEA anticipates a growth factor of three over the next five years. Even with video conferencing as a requirement, the total bandwidth in each region is relatively modest. However, experience shows that this network could experience throughput limitation during the daily transfer of remjote monitoring data under certain circumstances.

Table 2. End of First Year Deployment.

Region	EMEA	NA	NEA	PRIM	SAA
SGS-S	43	1	28	4	4
SGS-M	2	6	0	0	1
Bandwidth (kbps)	384	384	256	256	256

2.3. Deployment strategy

There are two fundamental methods for deploying a network of this size. The first is a ‘turn-key’ approach provided by a single prime contractor. In this case, the contractor provides a design, the components, performs the testing, and delivers the functioning system. Establishing a satellite network in this manner requires less human resources from the IAEA but can lead to a lack of understanding on how the systems works, leaving the IAEA dependant on the contractor. It would also be a more expensive option than the approach described below.

The second approach is for the IAEA to purchase individual components that meet its needs and to subcontract work to a number of companies. This allows more control in upgrading the network but the downside is that the IAEA takes responsibility for the entire network.

The studies recommended the second approach. This is because this approach would use standard DVB-RCS products that can easily be exchanged and thereby reduce the upgrade overhead. Further, it would be more cost effective as DVB-RCS services are widespread with more competitive pricing among the large group of vendors.

Once the providers have been selected and the services in the particular regions have been specified (e.g. the number of C & Ku- band systems needed) a bulk order for terminals (e.g. satellite dishes) can be made. One of the studies suggested that a specialized IAEA team could perform up to 80 satellite installations per year. The IAEA projects that a global IAEA network could be as large as 440 sites.

2.4. System costs

System costs for the connections of 440 sites were divided into 'up front capital costs' (CAPEX) and 'yearly operating expenses' (OPEX). Table 3 shows these costs broken down by sites and mobile usage. The CAPEX are clearly driven by the number of sites, and the OPEX are driven by the required bandwidth.

Table 3.

Item	CAPEX	OPEX/year
Monitored Sites	4,768 $k	3,601 $k
Inspection teams	84 $k	12 $k
Mobile Nodes	237$k	63 $k
Total	5,089 $k	3,676 $k

One of the studies explored this feature, proposing several options that would minimize the OPEX, as shown in Figure 3. The top bar in the graph identifies a fully featured network costing $3.6M annually, which is used as the 'reference system'. Option 1 shows the OPEX to be $2.4M per year without the video conferencing feature. Option 2 is of particular interest because it allows for video conferencing but in specified time slots of 8 hours per day that could be, for example, from 09:00 - 17:00 local time. In that way a video conferencing session could be conducted during normal local office hours. Thereafter, the bandwidth would be automatically reduced when only remote monitoring data transfers are taking place. This option reduces the OPEX to a more attractive figure of $1.8M per year.

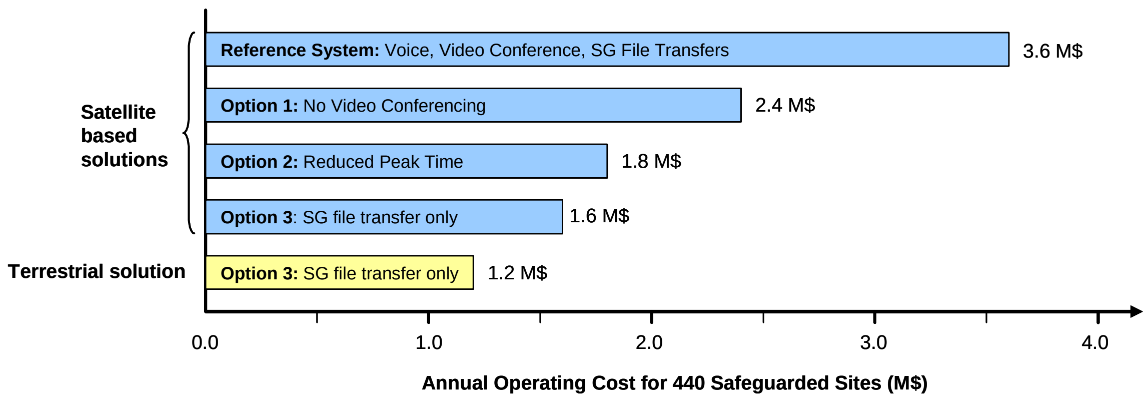

FIG. 3.

To put this in context, the IAEA is currently spending about $72K per year on terrestrial links for the transfer of remote monitoring data from 90 remote sites and obtains just state-of-health data from an additional 30 sites.

3. Study findings and recommendations

These studies produced the following findings and recommendations:

(a) A fully compliant satellite network can be built using components that are commercially available today.

(b) Sharing bandwidth among sites in a given region makes the satellite network more efficient and cost effective.

(c) A pilot project covering a single region should be deployed to gain experience with performance, traffic loading and reliability.

(d) For mobile solutions, BGAN terminals are recommended and should be purchased for evaluation.

In conclusion, whether the IAEA decides to fully implement a global satellite communication network or just supplement the existing terrestrial network, the equipment and recommendation in these studies are applicable.

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Design of safeguards systems for authentication

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Abstract. To permit valid conclusions to be drawn from safeguards data, it is essential that this data is known to be authentic. That is, it must be known that the data originated from the intended source, that the data was not changed in transit, and that it is not a repeat or delayed copy of previous data. Safeguards systems and equipment should be designed with authentication included, instead of attempting to add security later. Failure to integrate authentication measures early in the design results in systems that are expensive or possibly impossible to deploy securely. This paper gives some of the factors that should be considered and some of the methods that can be employed for ensuring high security while minimizing cost. The paper is a compilation of some of the lessons learned by the authors during many years of working with the authentication of these systems.

1. Introduction

A wide array of equipment is used in safeguards verification activities. In order to draw accurate, independent conclusions from the data from this equipment, the integrity of the equipment and its associated data must be ensured. This paper outlines processes and tools that can be used for providing a high level of assurance that all equipment and data gathered by that equipment used for safeguards purposes functions according to its design, cannot be tampered with without detection and that the equipment systems have no vulnerabilities that might be exploited by an adversary in an attempt to circumvent safeguards.

The term "authentication" can be confusing, since it takes on different meanings in different situations. It can refer to a cryptographic process in which a digital signature used to verify the authenticity of a piece of data is calculated and incorporated into the data set. It can also refer to the inspection process used to verify that a piece of equipment has not been altered and is therefore still "authentic". "Authentication" is also the term used to describe the process of verifying a person's identity when they attempt to gain access to a facility or a computer. In this paper, the term "authentication" will be used in a general sense, to refer to the entire process of verifying that a system is secure and can be trusted to provide data that accurately reflects conditions at the monitored facility without concerns that an adversary could manipulate the equipment or the data to produce false results.

Equipment security begins in the design phase and continues throughout the life cycle of the equipment, including decommissioning of security critical modules. It is difficult, and sometimes impossible, to add security to a system that has not been designed with securing the system in mind.

Vulnerabilities can also be introduced when approved equipment is assembled into a safeguards system, even if each part of the system is secure.

2. Assumed Adversary Characteristics

Before a secure system can be designed, the security threat that the system must address should be understood. The assumed primary adversary used for the purposes of the design of safeguards equipment and the vulnerability review/assessment is the plant operator and/or the host state. This threat agent includes the following capabilities and characteristics:

- Knows everything about the system except the passwords and secret or private cryptographic keys.
- Can likely obtain a copy of the equipment on which to develop and test tampering scenarios.
- Can draw on the computing capabilities of a national entity.
- Has experts with extensive knowledge of cryptography and system penetration techniques.
- May know operating system vulnerabilities that have not yet been made public and are not yet addressed in commercial security software.
- Can draw on extensive manufacturing capabilities – can produce exact counterfeits of enclosures, seals, and other equipment that might be damaged during a tampering attempt sometimes even when these items have been designed with counterfeit prevention in mind.
- Has complete physical access to all equipment outside the secure tamper indicating enclosures (STIEs), including all communications/signal cables.
- Has the ability to measure electromagnetic emissions from operating equipment to discover security related information, such as when a triggered measurement is being taken.
- Can alter the measurement configuration unless measures are taken to prevent this alteration, for example, radiation sources or shielding can be introduced to influence the nondestructive assay of a sample.
- Can produce complex and sophisticated radiation sources for use in tampering scenarios.
- Has extensive time, resources and access to the equipment's location to assess its operation under a range of scenarios over which the adversary has control.
- Will likely not pursue a denial of service attack against the equipment if the resulting loss of safeguards data will require a reverification of the material at the site. Other threat agents might pursue a denial of service attack, so this type of attack must be considered, but these agents are assumed to have much more limited capabilities, knowledge of the systems, and access to the equipment.

3. System Authentication Approach

System authentication can best be visualized by looking at a simplified system architecture drawing as shown in Figure 1. The monitoring system consists of a Data Store, a Communications and Power Management System ("collect computer") and attached data generators and sensors. The system design should include methods for detecting any attempt to alter the data, starting at the physical phenomenon being measured all the way through to the review station. This can be done through physical means, such as the use of sealed tamper indicating enclosures (STIEs), or through the use of cryptographic processes.

A variety of data generators and sensors may be connected to the collect computer, and the exact quantity and type will vary from one facility to another. Typical data generators and sensors include:

- Optical surveillance (video cameras, digital imaging),
- Seals (passive, active, remotely verifiable),
- Nuclear detection & measurement devices (γ, n), and
- Physical sensors (switches, devices for measuring physical properties).

The term "sensor" refers to the component which measures the quantity of interest. Raw information from the sensor is then processed by a "data generator" to produce an electronic representation which is meaningful to the balance of the monitoring system. In some cases the sensor is remote from the data generator, for example a radiation sensing element may be remote from the data generator. Thus although Figure 1 shows integrated sensors and data generators, in some cases these devices may be in two parts, with a communications link between them.

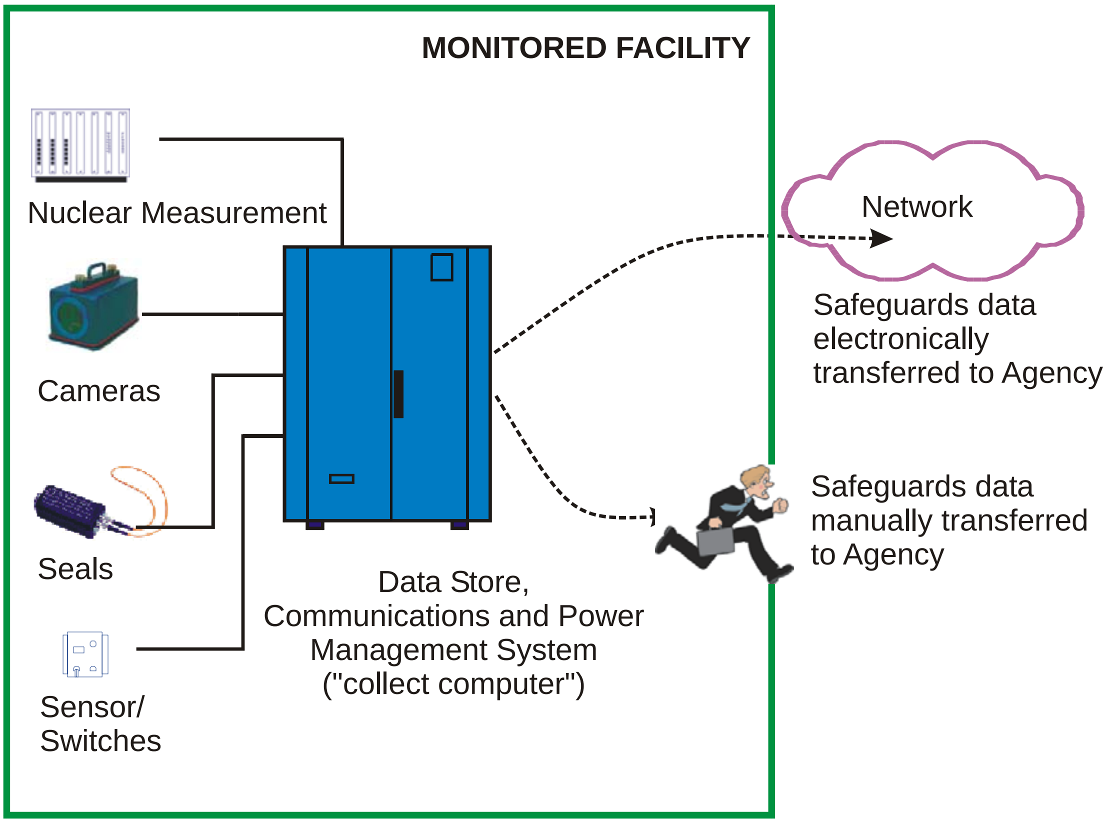

Figure 1: System Architecture

The collect computer, data generators and sensors, and the links between them, are installed within a physical protection boundary provided by the IAEA Member State operator of the facility. This boundary will include physical access controls which restrict access to the areas of the facility in which IAEA equipment is installed to persons authorized by the facility operator. This provides some protection to the equipment, but of course provides no protection against the facility operator.

Sensor and data generator data, and other data such as State-of-Health (SoH) data, is accumulated by the collect computer and forwarded to the IAEA in Vienna or to one of the regional offices of the

IAEA. For systems without a data network connection, the collect computer will accumulate one to several months of data and an IAEA inspector will visit the site and manually move the data using transportable electronic media (e.g. magnetic tape, disk).

There are many areas in the system architecture where an adversary might mount an attack. Each of these areas must be carefully considered, and a comprehensive plan for protecting the entire system must be developed.

The first area to be considered is between the physical phenomenon and the sensor. For example, attacks mounted in this area on video surveillance systems are called “before the lens tampering” and might consist of a static photograph or similar image placed before the camera. In radiation monitoring systems, this tampering might take the form of additional radiation sources or shielding being brought into the sensing area of the detector. Additional sensors, collimating shields, or other means must be designed into the system to address these threats.

The sensor itself is also vulnerable to tampering and must be protected inside a STIE. The data generator should be inside the same STIE, and the data generator should apply cryptographic authentication to the data if possible. If these devices cannot be inside the same STIE, the cable between them should be protected, probably through the use of tamper-indicating conduit. If this approach is used, the cost of inspecting the tamper-indicating conduit should be included when doing the cost-benefit analysis of the measurement or monitoring system.

Any device or software module that, if compromised, would allow an adversary to tamper with the equipment or data to alter safeguards conclusions with a low probability of detection is known as a security critical component. Examples of security critical components include most sensors, data generators, cables carrying data that are not protected by cryptographic means, and software used to verify cryptographic signatures. Security critical components are almost always subject to a vulnerabilty analysis (VA) conducted by experts external to the Agency.

If cryptographic data authentication cannot be applied to the data at the data generator, then the data collection computer must be treated as a security critical component, as well.

It is highly recommended that the number and complexity of the security critical components in the system be minimized, since special procedures and considerations apply to these components which make them considerably more expensive and more difficult to maintain. The designs of all the security critical components should be frozen after the VA has been performed. The VA process will require the source code for all security critical software modules, and those modules will also almost certainly be frozen after the VA. No changes to these hardware or software modules should be allowed without a further VA. Therefore, it can be very difficult to make changes to these security critical components during their operational life. Minimizing the complexity of the modules also minimizes the probability that changes will be needed. It is also recommended that security critical modules be designed to be completely independent from other operational modules that are likely to be changed to accommodate user requests.

It is almost impossible to secure a general purpose computer with a complex operating system.Virus scanners and other anti-malware tools are not effective against high-level threats, since these tools generally only detect threats that have already been identified. A high-level adversary will use tools that are not addressed by the anti-malware tools. In fact, if the adversary knows what tools will be used to detect his attacks, he can design his attack tools to avoid them.

Unfortunately, general purpose computers are almost impossible to avoid in a modern monitoring system. There are still ways to work around the insecurity of the computer. The best method is to move the security critical tasks to other devices, by using cryptographic authentication in the data generator, for example. When the data includes a cryptographic signature, the adversary cannot modify it without detection, so the computer is no longer a security critical device.

In some cases it is possible to adequately isolate a computer processing unsigned data from the attacker through the use of a firewall, virtual private networks, and other similar devices. This isolation can be effective because of the nature of the attack against these computers. The adversary must not only program the computer to generate false data, he must find a way to control the computer to provide accurate results during calibration and testing with known sources but to give altered results when processing real data. The trigger to switch between accurate and falsified results is often referred to as a “hidden switch”. There are many ways to implement such switches, and a discussion of this topic is beyond the scope (and classification level) of this paper. If it is possible to adequately cut off the adversary’s ability to trigger his hidden switch, the computer can be considered secure.

Some hardware has been certified as meeting certain security standards, either under the US National Institute of Standards and Technology’s Federal Information Processing Standards (FIPS) or under the standards of the international Common Criteria. The use of security hardware certified under Agency accepted standards, such as firewall and virtual private network units, is recommended. These units might be employed without requiring a VA, and the security critical functions that they provide will be segregated from more complex equipment.

The system design should also include a security plan that includes a list of the security critical components and an inspection plan for maintaining system security. The system is also often subjected to a VA, and the security plan should be evaluated as part of that analyis. Any changes to the system or the operating procedures of the system that impact the security of the system will require an additional VA.

Once the authentication approach for the system has been defined, a set of functional authentication requirements for the various system devices can be developed.

4. Device Authentication

Any device that is a security critical component of a system must be designed with authentication in mind. It must be possible to establish trust in the device’s design through the VA process. After manufacture, it must be possible to verify that the device matches the design that was assessed. After the device has been fielded, it must be possible to verify that it has not been tampered with if there is ever a possibility of unathorized access to the equipment.

4.1.Design Phase

Security critical components should be designed to be simple, modular, inspectable, and verifiable. Maintaining these four characteristics will minimize the cost of the VA while increasing security, and the devices will also be less expensive and easier to maintain in the field. Each of these characteristics will be explained briefly.

Simple – The device should not have any extraneous functionality and should be easy for a VA expert to understand completely.

Modular – It is much easier to establish and to verify the security of a piece of equipment or a system if it is composed of modules with well defined boundaries. This allows the VA team to verify the authenticity of individual modules instead of trying to assess a complex system all at once (although an assessment of the system as a whole cannot be fully avoided).

Inspectable – In order to establish and maintain trust in the authenticity of the device, inspectors or technicians must be able to inspect the device and verify that it exactly matches the device that underwent the VA. For example, certain key components such as the read only memory (ROM) that contains the device’s firmware must have the same memory size and must contain the same firmware as the original component. During the VA, the source code for the firmware will have been inspected for exploitable security flaws. Then the firmware will have been verified to be a properly compiled result of the assessed source code. Provisions must be made to allow the inspector or technician to verify that the firmware is an exact replica of the firmware examined in the VA. Systems should be designed with inspection in mind because it is very difficult if not impossible to adequately inspect a system that was designed without considering these factors.

Verifiable – There should be methods to easily verify that the device has not been altered without performing a detailed internal inspection. This verification might take the form of monitoring or inspecting a tamper-evident coating, security tags on critical components, or possibly the verification of the integrity of the firmware from outside the device. These verification methods are very difficult to add to a device after production has started.

The device will also be more secure if vulnerability analysis experts are allowed to review the design as it is developed. This will result in a more secure device and can also result in considerable cost savings by avoiding major changes to the equipment late in the design process. Some equipment is not subjected to its first VA until after manufacturing has started, which can lead to expensive reworking of the manufactured devices.

4.2.Manufacture and Receipt of Equipment

In most cases, it is reasonable to trust the integrity of the manufacturer of the equipment. However, the equipment should be inspected upon receipt to verify that the devices exactly match the device that was subjected to the VA. In most cases, this can be adequately addressed by verifying part numbers on the critical integrated circuits and verifying that the firmware matches that generated during the VA. For products under high volume production, a “random” selection from the production line may provide increased assurance.

In some cases, however, more extensive measures may be necessary, such as microscopic inspection of the internal structure of critical integrated circuits. Several years ago, a team of experts in the United States, the Authentication Task Force (ATF), studied methods for assuring the authenticity of equipment in extremely critical circumstances. A discussion of these procedures is beyone the scope of this paper, but interested readers should consult the ATF’s report[1].

4.3.Maintenance of Security and Subsequent Verification

All security critical components should be kept under seal at all times that a potential adversary might have physical access to the equipment. The seals and their associated STIEs can provide evidence of any attempt to tamper with the equipment.

However, the seals and STIEs cannot adequately protect the equipment throughout its life cycle. Eventually, a customs inspector will cut a seal to inspect the shipment. A security critical component will be discovered to have been left unattended while the inspector or technician left the room. The seal on an enclosure might have been accidentally damaged, which might bring its integrity into question. It is unreasonable to dispose of the equipment in these situations, so a plan for verifying that the device has not been tampered with must be established.

There are many possible methods for reestablishing trust in these situations. First, the integrity of the firmware should be verified. In the case of computers, the BIOS can be verified and a new hard disk installed with a trusted version of the software. Microscopic examination of solder joints on critical components can verify that the device has not been replaced. If conterfeit resistant tags have been applied to critical ICs, the integrity and authenticity of the tags can be verified. The methods used will depend on the type of equipment and the possible attack scenarios against that equipment.

4.4.Decommissioning of Security Critical Components

In some cases, security critical components may include sensitive information after they are taken out of service. An adversary might be able to use the component itself as a counterfeit for a unit that is still in service. A secure method for protecting the sensitive information and/or securely disposing of sensitive equipment must be defined in the security plan.

5. Supporting Technologies

5.1.Cryptographic Data Authentication

Data (or software or firmware that must be verified) can be protected against undetected alteration by adding either a message authentication code (MAC) or a digital signature to it. An attempt to modify even a single bit in the data will be immediately obvious when the MAC or the digital signature is verified.

The preferred approach is to use a digital signature, which is based on a public key algorithm. The preferred algorithms are RSA[2], DSA, and ECDSA[3]. In public key systems, a different key is used to verify the signature from that which was used to generate the signature. The private key used to generate the signature can be stored securely in the data generator without any need for it to exist in any other equipment or for any person to have access to the key. The public key used to verify the signature can be known by everyone without compromising the security of the system. It is only necessary to prevent adversaries from substituting their own public key for the real key. This is relatively easy to do, and public key systems are much easier to implement securely than symmetric key systems.

In some cases, however, the data generator does not have the computing resources to implement one of the public key algorithms, or the data architecture will not support the size of signature required. In these situations it is necessary to use the symmetric key based MAC, preferably using the keyed hash algorithm (HMAC[4]) using the secure hash algorithm (SHA[5]). In the MAC systems, the same key is used to generate and verify the MAC. This means that anyone who can gain access to the key needed to verify the MAC can also generate bogus data that will pass the verification process. Thus these systems require that the highly sensitive keys be stored in many locations and may need to be carried by inspectors and technicians to remote facilities. It is much more difficult to implement these systems securely than the corresponding public key systems.

5.2.Tamper Indicating Enclosures

Tamper indicating enclosures can range from the size of a button to an entire room. The primary requirement for these STIES is that any attempt to tamper with the equipment inside will leave easily detected evidence on the surface of the enclosure. The preferred material for these enclosures is aluminum that has been anodized in a light color, preferably light blue or gold. However, anodized aluminum is not practical for large enclosures, so other materials and coatings are often used.

The STIE will almost always be subjected to a VA, often in conjunction with the VA of the device that it protects.

Any time that a STIE is used, a plan should be implemented for periodic inspections of the integrity of the enclosure, including an inspection of the inside surfaces, since it is much harder for an adversary to repair damage to the inside of the sealed enclosure.

5.3.Tamper Indicating Devices

Although there are a wide assortment of seals available on the market, only the very few that have been approved by the Agency provide the necessary security to protect these measurement and monitoring systems.

Reflective Particle Tagging (RPT)[6] might be useful for identifying critical ICs and other components used in these systems. These tags were developed by the US government for treaty verification purposes. No practical methods have been found to transfer one of these tags from one component to another or to counterfeit a tag. The tags are easily to make and apply in the field, and the reader equipment can be made to be easily portable. Unfortunately, there are currently no commercially available RPT readers.

Conformal coatings on circuit boards will make the process of replacing individual components on the boards. These coatings can be especially effective for this purpose if taggants or other unique identifiers are applied to the board in such a way that the coating cannot be removed without destroying the identifier.

5.4.Inspection Aids

Microscopic inspection of the circuit boards can be very effective in detecting the substitution of components. There are small hand-held microscopes available for this purpose. Digital cameras mounted on microscopes or with macro lenses are also useful for these inspection activities.

6. Conclusions

It is possible to ensure the integrity and authenticity of data and equipment against very high threat levels, but in order to do so, the equipment and the measurement or monitoring system must have been designed with the necessary security measures in place.

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Verification of irradiated uranium targets by NDA and DA techniques

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Abstract. Irradiation of uranium (U) targets in a research reactor can be one of the declared activities in a facility under safeguards. The IAEA performs verification of such targets using non-destructive assay (NDA) and destructive analysis (DA) techniques. This paper describes on-site inspection activities and evaluation of the measurement results. High resolution gamma spectrometry (HRGS) can be applied for the measurement of the fission products and U enrichment (for low irradiated targets). The model used for the data interpretation is presented. The irradiation and cooling times are estimated using an irradiation code. The DA data are used for independent verification of the cooling time and correction of the irradiation code parameters.

1. Introduction

One of the declared activities carried out in research reactors can be neutron irradiation of various U targets, such as encapsulated UO_2 in pellet or powdered forms. The U mass in one target can vary from several grams to several hundred grams. Irradiation of larger U masses in one target would be inefficient because of the neutron flux degradation inside the target. The irradiation time can vary from several hours to one to two years. The irradiated targets discharged from the reactor should be cooled for a certain period of time before any other measurement or chemical separation is performed. The cooling time can vary from half a year to several years or even longer. The State's declaration normally provides information on the initial U enrichment, the quantity of the irradiated U and the irradiation history of the targets.

The objectives of the verification activity are:

— To confirm that targets were irradiated for the declared period of time;
— To confirm the irradiation parameters -- namely, cooling and irradiation time;
— To confirm that the declared material was irradiated; and
— To confirm that the declared quantity of material was irradiated.

Some verification activities (mainly NDA measurements) can be performed directly at the site, whereas other activities require taking samples which should be analysed by destructive methods in a laboratory.

2. On-site activity

During the on-site inspection, selected targets should be counted, weighed and measured by HRGS. Counting of the targets could be accompanied by measurement of the gamma dose rates. One can expect dose rates of up to several tens of mSv/h at 10 cm and up to 1 mSv/h at 1 m distances from the target.

Weighing of the targets can be combined with measurement of their dimensions and photographing. The balance used should be shielded with lead to reduce the radiation dose for personnel.

The data collected during item counting, measurements of the contact activity and weighing allows grouping of the targets into several strata. The strata contain targets with similar physical (cladding, dimensions, weight, etc.) and irradiation (gamma dose rates) parameters.

The selected targets from every stratum should be measured by HRGS. A large-size planar HPGe detector can be used. This detector has high-energy resolution and efficiency allowing the simultaneous measurement of the 100 keV energy region in connection with the MGAU code and the 500 keV – 1500 keV energy region where most of the fission products have gamma lines. The detector head is shielded with a lead collimator. The measured target should be far enough from the detector to avoid high-count rates, which could overload the detector and spectrometric electronics. A Cd filter in front of the detector head can help also to reduce the high-count rate. Additional lead shields should be installed to avoid detection of radiation from sources other than the measured item. Regular background measurements help in controlling sources of radiation in the measurement area.

Based on the HRGS results (for instance, the $^{134}Cs/^{137}Cs$ ratio is proportional to the irradiation time) some targets should be taken for DA. The parameters important for transportation of the targets are gamma dose rates and quantity of uranium in the targets.

3. Activities of radionuclides in the targets

The objective of the gamma spectrometric measurements is to determine the activity levels of gamma emitting isotopes in the targets. Table I lists typical isotopes that could be found in the measured gamma spectra. ^{60}Co is a product of neutron activation of ^{59}Co, which is a component of the stainless steel and an impurity in Al alloys. Both stainless steel and Al can be used as a cladding material in the targets. The U K X-ray lines could be present in the spectra. They are excited mainly by ^{137}Cs gamma rays.

Table 1. Gamma emitting isotopes in the targets.

Isotope	Half-life, y	Key γ line, keV	Comment
^{137}Cs	30.04	661.6	Fission Product (FP)
^{60}Co	5.27	1173.2	Activation product
^{134}Cs	2.06	795.8	Product of neutron activation of ^{133}Cs(FP)
^{106}Ru	1.023	621.9	FP, generated mainly in ^{239}Pu fissions
^{125}Sb	2.758	427.9	FP
^{144}Ce	0.780	696.5	FP
^{154}Eu	8.593	1274.4	Product of neutron activation of ^{153}Eu(FP)
^{155}Eu	4.761	105.3	FP
^{235}U	7.038e8	185.7	
^{234}Th		63.3	
^{234m}Pa		1001.0	

The Genie 2000 software [1] can be used for identification of the nuclides in the spectra by their gamma lines, determination of the absolute efficiency of the gamma spectrometer and calculation of the absolute activities of the radionuclides. A nuclide library and the absolute efficiency curves are necessary to perform the necessary calculations. The library should include the isotopes mentioned above. It is created by the Nuclear Editor module.

The ISOCS (in-situ object counting system) software package can be applied for the evaluation of the absolute efficiency of the gamma spectrometer. It can be used if the detector was characterized at the Canberra Industries factory. The ISOCS package calculates the absolute efficiency of the HPGe detector for the set of typical measurement geometries and sample types. Three types of information must be specified for the ISOCS calculation:

— Detector characterisation data and collimator design;
— Geometrical arrangement (distance detector – sample, sample elevation, absorbers between detector and sample); and
— Sample parameters (shape, size, materials used).

All these data are introduced in the template selected for the evaluation. It is assumed that the template 'simple cylinder' has been used. A cylindrical-shaped container filled with radioactive material describes the target in the template. The parameters of the cylinder are: material and thickness of the container, its internal diameter and height, and material and height of the radioactive source inside the container. The operator can provide this data.

The declared UO_2 density can be verified either by DA or by the evaluation of the HRGS measurement results with Genie 2000 module LACE (line activity consistency evaluation). The spectrum of a low irradiated target can be used. It has gamma lines of ^{234}Th and ^{234m}Pa in the energy range from 63.3 keV to 1001 keV. These isotopes are in secular equilibrium and their activities are equal to the ^{238}U activity. The activity calculated with the low-energy gamma lines will be equal to the activity calculated with the high-energy gamma lines if the density of the UO_2 used in calculating the absorption of gamma rays with the ISOCS is correct. Figure 1 shows three graphs of the LACE analysis results for three values of UO_2 densities: 4 g/cm^3, 2 g/cm^3 and 8 g/cm^3. The gamma spectrum of the test UO_2 sample was used in calculations. One can see that a UO_2 density of 4 g/cm^3 gives no positive (underestimated density) or negative (overestimated density) trends in the energy interval from 63.3 keV to 1001 keV.

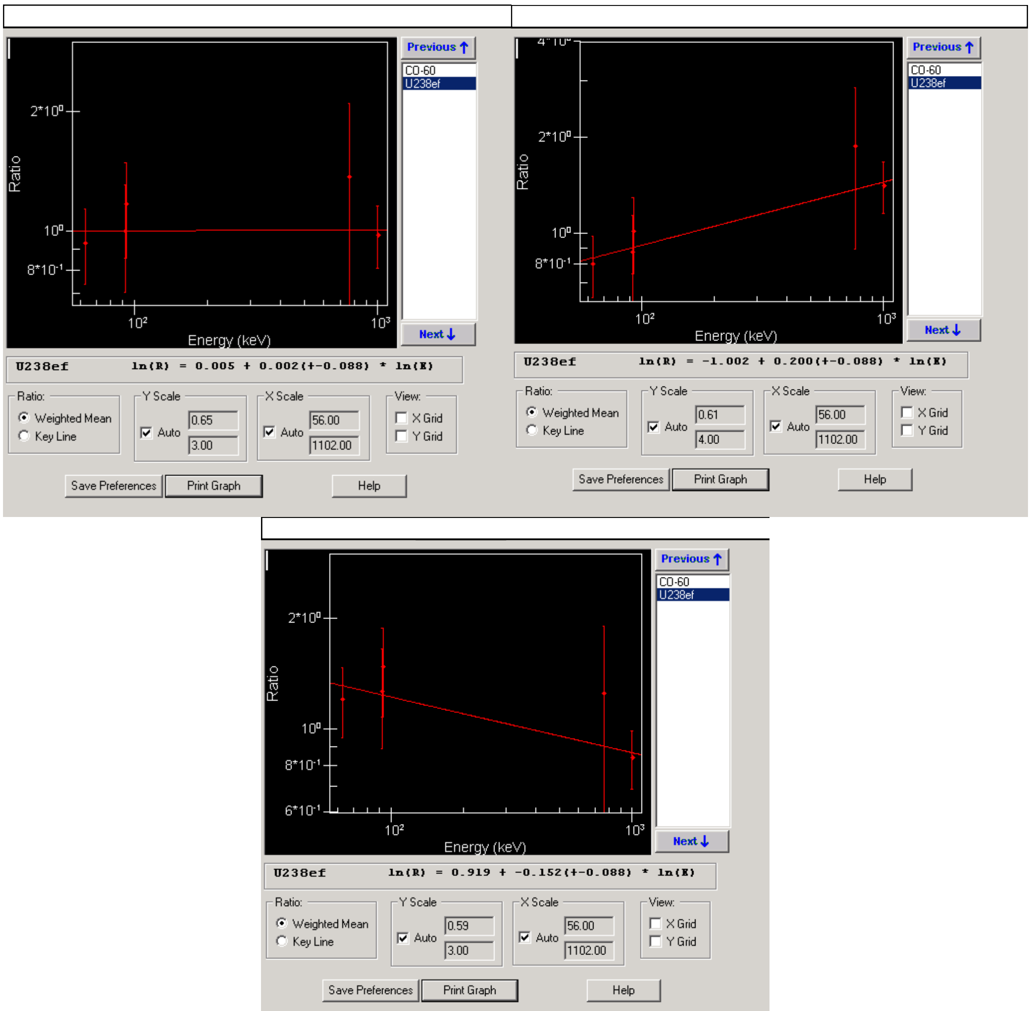

FIG. 1. *LACE analysis of the UO_2 density in the low irradiated U target.*

4. Uranium weight

The quantity of irradiated U is estimated using weighing data and the declared tare weights. The tare weights can be verified using drawings of the targets supplied by the operator. The targets taken for DA can be analysed by neutron radiography to confirm operator data on the target's design. The tare weight can be verified directly after removing the UO_2 from the cladding in a hot cell.

5. Irradiation and cooling time

The measured activities of the fission products and U content in the samples can be used for the evaluation of the irradiation and cooling time. The cooling time is the time interval between the end of irradiation and the measurement date. The irradiation history can be described as one irradiation period without interruption and with a constant neutron flux. The irradiation code Nuclear Analysis 2.0 (NA) can be used to estimate expected radionuclide concentrations [2].

The irradiation and cooling time is evaluated by fitting of the fission product activities or activity ratios calculated with the NA code to the measured values. The NA code models the irradiation process using a two-group approximation for the neutron flux (i.e. thermal and epithermal). The two-group cross sections of more than 2000 isotopes are included in the calculations. Shielding effects in the resonance energy region are not taken into account. The input data of the NA is initial inventory and irradiation parameters. The initial inventory is initial U isotopic composition. The irradiation parameters are the coolant temperature, cadmium ratio (neutron spectrum hardness) and neutron flux. The data depend on the design of the research reactor and can be taken from open sources and the IAEA's design information questionnaire (DIQ).

The printout of the NA calculation is a table of the activities of the selected isotopes (fission products measured by gamma spectrometry) versus irradiation.

The fitting of the calculated activities to the measured values is performed by the least square method. Figure 2 shows the flow chart of the fitting procedure. The minimised parameter is CHISQUARE defined by the equation:

$$\text{CHISQUARE} = \text{SQRT}\left(\left(\Sigma\, w_i\, [A_i(T_{irr}, T_{cool}) - A_i\,(\text{meas})]^2\right)/N\right),$$

where

$A_i(T_{irr}, T_{cool})$ and A_i (meas) are calculated and measured activities of the isotope i,

N is the number of the fitted isotope activities,

$w_i = 1/[\Delta A_i\,(\text{meas})]^2$ and

ΔA_i (meas) is the uncertainty of the activity of the isotope i reported by Genie-2000.

The sum is taken over the measured isotopes. The weighted deviation is

$[A_i(T_{irr}, T_{cool}) - A_i\,(\text{meas})]/\Delta A_i\,(\text{meas})$.

Two types of calculations can be performed:

— Type I. Minimisation of the absolute activities of the FP (^{137}Cs, ^{134}Cs, ^{106}Ru, ^{125}Sb and ^{154}Eu); and
— Type II. Minimisation of the activity ratios $^{134}Cs/^{137}Cs$, $^{106}Ru/^{137}Cs$, $^{125}Sb/^{137}Cs$ and $^{154}Eu/^{137}Cs$.

The flow diagram for the type II minimisation is similar to the flow diagram shown in Figure 2 except that this type of minimisation does not require the weight of U in the target. The result of this fitting is less dependent on the operator data. Another advantage is that irregular samples such as boxes filled with several targets can be evaluated by this fitting. The disadvantage of the method is that the number of experimental points is less and the method cannot be applied if only the ^{137}Cs activity is measured.

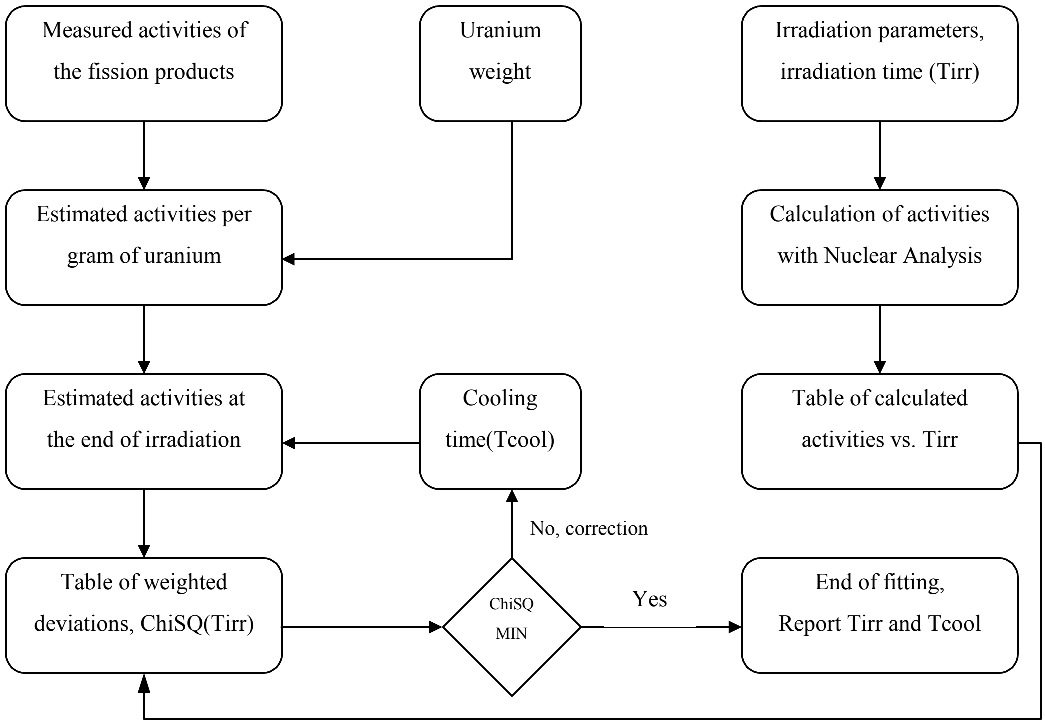

FIG. 2. Flow diagram of the fitting irradiation and cooling time.

Figure 3 shows the weighted deviations of the measured isotope activities versus irradiation and cooling time for the U target irradiated in a TRIGA reactor (model case). The neutron flux 3 x 10^{13} $n/cm^2/s$ was chosen for the calculations. The CHISQUARE is also shown on the figures.

The point where deviation of the given isotope activity (activity ratio) crosses the “0” horizontal line corresponds to the best fitting of the irradiation (cooling) time for this isotope. The scattering of the irradiation and cooling time for the individual isotopes from the value taken at the minimum CHISQUARE indicates uncertainty of the estimated values. The difference between maximum and minimum estimates is 1 year (10%) and 25 days (25%) for the cooling and irradiation time.

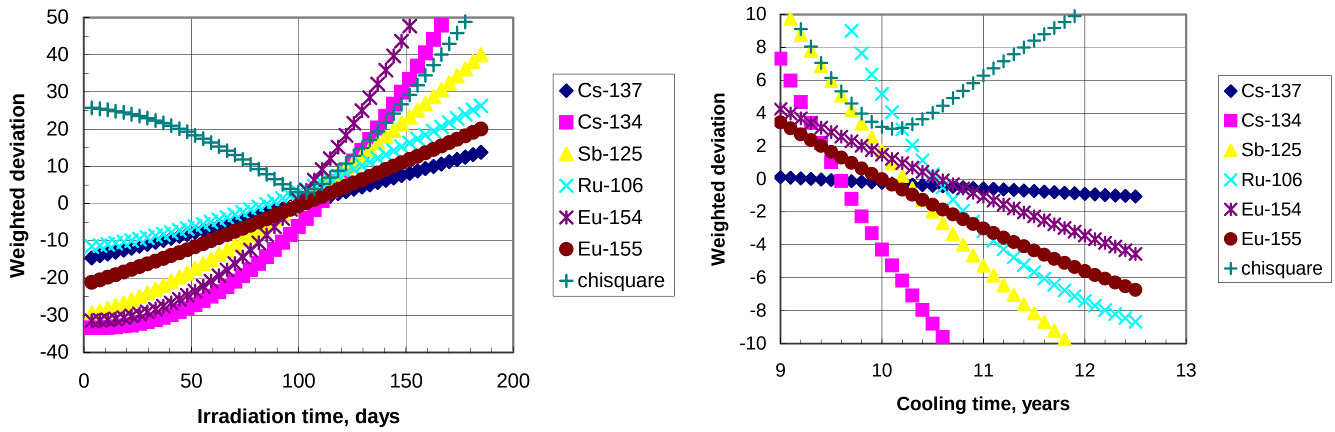

FIG. 3. Weighted deviations of the calculated activities from the measured activities vs. T_{irr} and T_{cool}..

Figure 4 shows the CHISQUARE as functions of both irradiation time (X axis) and cooling time (Y axis). The ellipsoid in the centre of the Figure shows an area where cooling time (9.8-11 years) and irradiation time (80-120 days) provide minimum deviations of the calculated activities of the fission products from the measured values.

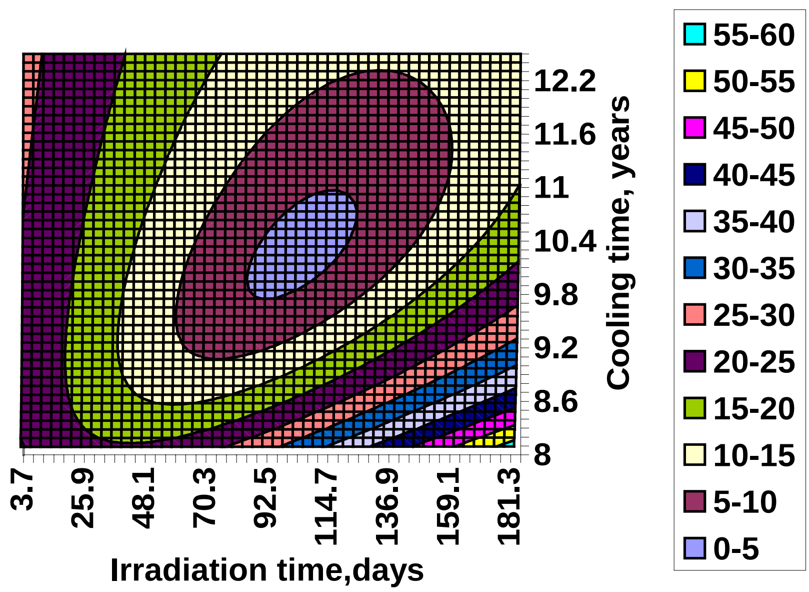

FIG. 4. The CHISQUARE profile.

Some uncertainties in the irradiation and cooling time estimation may arise because of the simplified irradiation process modelling in the NA code. The DA data allow correction of the neutron cross-sections in the NA code and improve the accuracy of the estimations based on the HRGS data.

6. Destructive analysis

The sub-samples of the irradiated UO_2 can be removed from the targets taken for DA in a hot cell after an examination of the capsules by neutron radiography. DA provides the uranium and plutonium content and isotopic composition. The cooling time can be estimated using the ^{241}Am/^{241}Pu ratio.

At least four aliquots should be taken from every sample. Two aliquots will be used for the IDMS measurement of the U and Pu concentration and isotopic composition, using ^{233}U and ^{242}Pu spikes. Two other aliquots will be used for the ^{241}Am measurement. One of them should be spiked with ^{243}Am. The TOPO (trioctylphosphine oxide) extraction method can be used to separate U and Pu from FP/Am for all aliquots. Additionally, separated Am + TRU fractions should be measured for the ^{241}Am activity by gamma and alpha spectrometry

The U isotopic composition allows verification of the type of irradiated U (i.e. depleted, natural or enriched U). This data can be confirmed by HRGS measurement of the low irradiated targets (see above).

The DA data can be used to evaluate the ^{238}U, ^{239}Pu and ^{235}U neutron cross-sections corrections. The U/Pu ratio is sensitive to the ^{238}U neutron absorption cross section, which is shielded strongly in the neutron capture resonance energy region. The measured U/Pu ratio provides a more correct value of the ^{238}U resonance integral and improves the accuracy of the Pu fraction estimate obtained with the NA code.

7. Conclusions

Verification procedures for irradiated U targets are described. The main objectives of the verification are to confirm the operator's declarations regarding type and quantity of the irradiated material, irradiation time and cooling time. The use of HRGS measurements combined with DA measurements allows the verification goals to be successfully achieved.

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A combined calorimetry, neutron coincidence counting and gamma spectrometry system (CANEGA) for enhanced plutonium mass and isotopic assay

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Abstract. A measurement approach for enhanced plutonium assay combining three passive non-destructive measurement techniques – calorimetry, neutron coincidence counting and gamma spectrometry (CANEGA) – has been designed and evaluated. The combination of the measured quantities obtained from the three techniques not only leads to an improvement of the plutonium assay through redundant and complementary measurement information, but also provides a more complete fingerprint for any plutonium-bearing sample under assay. Further, the combined measurement information allows one to derive a more reliable estimate for the ^{242}Pu isotope abundance not directly measurable by gamma spectrometry. A conceptual feasibility and design study for a transportable CANEGA prototype instrument has been carried out with the aim of defining the most promising and advantageous instrument configuration.

1. Introduction

The adopted primary non-destructive assay (NDA) approach in Safeguards for Pu mass measurements is to combine passive neutron coincidence counting (PNCC) with high-resolution gamma spectroscopy (HRGS). The Pu isotopic abundances are determined from gamma spectra taken by HRGS and analysed by codes like MGA and FRAM. However, there are some inherent limitations and drawbacks of this NDA approach for the Pu mass determination. The main limitation accrues from the missing information on the abundance of the isotope ^{242}Pu, which significantly contributes to the measured neutron coincidence rate in PNCC, but cannot be determined directly by HRGS. This problem is only partially solved through the application of isotope correlations estimating the relative abundance of ^{242}Pu from relations to ratios of other plutonium isotopes measurable by HRGS. Any error on the ^{242}Pu determination affects the performance both of the plutonium isotope abundance measurements and of the quantitative determination of the amount of plutonium. Another drawback of the combined PNCC+HRGS approach is related to the PNCC measurement itself, which is not really tamperproof (traces of ^{244}Cm, for example, can quickly invalidate the PNCC measurement), and which even for small samples requires careful neutron multiplication corrections in order to arrive at unbiased measurement results.

At the Institute for Transuranium Elements (ITU), Karlsruhe, we are also evaluating the alternative NDA approach for Pu mass determination, namely the combination of calorimetry and HRGS. Although calorimetry is so far not represented among the NDA instrumentation of the Safeguards authorities, it yet provides some attractive features compared to PNCC. Its main asset is certainly the ease of measurement interpretation through the insensitivity to all kinds of sample properties. From

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our recent investigations and application of calorimetry to small sample measurements of gram amounts of representative reactor-grade plutonium materials we have gained pertinent experiences with this technique with typical samples encountered in Safeguards verification measurements [1].

The experiences gained from the parallel use of calorimetry and PNCC in conjunction with HRGS for the non-destructive plutonium assay have led to the proposal of a combined calorimetry, neutron

coincidence counting and gamma spectrometry (CANEGA) measurement approach for the non-destructive plutonium assay [2, 3]. The combination of the directly measured quantities from the three techniques not only allows a fairly good determination of the ^{242}Pu isotope abundance, but also contributes through the use of redundant and complementary information to an improved plutonium assay. The combined NDA approach also proved to be of value for the assay of special types of nuclear materials, for which a single NDA technique normally is not able to give an adequate measurement answer. Such kind of special nuclear materials arise, for example, from R&D work at ITU on future fuel cycles including minor actinide transmutation.

While in a Safeguards analytical laboratory like ITU the three NDA measurements can be performed, in principle, with resident separate instruments, it would be yet of practical advantage for a potential on-site application of CANEGA to integrate the three NDA techniques into one single, transportable instrument. A corresponding feasibility and design study has been carried out and will be discussed in this paper.

2. Concept and performance of CANEGA

The principle of the CANEGA assay approach, its performance as well as the improvements to be expected from the combined NDA measurements have been previously described [2, 3]. Only a brief summary with some additional performance data is therefore presented below.

2.1. Concept for the ^{242}Pu determination

The combined calorimetry, neutron coincidence counting and high-resolution gamma spectrometry measurements can directly determine a total of six quantities from a plutonium sample:

the thermal power *P*, the amount of m_{240}-*effective* and the plutonium isotope weight ratios m_{238}/m_{239}, m_{240}/m_{239}, m_{241}/m_{239} and the ratio m_{Am}/m_{239} from gamma spectrometry. In the following we denote the isotope weight ratios relative to ^{239}Pu as R_{238}, R_{240}, R_{241} and R_{Am}.

Taking the ratio of the thermal power P over m_{239} yields:

$$P/m_{239} = P_{238}\cdot R_{238} + P_{239} + P_{240}\cdot R_{240} + P_{241}\cdot R_{241} + P_{242}\cdot R_{242} + P_{Am}\cdot R_{Am}\ , \qquad (1)$$

where the quantities P_i denote the specific thermal power of the respective isotope. In Eq. 1 the term $P_{242}\cdot R_{242}$ can be reasonably neglected without introducing a significant error (typicall less than 0.2%) because of the very low specific thermal power P_{242} of ^{242}Pu. Eq. 1 then allows to calculate the quantity m_{239} from the measurement observables P, R_{238}, R_{240}, R_{241}, R_{Am} and the known specific heat values P_i. With the knowledge of m_{239} the isotope ratio $R_{242} = {}^{242}Pu/{}^{239}Pu$ is easily calculated from the following equation containing only known constants and measured quantities:

$$m_{240eff}\,/m_{239} = \gamma_{238}\cdot R_{238} + R_{240} + \gamma_{242}\cdot R_{242} \qquad (2)$$

In this equation we are using for the coefficients γ_i , which proportion the measured Reals rates from ^{238}Pu and ^{242}Pu relative to ^{240}Pu, our experimentally determined values reported in Ref. 4. With the value for R_{242} thus obtained, and with the measured gamma-spectrometric values for R_{238}, R_{240} and R_{241} the new isotopic composition is then calculated to yield improved values for P-eff and ^{240}Pu-eff as input data for the Pu mass determination from calorimetry and PNCC.

2.2. Performance of CANEGA

In addition to results previously reported we have applied the CANEGA concept to another set of comparative NDA measurement data recently produced at ITU. Sample data for the set of 19 PuO_2 powder, MOX powder and MOX pellet samples used for the test are listed in Table 1. In terms of sample amount and isotopic composition the samples are representative for the type of verification samples encountered in the routine Safeguards analyses at ITU.

The non-destructive calorimetry, PNCC and HRGS measurements were performed before sample dissolution for the classical destructive analyses, viz. IDMS and/or titrimetry for the U and Pu concentration, and TIMS for isotopic composition, which provided the reference values for comparison. The calorimetry measurements were carried out with a high-sensitivity small sample calorimeter (Model TAM III from Thermometric AB), while for the combined PNCC and HRGS measurements the samples were counted in the so-called OSL neutron/gamma counter [5]. Both the calorimeter and the neutron counter were calibrated with the same reference material (a sealed sample with 500 mg of isotopically pure $^{240}PuO_2$). The thermal power of the samples (in mW) as measured in the calorimeter, and the neutron-coincidence ('Reals') rates obtained from the samples with OSL counter (detection efficiency = 40%) are also listed in Table 1 for information. The right-hand column in the Table quotes for the individual samples the approximate counting times required to reach a 0.2% counting precision for the Reals rate.

Table 1. Sample and measurement data for the analysed samples.

Type of sample	Sample mass (g)	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Am-241	P (mW)	Reals (cps)	Time for 0.2% NCC precision (h)
				wt. %						
PuO2	1.03	2.76	51.48	28.45	9.05	8.26	1.98	18.56	51.12	1.4
PuO2	1.24	2.84	51.58	26.97	10.25	8.37	0.74	21.93	61.47	1.1
PuO2	1.02	2.65	53.65	27.50	7.91	8.29	3.28	18.61	48.67	1.4
PuO2	1.03	2.35	55.26	27.05	7.66	7.68	1.91	16.27	47.32	1.5
PuO2	1.04	2.16	56.72	26.46	7.87	6.78	2.69	16.12	45.10	1.5
PuO2	1.06	1.68	59.00	25.73	7.50	6.09	2.39	13.82	42.35	1.6
MOX pellet	6.08	3.11	51.69	27.29	9.41	8.50	0.79	6.30	16.96	4.1
MOX pellet	7.91	2.93	52.27	27.00	9.51	8.30	0.80	9.06	25.13	2.8
MOX pellet	7.56	2.31	54.20	26.47	9.54	7.48	0.95	10.08	31.04	2.2
MOX pellet	7.25	2.43	54.79	26.06	9.23	7.48	0.89	8.54	25.25	2.7
MOX pellet	5.68	2.04	55.92	25.76	9.44	6.85	0.88	6.03	19.13	3.6
MOX pellet	6.62	2.11	57.21	26.40	7.39	6.89	2.40	5.39	15.52	4.5
MOX pellet	7.80	1.95	57.27	26.37	7.60	6.80	2.28	6.09	18.31	3.8
MOX pellet	7.23	2.04	57.30	26.14	7.78	6.74	2.42	9.65	27.88	2.5
MOX pellet	6.82	1.95	57.50	26.19	7.69	6.67	2.24	8.72	25.98	2.7
MOX powder	1.54	2.95	52.06	27.09	9.56	8.33	0.76	7.02	18.86	3.7
MOX powder	5.08	2.15	55.79	26.14	8.88	7.05	0.96	4.18	12.79	5.4
MOX powder	4.92	2.04	55.91	25.76	9.44	6.85	0.88	5.11	15.71	4.4
MOX powder	4.94	2.00	57.51	25.42	8.49	6.58	1.26	4.79	14.29	4.9

Three different sets of data for the isotope ratio R_{242} have been evaluated for comparison: one set derived from the CANEGA approach (applying Eqs 1 and 2), and two further sets derived from isotope correlations of the type $R_{242} = a\cdot(R_{238})^b \cdot (R_{240})^c$. In each case only gamma-spectrometric isotope ratios evaluated with the analysis code MGA (version 9.5) were used as further input data.

Appropriate coefficients a, b, and c in the above isotope correlation depend on the type of plutonium. According to our established criteria for the categorisation of the material type [7], the plutonium in all of the samples has been identified as PWR plutonium. With this classification we have applied two different sets of coefficients:

$a = 1.313, b = 0.33, c = 1.7$ as previously recommended [6], and
$a = 1.441, b = 0.484, c = 1.149$ as recently evaluated at ITU [7].

The R_{242}-values obtained from the above correlations and from the combined calorimetry, PNCC and HRGS measurements (CANEGA) were then compared with the "true values" from mass spectrometry (TIMS). The average percentage differences and their standard deviations for the given set of 19 measurement samples are given in the 2nd column of Table 2.

Table 2. Average percentage difference and standard deviation between R_{242}, P-eff and ^{240}Pu-eff values calculated with HRGS /MGA and TIMS isotopic data.

Source of R_{242}	R_{242}	P-eff	^{240}Pu-eff
Isotope correlation (previous coefficients)	-1.23 ± 4.23	-0.08 ± 0.64	-0.64 ± 1.04
Isotope correlation (ITU coefficients)	-0.17 ± 3.54	-0.15 ± 0.61	-0.40 ± 0.85
CANEGA	1.11 ± 2.02	-0.14 ± 0.51	0.31 ± 0.64

We note that the R_{242}-values obtained from the two applied correlations and from the CANEGA approach show comparable average differences to the TIMS reference values. However, the scatter of the CANEGA results relative to the TIMS values is reduced by about a factor of 2 compared to the correlation data.

One remark concerning the absolute accuracy of the R_{242}-values derived from CANEGA may be appropriate. The R_{242} values calculated from Eqs. 1 and 2 critically depend on the accuracy of the measured thermal power P from calorimetry, and on the measured mass of m240-effective from PNCC. Our previous sensitivity studies [2] have shown that for higher-burnup plutonium a bias in the measured value of P or in the measured value of m240-eff would each propagate as a roughly three times larger bias into the derived value for R_{242} (but with opposite signs). While with calorimetry the thermal power P can be normally measured with high accuracy, it is by no means a matter of course to achieve in real measurements a bias-free determination of m240-eff from PNCC, because even for small samples the measured Reals rates require corrections for neutron multiplication effects of up to 4%. This is illustrated in Fig. 1, which displays for the given set of samples the corresponding correction factors determined from MCNP calculations. A systematic bias of slightly more than a quarter of a percent in the PNCC measurement of the effective ^{240}Pu mass would therefore automatically lead to a bias of about 1% in the derived R_{242}-values.

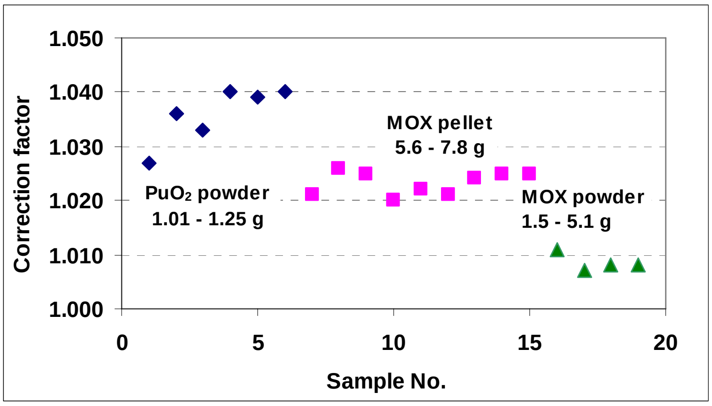

Fig. 1. Correction factors for neutron multiplication obtained from MCNP calculations for the set of samples listed in Table 1.

The R_{242}-values from the isotope correlations and from CANEGA were combined with the gamma-spectrometric ratios for R_{238}, R_{240} and R_{241} to obtain the complete isotopic composition from gamma spectrometry. With the respective isotopic composition the isotope-specific quantities P-eff and ^{240}Pu-eff required for the Pu mass evaluation from calorimetry and PNCC were finally calculated. The average percentage differences relative to the corresponding values obtained with TIMS isotopic data are listed in Table 2. It is obvious that the P-eff values are not as sensitive to R_{242} as the ^{240}Pu-eff values, which show some improvement with the improved R_{242}-values from CANEGA.

3. Design study for a CANEGA instrument

Having demonstrated the principle benefits of the CANEGA approach for an improved plutonium assay we decided to take a further step with a feasibility study for the design of a CANEGA instrument. For this purpose ITU has commissioned A.N. Technology, Wallingford, UK, which has experience with all three NDA techniques involved, to carry out a corresponding feasibility study into the design of a transportable combined calorimeter, neutron and gamma measuring device for on-site use in nuclear facilities. The study has been conducted in two phases, with phase 1 as a conceptual design phase reviewing design options, and phase 2 performing some detailed modelling for the finally selected configuration.

3.1. Design considerations

The design considerations were addressing all aspects of the system. This includes the overall performance, hardware and software requirements for the system as well as the specific requirements for each measurement type, i.e. calorimetry-, neutron coincidence- and gamma measurements. The basic specifications were:

- A CANEGA system for small sample measurements (gram-size PuO_2 powder or MOX powder and pellet samples);
- Measurement cavity with dimensions of 40 mm dia x 80 mm heigh;
- Neutron detection efficiency as high as possible, ideally close to 40 % as obtained with the existing OSL neutron/gamma counter installed in the Euratom on-site laboratories;
- Calorimeter sensitivity as large as possible (larger than 100 μV/mW) assuring a measurement repeatability of 0.1% at a thermal sample power of 10 mW;
- High-resolution HPGe detector for the low-to-medium energy range (up to 400 keV) subtending a solid angle relative to the sample of not significantly smaller than 10^{-2} sr.

It has been realized that the specifications were ambitious, and that at the end probably some compromises in terms of performance would have to be made in view of the sometimes conflicting requirements, especially for the calorimeter and neutron measurements.

3.2. Design options

Several designs have been considered to arrive at an optimum design for a combined measurement system. Advantages and disadvantages of each design have been assessed with respect to thermal block/moderator features, practicality of construction and ease of operation and maintainability.

After initial review of the design it was realised that a 40 mm diameter by 80 mm high sample chamber had implications for the performance and size of the system. This requirement was reduced to a 30 mm diameter by 80 mm high chamber.

For optimum performance the calorimeter should be preferably of the twin-cell design, with identical measurement and reference chambers. This leaves options for two fundamental configurations:

a) A side-by-side design, more closely resembling the classic design of ANTECH's small sample calorimeters, whereby both cups are mounted eccentrically in the thermal block as shown in the example given in Fig. 2.

b) An over/under design, in which the sample cup is placed axially directly above the reference cup and both are thermally linked. An example of this type of configuration is shown in Fig. 3.

The investigated options are essentially variations upon these two configurations, differing in the layout of the moderator/thermal block and in the way the gamma detector is incorporated into the system. There were in total eight different options that have been selected and reviewed.

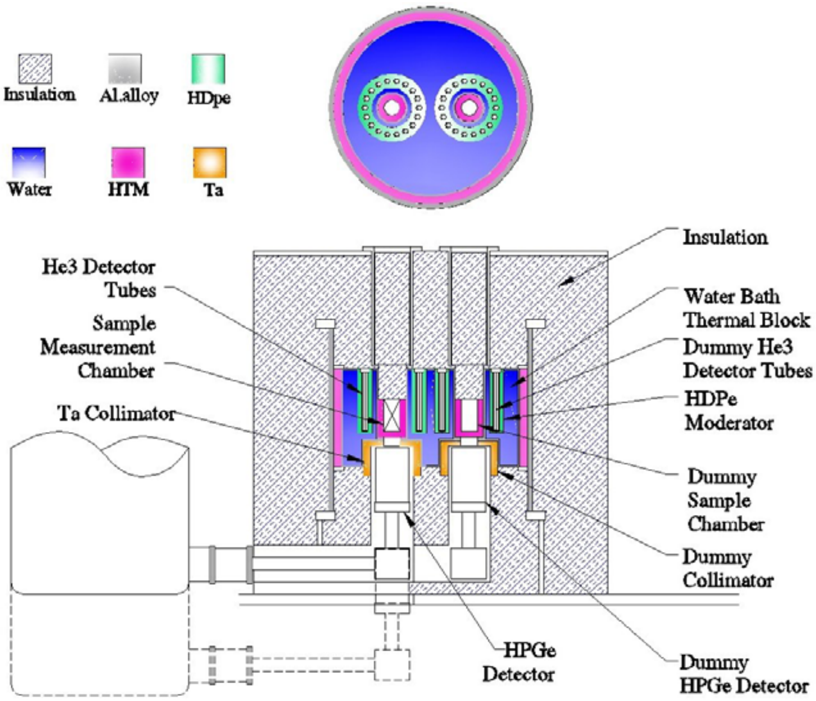

Fig. 2. Example for a side-by-side design.

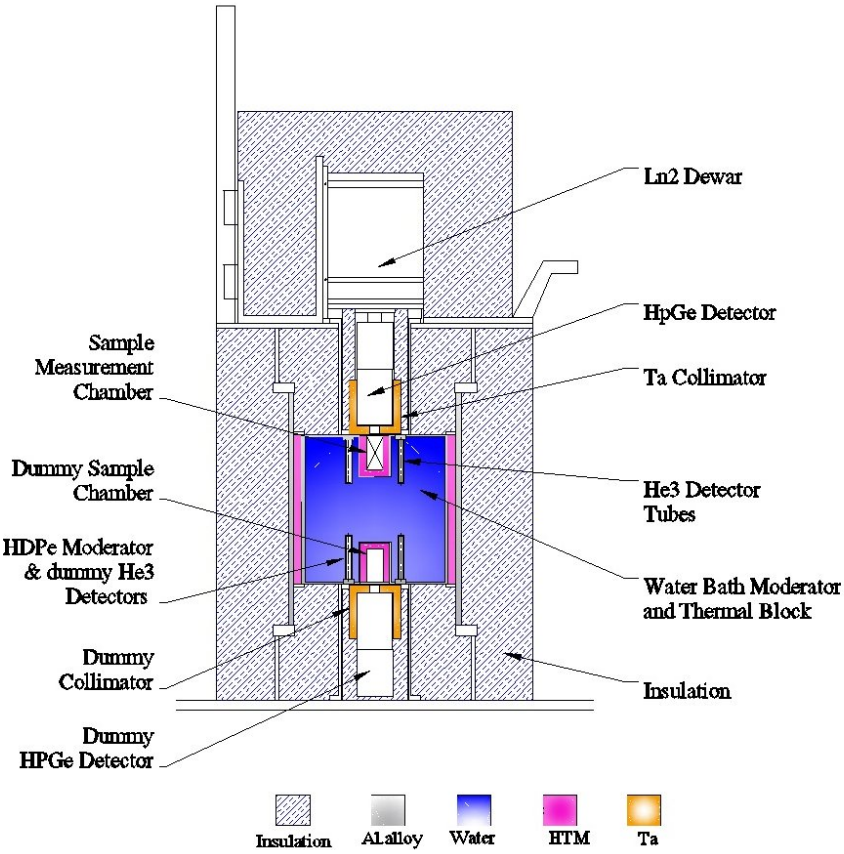

Fig. 3. Example for an under/over design.

3.3 Proposed configuration

The option finally selected for further optimisation through modelling is a side-by-side configuration similar to that shown in Fig. 2, with ^{3}He tubes in close proximity to the sample chamber and a side-mounted gamma detector. The moderator/thermal block assembly consists of a mixed polyethylene/water configuration as shown in Fig. 4 in a simplified vertical section of the detector.

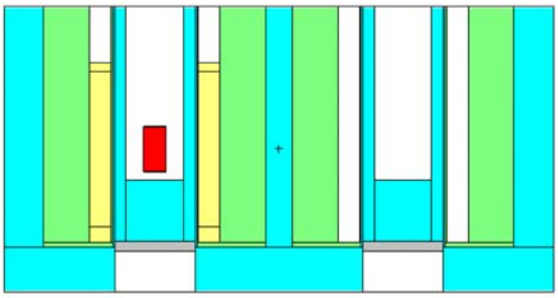

Fig. 4. Simplified cross-section of side-by-side model (gamma detector omitted).

3.4 Monte Carlo modelling (neutron measurement)

The initial model consisted of a single row of 12 x 4 atm ^{3}He tubes with an active length comparable to the height of the sample (80 mm) mounted in a polyethylene moderator positioned in close proximity to the sample as per the sketch shown in Fig 4. The rationale behind this design was to look at the efficiency of a compact (low height) thermal element. This design yielded an efficiency of approximately 12%, far below that required.

The reasons for the low efficiency were considered to be due to a combination of factors such as tubes having a short active length, low ^{3}He gas fill pressure, not enough detectors and an insufficient volume of moderator.

Following this the design was refined to include a double row of He^3 tubes with a much longer active length above and below the chamber. The thickness of polyethylene moderator was increased, the tube fill gas pressure varied and the respective radial distance of each row of tubes from the centre of the sample axis also varied.

The results of the modelling conclude that 40 % efficiency can be met by using a double ring of ^{3}He tubes, one ring positioned at a distance of 6.65 cm and the other 11.15 cm from the vertical axis of the sample. It can be achieved with a fill gas pressure of 6 atm and active length of 24 cm or a fill gas pressure of 8 atm and active length of 22 cm.

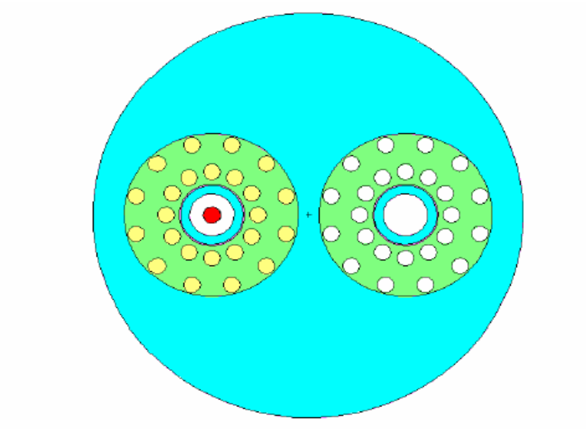

Fig. 5. Optimised neutron detector array.

Achieving maximum efficiency will need to be balanced with other aspects of the design relating the gamma and calorimeter measurements. The polyethylene moderator is in close proximity to the outer wall of the sample chamber. This directly influences how closely the gamma detector can be coupled to the bottom of the sample and the amount of water that can be circulated between the moderator and sample chamber wall - hence the cooling efficiency. These are design issues that will need to be taken into account before any such system is manufactured.

3.5 Modelling gamma measurement

The gamma detector is mounted under the sample, looking up through the base of the sample chamber. Surrounding the detector head is a tantalum shield with an aperture through which the sample is viewed. The performance of the gamma detector has been modelled for a number of cases of varying detector to sample distances and absorbing media.

For the initial model the PNNL (Pacific North-west National Laboratories) simulation code SYNTH was used to generate synthetic gamma ray spectra after defining various parameters for the sample, source, absorbers, detector and electronics. The basic arrangement proposed for the gamma component of the CANEGA system was configured and parameters varied to produce a range of spectra.

A coaxial Ge detector was selected as it was considered the most appropriate to enable the correct parameters for the detector to be modelled. For this option the efficiency, diameter and length can be changed to match most shapes of detector. The resolution can be varied but SYNTH doesn’t enable a resolution of 0.6 keV to be achieved in the 100 keV region of the spectrum. This, therefore, means that the SYNTH spectra cannot be used directly with MGA to give a realistic value for the accuracy of the MGA result for real spectra.

There are a number of peaks that are isolated and not part of a multiplet, ^{239}Pu at 129 keV, ^{241}Pu at 148 keV, ^{238}Pu at 152 keV and ^{240}Pu at 160 keV. These peaks were used for a cross comparison with real spectra. The real spectra were analysed with MGA and spectra with an error on the specific power value of less than 1% were chosen. The gross and net counts in the above peaks were then counted using a set of ROI's. These ROI's were then used to calculate the gross and net counts in the peak for the SYNTH spectra. Since these peaks are isolated the counts in these peaks will be comparable between the real and SYNTH spectra. It is assumed that if the SYNTH spectra have a greater number of net counts in the peak, then that SYNTH model will, for a real measurement with a detector of resolution 0.6 keV, give an error on the specific power value of less than 1%. A summary of the cases modelled is shown below.

Table 3. Summary of results for a range of gamma modelling parameters.

Case (Note 1)	Pu Isotopic composition	Distance source-to-detector (cm)	Detector cross sectional area (cm^2)	Detector depth (cm)	Absorbing material (cm) (Note 2)	Count time (h)	Net counts OK (Note 3)
1	Recycled MOX	18	3.6	1.5	1 Al	1	All OK
2	Recycled MOX	18	1.6	1.5	1 Al	1	129keV Pu239 failed
3	AGR	18	1.6	1.5	1 Al	1	129keV Pu239 failed
4	AGR	18	3.6	1.5	1 Al	1	All OK
5	AGR	18	3.6	1.5	1.5 Al 5 H_2O	1	All OK
6	AGR	25	1.6	1.5	1.5 Al 5 H_2O	1	129keV Pu239 failed
7	AGR	25	3.6	1.5	1 Al	1	All OK
8	AGR	25	3.6	1.5	1.5 Al 5 H_2O	1	All OK

Notes:

1. The basic arrangement was as follows:
   a. 1g sample of Pu metal, 0.1749 cm^2 cross sectional area, 0.3 cm thick.
   b. Coaxial Ge detector with dimensions varied as per columns 4 and 5
   c. Detector end cap thickness 1mm Aluminium
   d. 8192 channels defined with a gain of 0.075 keV/ch
   e. Count time of 3600 s
2. The remaining absorbing material that makes up the total distance between source and detector is air.
3. In the cases where the 129 keV peak failed the counts in the peak were between 4958 and 8060 compared to the real spectra values of between 9472 and 9864. This indicates that with double the count time, i.e. 2 hrs, the required number of net counts in the peak could be achieved.

Detailed gamma modelling suggests that if a 3.6 cm^2 cross sectional area by 1.5 cm deep coaxial detector is chosen it could be up to 25 cm away from the sample and view it through up to 5 cm of water, 1.5 cm of aluminium and 18.5 cm of air. The detailed design may allow us to decrease this distance and the amount of absorber material but this shows that the minimum requirements of 1 % precision at 100 keV in a 1 h counting time can be met with the chosen concept design.

3.6 Modelling calorimeter measurement

The same design used to model the neutron and gamma measurement criteria was used to model the calorimeter properties.

The heat output from a sample is measured with a series of thermopile junctions installed in an annular air gap between the inner cylinder forming the sample chamber and an outer cylinder forming a leak-tight barrier between the sample chamber and water bath. The thermopile array measures the heat flowing from the sample to the water bath and provides a voltage output proportional to the heat flow.

The sample measurement chamber construction has been modelled using two types of heat flow sensors:

a) An array of close coupled pad thermopiles mounted on flat surfaces machined on the outer surface of the inner cylinder,

b) A single continuous strip of thermopile junctions installed to run in vertical strips within the air gap between the inner and outer sample chamber cylinders.

Both types of sensors exceed the requirement for a measurement sensitivity greater than 100 μV per mW. The design utilising pads offers the best solution as it provides the greatest signal output per unit area and could potentially give up to 4 mV per mW. Vertical strips of thermopile junctions would be expected to give up to between 0.1 to 0.5 mV per mW.

A measurement repeatability of 0.1% at 10 mW is routinely achieved with ANTECH's Model 601 Small Sample Calorimeter. This employs an aluminium thermal block to balance the need for fast measurements with thermal stability and repeatability. The proposed design for a CANEGA system will use a large volume water bath which has a higher thermal inertia, hence is more stable, in conjunction with more sensitive measurement and control electronics.

Further modelling demonstrated that to minimise the heat distribution error over the volume of the measurement cavity it will be essential to have an insulating layer of air between the bottom of the sample chamber and the outer chamber to minimise the heat flow through the bottom of the sample. In addition it will be important to have a highly insulating plug unit to prevent heat leaks from the top of the sample.

4. Conclusion

We consider the combined calorimetry, neutron and gamma measurements a viable approach for an improved non-destructive plutonium assay in smaller verification samples. By modelling the individual measurements in a combined instrument for a 30 mm diameter by 80 mm high sample chamber, and refining the model as we proceed, we have shown it is feasible to achieve a neutron measurement efficiency of 40 % and, although this constrains the gamma measurement by having to place the detector further away from the sample than desirable and view the base of the sample through a medium of water, air and aluminium it is possible to meet the gamma measurement requirements. Given the time it takes for the calorimeter and neutron measurement to complete, conducting the gamma measurement for up to 2 hours, or more, in parallel with the calorimeter measurement would improve the measurement accuracy.

As anticipated, incorporating neutron and gamma measurement systems into the calorimeter thermal element affects the calorimeter measurement. By utilising a high density of thermopile junctions to measure the sample heat output the required sensitivity can be achieved. Furthermore, using a water bath construction in conjunction with more sensitive measurement and control electronics improves the stability, hence, repeatability of the system.

Minimising the heat distribution error across the sample presents challenges. It is likely that forced circulation of the water, in particular in the region between the sample chamber and the polyethylene moderator rings, will be required to obtain good heat transfer. Being able to incorporate a small volume of water, typically up to 5 cm depth, between the gamma detector window and the sample base is beneficial.

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Towards more investigative analytical methods for nuclear safeguards and nuclear security applications

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Abstract. The introduction of strengthened safeguards, the implementation of the additional protocol (INFCIRC 540) and the nuclear material intercepted from illicit trafficking led to a more investigative character of analytical measurements. Consequently, the new sample types triggered the transfer of analytical techniques from the environmental area, from materials science or from the geological or cosmological area to the safeguards community. Environmental analysis and nuclear forensic science are experiencing significant developments and profit from the interdisciplinary approach.

The more specific questions will be asked in safeguards with respect to a given sample, the more investigative analytical methodologies will be required and the more thorough, interpretative and comparative evaluation of results needs to be done. Specific applications, possibly in combination with only minute amounts of sample call for methods of high sensitivity, low detection limits, high selectivity and high accuracy. The selection of the method or combination of methods is done according to the sample and according to the information required. Data interpretation is calling for reference information, for comparison samples and for a thorough understanding of the processes taking place throughout the nuclear fuel cycle.

Introduction

In the early days after implementation of safeguards agreements (INFCIRC 153) and the Euratom regulation 3227/76, measurements of nuclear material were the backbone of the verification measures. These measurements served the verification of declared amounts of nuclear material. Consequently, measurement methods were put in place, which provided information on the uranium, plutonium or thorium content in a given material. In addition, the isotopic composition of uranium was measured for verification of the amount of fissile isotopes. The isotopic composition of plutonium was measured as well. However, apart from verification of the nuclear material accountancy, the information inherent to the nuclear material was not exploited.

When the IAEA started introducing strengthened safeguards and when the additional protocol was implemented, the mandate of the IAEA expanded from the verification of correctness of a state's declaration to comprise also the verification of the completeness of such declarations. The detection of undeclared nuclear material or undeclared nuclear activities requires establishing a comprehensive picture of a state's nuclear activities and checking the consistency of the declarations against other evidence. In consequence, a tremendous need for information at

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different levels arises in order to enable the evaluation required under a strengthened safeguards regime.

All types of information sources can be drawn upon: satellite imagery, design information verification, on-site inspections, sample taking (comprising nuclear material samples and environmental samples) and others. In the present paper, we want to expand on the information inherent to samples of nuclear material and the techniques required for reading this information.

The Information Challenge

Verification of the absence of undeclared nuclear material or nuclear activities is "a priori" a very complex task. The answer needs to be composed of a variety of indicators, which allow drawing conclusions on the completeness of a state's declaration. The samples of nuclear material and the environmental samples taken provide a useful source of information on the processes applied. Let us recall in this context two main prerequisites:

1. The production and processing of nuclear material leaves (inevitably) traces in the environment. This is fundamental to environmental sampling. Depending on the cleanliness of the process and on the quality of the installations, the range of detectable traces can be rather narrow. The experience gained in many years of environmental sampling in general and of particle analysis in particular has demonstrated the power of this methodology. But it highlighted also limitations, which are mainly to poor measurement accuracy, leading to ambiguities in data interpretation.

2. Every production process will leave characteristic patterns in the material. There are, however, a number of measurable parameters which may vary as a function of starting material, of process parameters, of reagents used, of storage conditions or of vessel materials. The complexity of the data and the interrelations between individual parameters require a careful step-by-step approach from measurement to data to interpretation to information.

The data obtained via different measurement techniques may be categorized into two types: exogenic data, i.e. data that is self explaining (e.g. the $^{235}U/^{238}U$ ratio pointing at the enrichment of the material and the intended use). The majority of data is endogenic data, i.e. data that can only be understood with the help of reference data (e.g. comparison against data from known material, comparison against model calculations). The latter category is certainly more difficult to understand and requires more resources before a conclusion can be drawn. Chemical impurities, isotopic composition of the nuclear material, isotopic composition of accompanying elements, particle size and microstructure are data which are accessible through measurements and which allow to build information. Having said this, we have to note that measuring safeguards samples has moved from simple concentration measurements of the major components to a detective work, seeking for evidence and for objective proof of the self defined working hypothesis.

The information which the measurements and the data interpretation have to deliver has to prove (or disprove) the absence of undeclared nuclear activities. Consequently, appropriate measurement techniques need to be applied and, more importantly, the characteristic parameters need to be identified. The conclusion to be reached at the end of this evaluation process is based on:

- Consistency of information
- Coherence between samples or materials

- Conformity of findings with declared processes
- Comparison of data

In contrast to traditional safeguards, such an evaluation is not based on quantities of material, but rather on certain qualities of material.

The Measurement Challenge

The challenge in performing such measurements of investigative character is twofold: on the one hand, a wide spectrum of parameters needs to be measured; on the other hand, those parameters providing the most significant information need to be identified. The instrumental techniques applied for this purpose are well established; however the analytical method they serve needs to be adapted to the specific needs of this type of investigative safeguards analysis. For developing such methods, we can benefit from experiences made in other fields of science: e.g. nuclear forensics, isotope geology or material science. In the following sections we would like to illustrate the development and application of investigative measurement method. In the following chapter we will show the conclusions that can be drawn at this stage from such data and highlight also the remaining challenges, particularly with respect to interpretation.

Stable isotopes

Very little attention has been attributed so far to the measurement of stable isotopes for nuclear safeguards purposes. In the field of geochemistry this methodology has been successfully applied and more recently it has been introduced in nuclear forensics.

The application of oxygen isotope ratio measurements for geolocation purposes has been demonstrated several years ago. A correlation between the geographic location of the production site of Uranium oxide samples and the shift in the $n(^{18}O)/n(^{16}O)$ could be established [1]. Moreover, it could be shown that the method is also applicable to individual particles, i.e. the oxygen isotope ratios established by "bulk" measurements using thermal ionization mass spectrometry (TIMS) could be reproduced on individual particles using secondary ion mass spectrometry (SIMS) [2]. This type of geolocation does obviously not identify a specific plant, yet it provides a parameter for attributing the material to a region.

Another parameter that has been widely used in geochemistry and in environmental sciences is the isotopic composition of lead. Lead isotopes may be primordial (natural lead) or the may be produced through the decay of uranium isotopes. The small variations in the isotopic composition of natural lead have been used to locate the origin of some (formerly used) fuel additives (mainly consisting of tetra-ethyl lead). The adaptation of this methodology for nuclear safeguards and nuclear forensics purposes has been studied [3]. It could be shown that the lead isotopic composition of yellow cake provides useful information to distinguish between natural uranium materials of different origins. The chemical separation of the lead from the uranium sample needs to be performed in a clean environment, as lead is omnipresent in our environment and the natural lead from dust particles or from chemical reagents would lead to biased results.

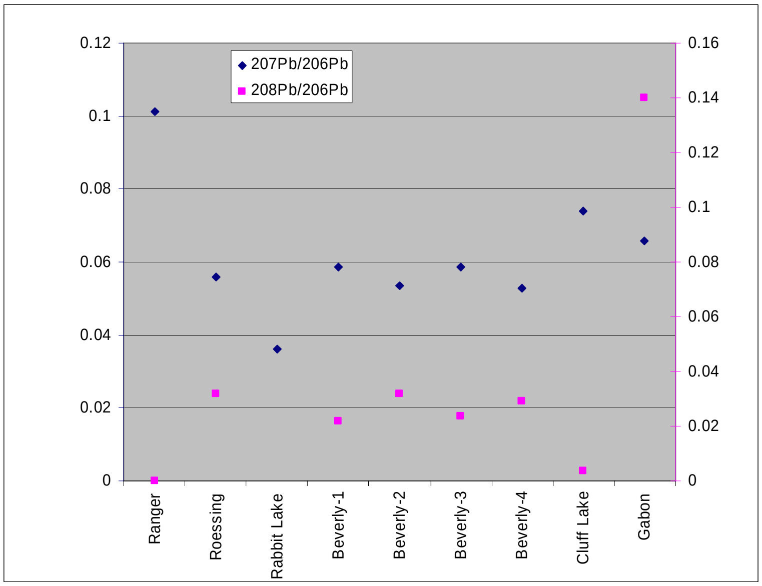

Figure 1 Lead isotope ratios observed in samples of yellow cake from different mines [2].

Lead is often used as shielding material for nuclear samples. This may introduce biases in the results as natural lead (from the shielding material) "contaminates" the lead contained in the sample. There are two possibilities for dealing with this problem: first, one can correct for all contributions from natural lead using the ^{204}Pb as pilot isotope. ^{204}Pb is not contained in radiogenic lead, and may therefore serve as indicator for the amount of natural lead present in a sample. The second option requires the availability of a reference sample from a suspected origin. In this case isotope mixture calculations can be performed, assuming a binary mixture between natural lead and the lead contained in the reference sample. An example is given in the table below, where a seized uranium ore sample had been wrapped in (natural) lead foil. The isotope mixture calculation showed that the measured isotopic composition can be fully explained by a binary blend of natural lead and the lead (as measured before) in uranium ore from Joachimstal (Czech Republic).

Sample	^{204}Pb	^{206}Pb	^{207}Pb	^{208}Pb
Fund-25	1.20	33.27	19.32	46.20
Nat. Pb	1.4	24.1	22.1	52.4
Joachimsthal	0.96	45.12	16.56	37.36
Mixture 56/44	*1.21*	*33.36*	*19.67*	*45.78*

Table 1 Measurement results of the lead isotope abundance (mole%) obtained for a sample of seized uranium ore (Fund-25) and for natural lead. The lower line shows the results of a blending calculation, assuming a mixture of 56% natural lead and 44% lead from uranium ore form Joachimsthal.

Highly accurate mass spectrometric measurements of isotope ratios allow the identification of small though significant and systematic variations in the isotopic composition of the elements due to natural isotopic fractionation. Through the availability of multi-collector ICP-MS instruments this methodology became also applicable to elements with a higher ionization potential.

Anionic impurities

Aqueous processing of nuclear material is encountered at a number of stages in the nuclear fuel cycle. In these aqueous processes mineral acids are frequently used. They leave anionic impurities in the material behind, together with those anions that were initially present in the starting material. We have studied such anionic impurities in samples of yellow cake from different origins. Depending on the type of ore from which the uranium was extracted and depending on the type of process applied and the associated chemical reagents used, the isotopic patterns generated in the yellow cake are significantly different. These patterns provide an additional clue for distinguishing materials from different origins or – if appropriate reference data is available – for relating a given material to a specific facility. Anionic impurities are easily measurable using ion chromatography. For data evaluation, the pattern of anionic species is more informative than the actual concentration values. Figures 2 and 3 show examples of chromatograms obtained from yellow cake samples from Canada and from Namibia [4].

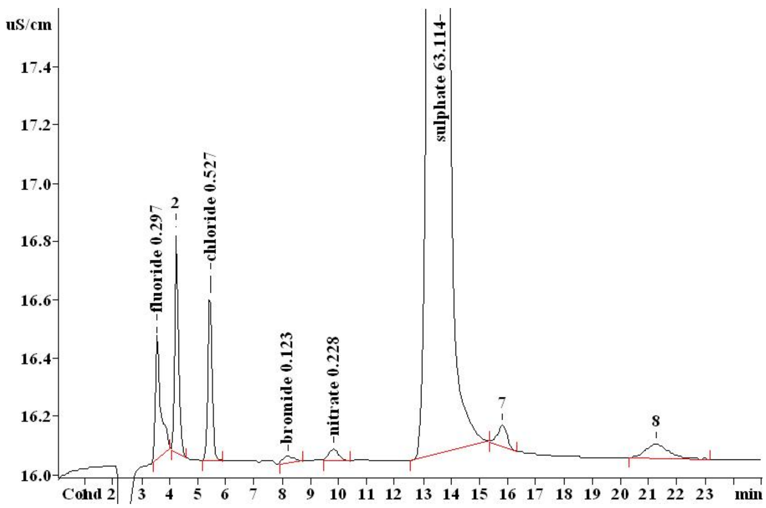

Figure 2 Anionic impurities in a sample from a Canadian Mine (Cluff Lake) as observed by ion chromatography.

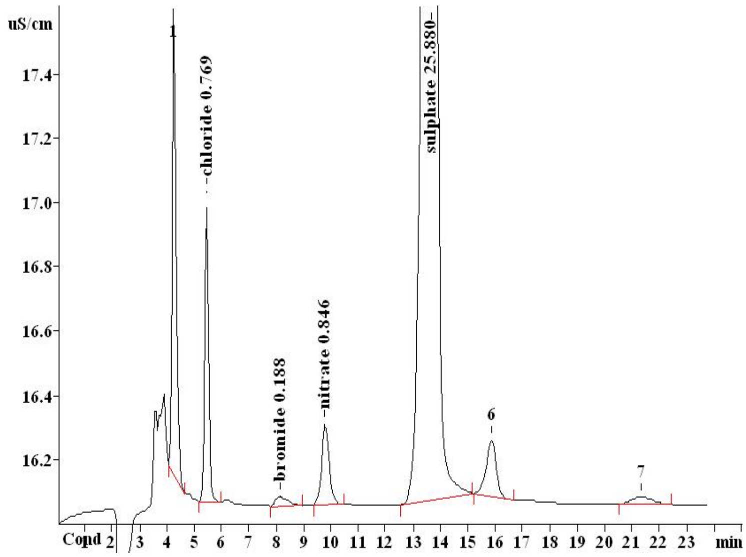

Figure 3 Anionic impurities in a sample from a Namibian Mine (Trünzing) as observed by ion chromatography.

Metallic impurities

Metallic impurities have already been investigated for "fingerprinting" oxidic and metallic materials. The concentration of such impurities is generally measured using ICP-MS (inductively coupled plasma mass spectrometry) or ICP-OES (Inductively coupled plasma optical emission spectrometry). Metallic impurities are present in samples of nuclear material at widely different concentration levels. In starting materials (e.g. ore concentrate) the impurities may have the character of accompanying elements and are present in relatively high concentrations. In intermediate products (e.g. yellow cake) the concentration of most of the chemical impurities has been drastically reduced. No further decrease of impurities is observed for most other products of natural uranium. The figure below shows metallic impurities in natural uranium compounds of different origins. Five samples from the same origin can be clearly recognized through their identical pattern of metallic impurities.

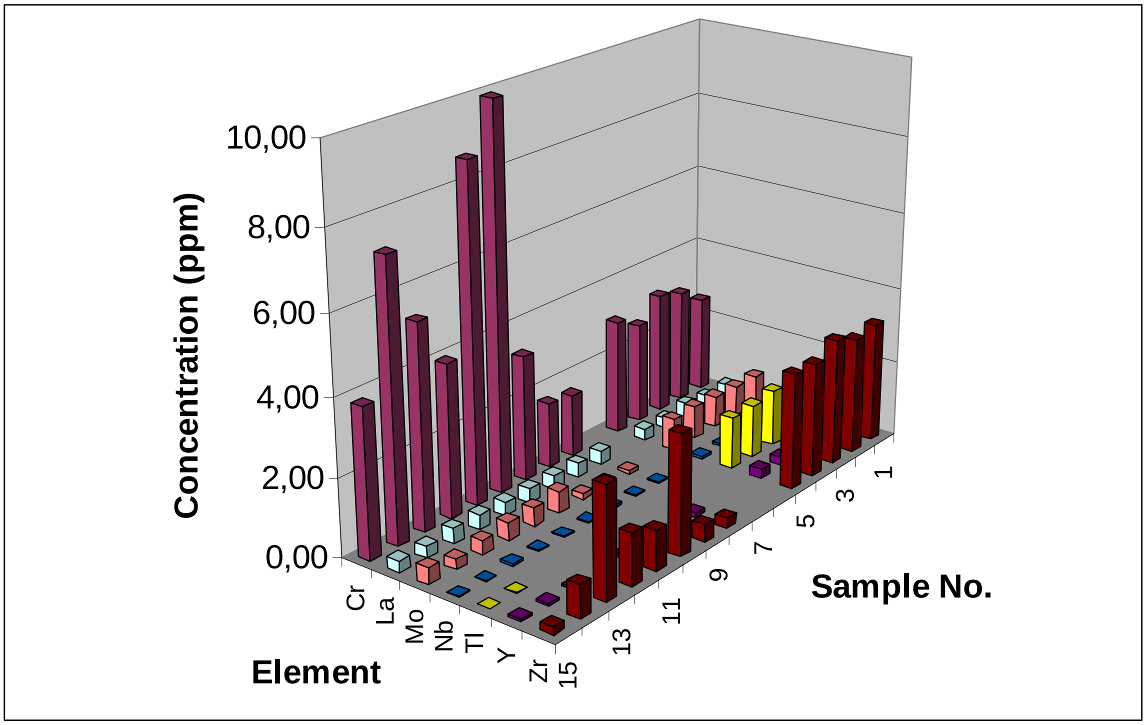

Figure 4 Selected metallic impurities in intermediate uranium products (natural uranium). Samples 1 to 5 are apparently of the same origin. The relative measurement uncertainty on the impurities is ± 15%.

Although metallic impurities are being used for identifying coherences between samples or batches of material, the systematics behind the impurity patterns are not well understood. As outlined above, metallic impurities may be carried into the material at different stages of the process. The concentration of some impurities may vary as a function of exposure time to the container material or to the storage tank, as they are leached from the surface of the walls. In sample analysis the concentration appears to be fluctuating randomly. One possible way out of this dilemma consists in looking at ratios of chemical elements. While the absolute concentration of the impurities may change, the ratio of certain elements will vary only within narrow limits. This applies in particular for elements of similar chemical behavior, like the rare earth elements.

Isotopic patterns

For long time the safeguards community has made use of the isotopic composition of nuclear material. Increased attention to the minor abundant isotopes in uranium was paid with the introduction of strengthened safeguards, when the need arose to establish capabilities for distinguishing between samples of (apparently) the same enrichment. The isotope abundance of ^{234}U and ^{236}U may help to verify coherence between different samples and consistency with declared operations. The presence of small amounts of ^{236}U will indicate a contamination with recycled uranium and hence point at reprocessing activities. In reprocessing plants, the isotopic composition of uranium and plutonium allow drawing conclusions on the reactor type in which the material has been irradiated. Table 2 shows the results of isotope abundance measurements (three sub-samples) on a sample seized in the context of a criminal investigation. Comparing the measured values to burn-up calculations, it has to be noted that uranium and plutonium are not originating from the same reactor type: plutonium shows an isotopic composition close to an

LWR reactor, while the uranium isotopic composition points at a natural uranium fuelled research reactor.

Isotope	**Isotopic Composition [Mass%]**		
	Q1.1	**Q1.2**	**Q1.3**
^{234}U	0.0159	0.0158	0.0158
^{235}U	0.3480	0.3501	0.3406
^{236}U	0.1383	0.1396	0.1361
^{238}U	99.4978	99.4945	99.5075
^{238}Pu	1.3168	1.3156	1.3213
^{239}Pu	59.6647	59.6101	59.8705
^{240}Pu	28.1990	28.2528	28.0606
^{241}Pu	5.3042	5.2993	5.3225
^{242}Pu	5.5153	5.5222	5.4250

Table 2 Isotopic composition of uranium and plutonium in a seized sample containing a radioactive liquor.

Isotope correlation techniques have been used in safeguards and in nuclear forensics. In particular the isotopic composition of plutonium is a useful indicator of the reactor type in which the nuclear material was produced. The neutron capture cross section of the individual plutonium isotopes vary in different way as a function of neutron energy. In consequence, the build up of plutonium isotopes is different in reactors with different neutron energy spectrum. This is reflected in the isotopic composition of plutonium. Knowing the plutonium isotopic composition, we can draw conclusions on the type of reactor in which the Pu has been produced.

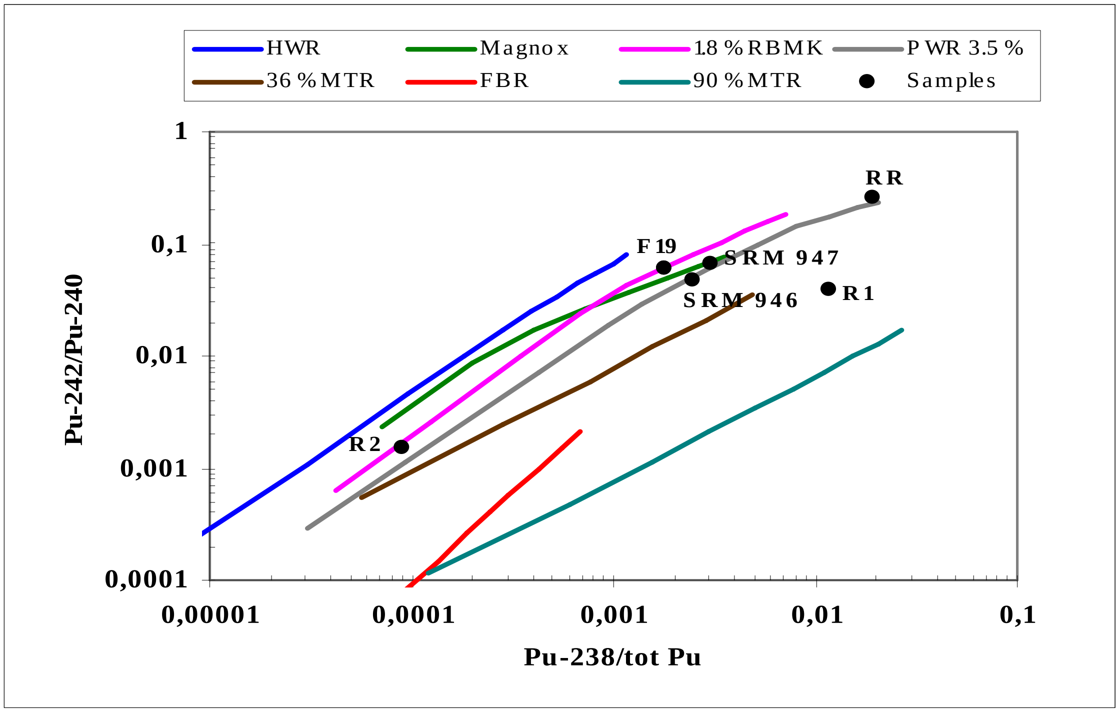

Figure 5 Pu isotope correlation for different types of reactors. SRM 946 and SRM 947 (both are NBS certified Pu reference materials) originate apparently from pressurized water reactors, as well as sample RR used in a round robin exercise. F19 and R2 denote seized materials, which can be attributed to an RBMK reactor.

Microstructure

Very little use has been made of microstructural information. Such information is obviously only partly of quantitative character and it is also phenomenological information. Still the particle size distribution, the grain size distribution and the surface structure of the particles are material characteristics which reflect the production procedure of the material. These data allow the direct comparison of samples enabling conclusions on coherence between samples. Figure 6 shows a comparison of two plutonium dioxide samples. The particles are shaped in platelets (scanning electron microscopy, SEM pictures top left and top right) and the two samples appear very similar. The evaluation of the particle size distribution does not show a significant difference between the two samples. Looking at the grain size distribution (lower two pictures obtained by transmission electron microscopy, TEM) a significant difference has to be noted. This points a two different production processes, indicating different origins of the two samples in question.

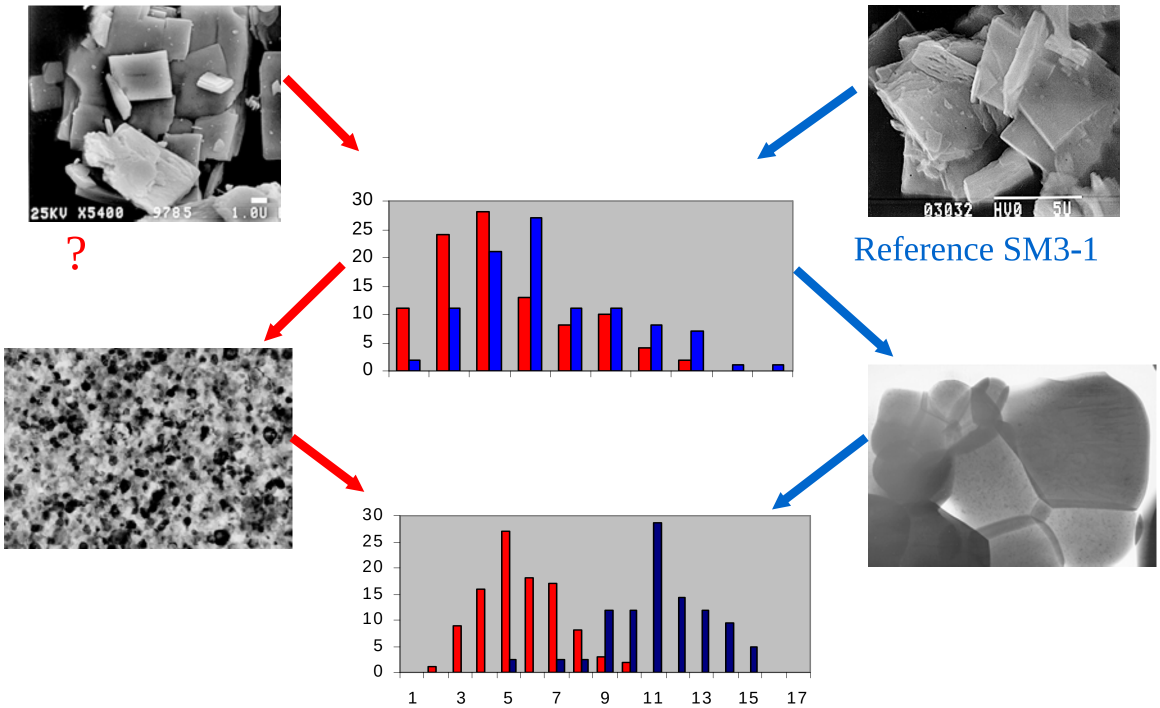

Figure 6 Comparison of particle size (top pictures) and grain size (lower pictures) in two samples.

Age determination

Age determination of nuclear materials makes use of the radioactive decay of these materials. Assuming a complete separation of the daughter products during the production process (i.e. during chemical purification of the material), we can determine the "age" of the material by quantifying the amounts of parent nuclide and of daughter nuclide. Age determination of plutonium is classically being performed by gamma spectrometry using the $^{241}Pu/^{241}Am$ parent/daughter ratio. The use of the uranium daughters of ^{238}Pu, ^{239}Pu and ^{240}Pu offers a consistency check [5], as the three parent/daughter relations should result in the same age – provided the separation of uranium was complete during processing of the material. Residual amounts of uranium isotopes will lead to biased results of the age determination. Figure 7 shows the relative bias of the age as determined for weapons grade plutonium using the $^{238}Pu/^{234}U$ relationship. The bias is a function of the age of the material (the older the material, the more ^{234}U is produced and the less any residual uranium will affect the result) and of the amount of residual uranium after the last chemical separation of the plutonium (the more residual uranium is left in the plutonium sample, the higher the bias will be). As can be seen from the model calculations, the parent daughter ratio used is very sensitive to residual amounts of uranium and thus lead to significant biases in the age determination. The data in the model calculations were obtained by combining burn-up calculations, decay calculations and isotope mixture calculations.

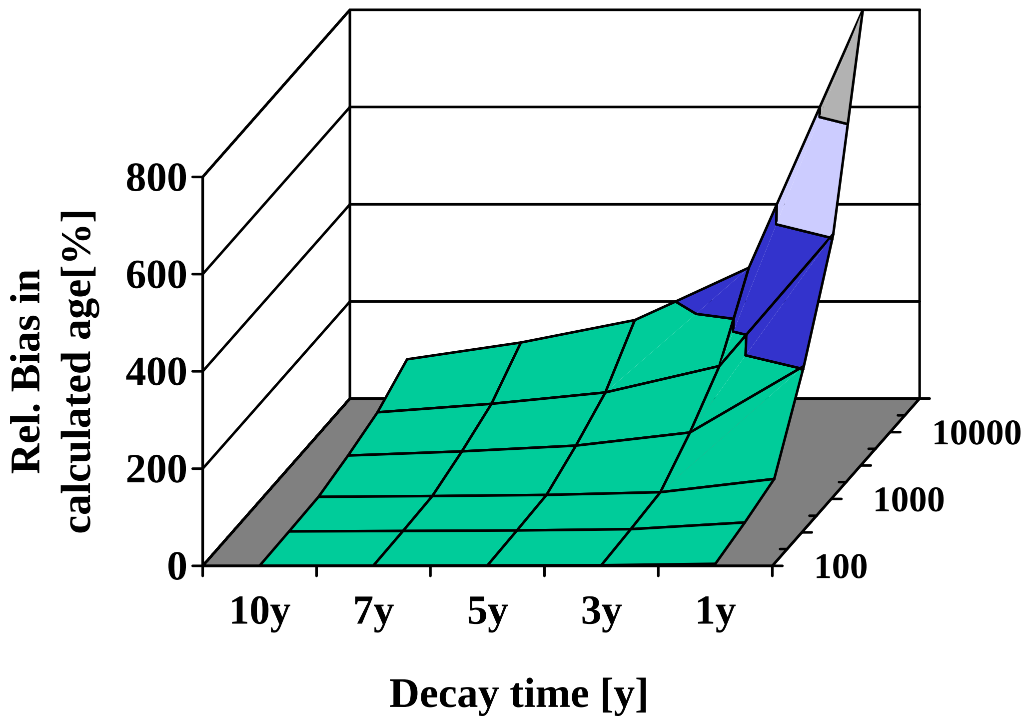

Figure 7 Relative bias in the age of weapons grade plutonium using the $^{238}Pu/^{234}U$ parent daughter ratio as a function of the age and of the amount of residual uranium.

Another interesting aspect in age determination is the question of the age of particles. Age determination of plutonium particles has been demonstrated earlier, the limitations are shown above. Age determination of uranium particles proves to be much more challenging, due to the long half-life of the uranium isotopes and to the limited amount of material available for measurement. The parent daughter ration $^{234}U/^{230}Th$ is being used for the determination of the last chemical separation of the uranium. The particles of interest in swipe samples from enrichment plants are typically only one micrometer in diameter. Based on this assumption, we can calculate the detection limit for age determination as a function of the age of the particles and of the enrichment. Assuming further a detection efficiency of 0.5% in the secondary ion mass spectrometer, we see from Figure 8 that age determination can only be successfully performed for particles of highly enriched uranium.

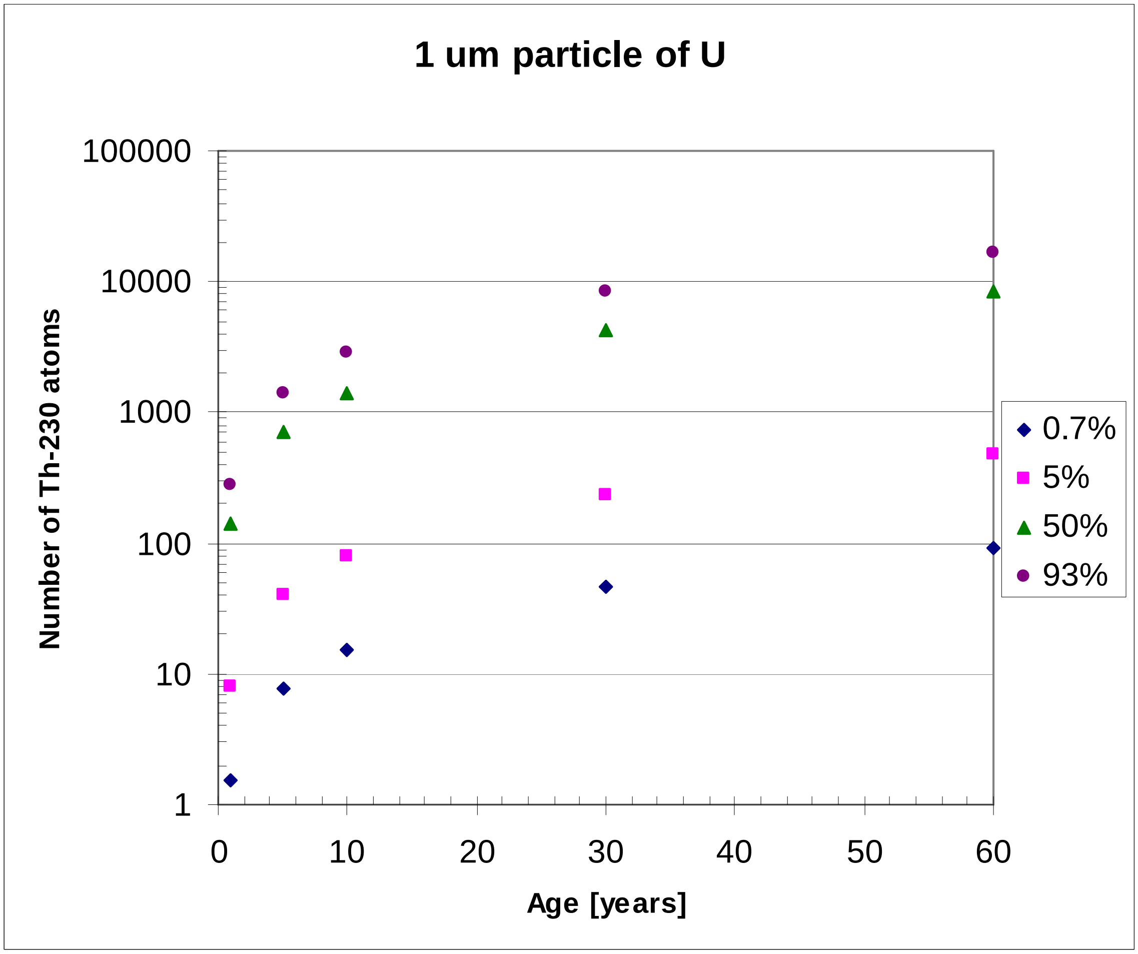

Figure 8 The decay of ^{234}U produces ^{230}Th. The number of ^{230}Th atoms contained in a uranium oxide particle (with an assumed number of 10^{10} atoms of uranium) depends on the age of the particle and of the initial enrichment.

The Evaluation Challenge

In order to properly evaluate the measurement data, the availability of reference information is required, in particular for endogenic data. To some extent the safeguards community can draw upon experience and data available in the geochemical community. Variation of the isotopic composition of the chemical elements have been studied in other contexts and in some cases cadastral registers of isotopic data are available. Such information is for instance available for $n(^{18}O)/n(^{16}O)$ in rainwater or for lead isotopes in natural lead. Information related specifically to nuclear material is less widely available. Data on metallic impurities in nuclear fuels are often subject to commercial confidence. Microstructure data have not been collected for safeguards evaluation purposes, most likely for the largely qualitative nature of this information.

In order to make best use of the additional information that can be obtained through the methods outlined above, a comprehensive set of reference data or of reference samples (i.e. samples obtained from known sources and produced though known processes from known starting materials) needs to be established. On the other hand, the information arising from the measurement methods outlined above requires a multidisciplinary team of analysts, covering the areas of chemistry, physics and material science.

Conclusions

The challenges associated with strengthened safeguards call for more investigative analytical methods. The verification of treaty compliance according to comprehensive safeguards agreements and according to the additional protocol are associated with a tremendous need for information. Part of the information required for the evaluation of the completeness of a state's declaration is inherent to the nuclear material. Advanced measurement methods need to be introduced in nuclear safeguards, methods that are very much of investigative character. Such methods are being applied in the area of nuclear forensics. Consequently, we will see a convergence of nuclear forensic and of classical safeguards analysis.

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Building knowledge by destructive analysis: The ESARDA Working Group on Destructive Analysis

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Presented by R. Wellum

Abstract. The ESARDA (European Safeguards Research and Development Association) Working Group on Standards and Techniques for Destructive Analysis (WGDA) was created almost three decades ago, when chemical methods were the workhorses of sample analysis for accountancy and verification purposes. Today, the majority of measurements for element and isotope assay are performed by radiometric techniques, i.e. applying 'non-destructive' methods, and chemical or physico-chemical methods are applied essentially for quality control purposes or when superior accuracy is required (e.g. bias defect detection). In modern safeguards, however, much more information on a sample may be required than simply its fissile material content.

The ESARDA WGDA acts as a forum for measurement experts to exchange experiences and views on the scientific and technical questions related to sample analysis. The group addresses issues on measurement quality (e.g. Target Values), on measurement methods, on sampling errors and on the analysis of specific types of samples such as environmental and nuclear forensic.

Introduction

Traditional safeguards involves "a quantitative verification of the accountancy of fissile material by independent verification measurements" [1]. This implies the comparison of two (or more) independently obtained measurement results. The fissile material accountancy of the plant operator is based on measurements. For verification purposes, the safeguards authority also has to perform measurements of the amount of nuclear material. These measurements normally involve measurements of bulk samples, i.e. mass or volume determination of a batch of material stored in a vessel, tank or other type of container. It furthermore involves element and isotope assay of the analyte of interest. The assay of a nuclear material can be achieved by taking a sample, and subsequent analysis of the sample - normally applying chemical or physico-chemical methods - is called 'destructive analysis' (DA). If the measurement of an entire item or batch is carried out using radiometric techniques we speak about 'non-destructive analysis' (NDA) as the analysis does not introduce a significant change to the item.

When samples of nuclear material were first analysed for accountancy and safeguards purposes, chemical methods were mostly used. The continuous developments and progress in radiometric techniques promoted the introduction of these techniques in verification analysis. Destructive Analysis was essentially applied when results of higher accuracy were desired. However, with the continuous improvement of non-destructive techniques, the number samples subjected to DA measurement has decreased although balanced to a large extent by an increase in the variety of the types of samples.

The scientific and technical questions associated with this essential safeguards task has lead to establishing dedicated working groups within ESARDA. The NDA working group deals with the questions related to non-destructive analysis and the DA working group addresses questions related to destructive analysis. The latter working group acts as a forum for the exchange of information on DA methods. This intense exchange of experience enables the group to form a unique pool of technical competence and to provide the Safeguards Community with expert advice on destructive analysis methods and procedures and their capabilities and applications in Nuclear Material Accounting and Safeguards.

Working Method

The scope of activities of the DA working group is defined in the Terms of Reference [2]. They are reviewed if and when the need arises. At present the following tasks have been identified:

1. Maintain a list of DA methods suitable for accountancy and verification purposes.

2. Determine the reliability of DA methods by continuing inter-laboratory measurement evaluation programmes. In particular the Group will:

   (a) identify analytical problems of common interest,
   (b) define the objectives and the design of the programmes,
   (c) evaluate the results,
   (d) consider formal QA requirements.

3. Promulgate Performance Values for Uncertainties in DA methods of nuclear material.

4. Recommend Target Values for Uncertainties in measurements in nuclear material, and promote international Target Values.

5. Recommend the implementation of new and improved methods.

6. Promote the systematic and correct use of Reference Materials by:

   (a) maintaining a list of Reference Materials available within the European Union and elsewhere,
   (b) maintaining information on their traceability,
   (c) advising on their distribution and transport problems.

7. Consider sampling problems and their significance for DA results (including sample treatment, conditioning, storage, transport, stability).

8. Promote the use of correct and internationally accepted terminology in measurements.

9. Support new developments of DA methods for Safeguard purposes.

10. Promote co-operation with other Working Groups.

In order to implement these tasks, the group establishes an action plan which typically covers a period of 2-3 years. The current action plan addresses the presently most important safeguards challenges with relevance to destructive analysis. The action plan includes the following items:

1. Promote the **improvement of DA measurement quality** for nuclear material accountancy and control purposes.
   This is achieved by
   - stimulating exchange of experiences between laboratories,
   - comparing different chemical and radiometric analytical methods,

- identifying ways to reduce waste from DA procedures,
- regularly reviewing external quality control programmes,
- encouraging participation in external QC programs and providing direct feedback to their organizers.

2. Address measurement problems arising from **new challenges in safeguards** and in related areas.
   The group therefore keeps abreast of the methodologies in nuclear forensic analysis, supports the development of methods for the measurement of alternative nuclear materials and follows the development of advanced fuel cycles with focus on new types of samples.
3. Support activities for the development and improvement of methods for determination of **nuclear signatures in environmental samples** To this end, the group maintains knowledge of recent developments in the area, compares different techniques and approaches, assesses the potential application for safeguards and related areas, oversees the results from external quality control campaigns.

Associated with the action plan is a list of success indicators that are intended as a metric for the progress achieved by the group. The implementation of the action plan is performed through the regular meetings of the working group, through contributions of (assigned) working group members, through topical meetings and workshops or through joint meetings with other working groups.

The success of the working group and the outcome of the discussions depend upon its members and their commitment. The contributors to the work of the DA working group come from the three main groups of actors involved in nuclear safeguards: the safeguards authorities (IAEA and Euratom Safeguards), the plant operators and research (or measurement) laboratories.

Activities and Contributors

In order to develop recommendations and solutions addressing the issues identified in the Action Plan, the Working Group brings experts in the field together at given intervals. This is typically achieved by 1 or 2 annual working group meetings together with specific topical meetings or workshops, that allow participants to discuss particular topics in depth.

Topical Meetings

At the Workshop on Measurement Quality (Geel, 2003), the main emphasis was on the experiences of laboratories in introducing ISO 17025 in the measurement work. This was recognised as a strenuous, quite possibly expensive, exercise, but one that was considered necessary by all the participants. The preparatory work involves writing working documents and there was considerable discussion on the best approaches to this. There was a good cross-section of experience in moving towards ISO 17025 certification among the participants.

A workshop was held on ' Metrology, Accuracy, Sensitivity, Selectivity - Nuclear Safeguards in the 21st Century', (Prague, 2004). In this workshop the future applications of DA analysis and the expectations for controlling the levels of uncertainties in measurements were discussed. In particular the role of reference materials in routine measurements was a topic of great interest. The dependence of measurement result uncertainties on the uncertainties of the reference materials used and the correct application of reference materials in the measurement process were two points looked at in detail.

Two workshops have been held in 2006: a workshop on 'Measurement Uncertainties in Nuclear Measurements', (Geel, 2006) and a Workshop on 'Radiometric Techniques for Screening of Environmental Samples', (Luxembourg, 2006), together with the ESARDA Non-destructive working group. There is a considerable overlap of interests between the two working groups and meetings like the workshop in Luxembourg allow workers active in both fields to come together and discuss areas of interest.

The value of these topical meetings is that it offers a unique opportunity for workers from different backgrounds using DA techniques to discuss methods, methodology and to agree on future needs and developments. By rigorously eliminating political bias and concentrating on the scientific matters an exchange of ideas, opinions and experiences can take place: a truly enriching and rewarding state of affairs.

Contributors

Contributors and regular participants to the Workgroup meetings come from the Safeguards authorities (Euratom Safeguards and IAEA), from industry (British Nuclear Fuels, AREVA, ENEA, ENUSA, URENCO) as well as from many research laboratories (EC-JRC-ITU, EC-JRC-IRMM, NBL, SCK.CEN, VTT Chemical Laboratory, NMCC Japan, NRG Petten, PSI Villigen).
These names are given as examples and the list is not complete as workers from other laboratories participate at workgroup meetings depending on the main topic of interest.

Conclusion

Destructive analytical methods will provide a significant contribution to future safeguards. By constantly improving the capabilities of the measurement techniques and areas of application, the inherent strengths of DA - its accuracy, selectivity, sensitivity and traceability - will remain the basis for future application of DA methods.

Destructive analysis has evolved from the workhorse of traditional safeguards, to being a sophisticated toolbox for addressing the most challenging measurement problems of safeguards in the 21st century.

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Particle-chemical analysis of uranium and plutonium

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Abstract. Uranium and plutonium particles were identified using scanning electron microscopy (SEM) and analyzed using inductively coupled plasma mass spectrometry (ICP-MS) and thermal ionization mass spectrometry (TIMS), following chemical decomposition. After the particles were identified with SEM, they were picked up with a micro-manipulator and transferred onto a piece of pure silicon wafer. The particles on the silicon piece were dissolved with nitric acid and the isotopic composition of U or Pu was measured with ICP-MS and TIMS. The results obtained by both methods showed good agreement with the certified values within the expected uncertainty. The measurement uncertainties obtained in this study were similar for both mass spectrometric methods.

1. Introduction

The analytical methods applied to environmental samples for safeguards can be roughly classified into two types: bulk analysis and particle analysis. The results of bulk analysis provide the average isotope composition of the elements U and Pu in a whole sample. Particle analysis measures individual particles in samples, thus giving more detailed information, although it is not usually possible to identify and analyze all of the particles in a sample. The method used at the IAEA Safeguards Analytical Laboratory (SAL) for analysis of U particles is, up to now, secondary ion mass spectrometry (SIMS). This method can be quite rapid for particle analysis but has the disadvantage that minor isotopes such as ^{234}U and ^{236}U are not measured precisely enough. SIMS also has difficulty measuring Pu isotopics because of isobaric interference from ^{238}U and ^{241}Am. Another possibility for particle analysis is the fission-track method combined with TIMS which is normally more precise than SIMS but is time-consuming and requires extensive experience.

In recent years, more precise, accurate and timely analysis of U, Pu and Am in various types of environmental samples are increasingly requested from the IAEA Clean Laboratory for Safeguards [1]. The Clean Laboratory routinely performs bulk analysis for U and Pu in environmental samples. In 2006, the Clean Laboratory started introducing a chemical-particle analysis method, using the SEM-EDX, the ICP-MS and the TIMS, in cooperation with the CLEAR Laboratory of the Japan Atomic Energy Agency (JAEA), in order to be able to obtain the results of chemical analysis and particle analysis from the same environmental sample.

A systematic study was performed by Pöllänen et al in 2006 [2] to investigate the radioactive particles coming from a nuclear weapons accident using the SEM–X-ray analysis; gamma-ray-, alpha-, liquid scintillation spectrometry; SIMS; and the ICP-MS. The quantitative and isotope analysis of U, Pu and Am in soil samples from the Palomares, Spain accident site (1966) were performed. The results of isotope measurements obtained with the ICP-MS and SIMS showed, however, significant differences and the authors proposed that this was an analytical artifact or could reflect possible inhomogeneities of isotope composition. They concluded that their study highlighted the importance of using different complementary methods for particle analysis.

In this study, a combination of particle analysis and chemical analysis techniques was demonstrated for particles of standard materials (NBS U500 and NBS 947). It involved picking up particles using a micro-manipulator attached to a SEM, followed by U and Pu measurement with the ICP-MS and TIMS. This combination is now termed 'partcile chemical analysis'.

2. The experiment

2.1. Sample preparation and instruments

2.1.1. Particle sample preparation

The particle samples were transferred onto a polished graphite planchet by the vacuum impactor method [3]. The particles on the planchet were fixed with an eicosan coating, followed by the heating of the planchet to avoid spreading of particles. The chemical treatments were performed in an ISO Class 5 (US Class-100) area in both the CLEAR Laboratory and the IAEA Clean Laboratory.

2.1.2. Scanning electron microscope and manipulator

The particles on the graphite planchet were observed by a scanning electron microscope (JEOL JSM-6700 F) in the CLEAR Laboratory. The particle pick-up was performed using a manipulator system attached to the scanning electron microscope. The planchet with particles was held in the SEM stage along with nine silicon pieces (see Figure 1) and each particle was transferred from the planchet onto a separate silicon piece [4]. One silicon piece was used for the blank.

The primary electron accelerating voltage for the analysis of morphology with SEM was 7 kV and 20 kV for energy-dispersive X-ray analysis (EDX). The acquisition time for EDX analysis was 300 s.

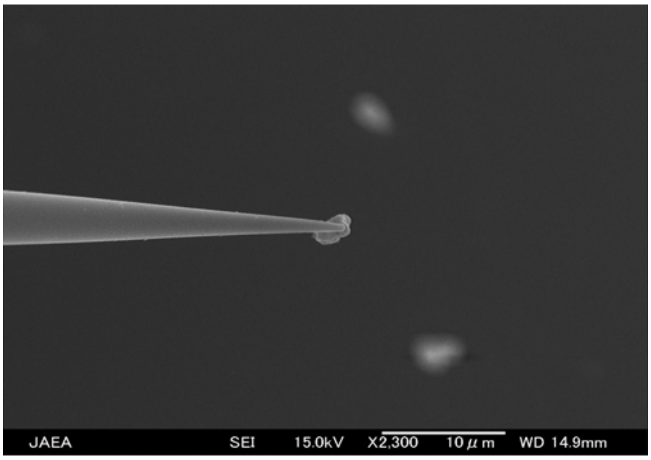


FIG. 1. Picking up a uranium particle with a quartz needle coated with evaporated gold and carbon.

2.1.3. ICP-MS measurements

The ICP-MS instrument used in this study was a Thermo Electron Element with double focusing magnetic sector field. Measurement conditions were: a) plasma RF power: 1148 W, b) sample gas flow: 0.98 – 1.05 L / min, c) sample uptake rate: 0.17 – 0.22 L / min, d) sampling time per isotope: 50 ms, e) scans per replicate: 400, and f) number of replicates: 5 and 7) resolution (M / ΔM): 300. [5].

The correction of mass bias was calculated from NBL U-500 (160 ppt solution) measurements using the following equation.

$R_t = R_m\ (1+C)^{\delta M}$,

where

Rt is the cerified value;

Rm is the average isotopic ratio measured;

C is the coefficient of mass bias; and

δM is the difference in mass, e.g. δM of U235-U238 is 3.

The calculated mass bias was -0.2% smaller than the certified value and was within the validation of the measurements.

2.1.4. *Thermal ionization mass spectrometry*

The isotope ratios of U and Pu were measured by a ThermoElectron Triton instrument using the ion counting detection mode and peak jump measurements. The sample was dissolved in 1 µl of 3M HNO_3 and loaded on a rhenium single-filament along with graphite.

2.2. *Chemical treatment of particles*

The chemical treatment was performed in an ISO Class 5 (US Class-100) area in both the CLEAR Laboratory and the IAEA Clean Laboratory. High purity nitric acid (Seastar Chemicals Inc.) and 18.2 MΩ (Emilli-Q Element) of de-ionized water were used for the chemical treatments.

2.2.1. *For ICP-MS measurements*

The particle sample on a silicon piece was put in a teflon vial to which was added 0.25 ml of 40% HNO_3 solution and then put in a microwave decomposition device. Thereafter, 4.75 ml of de-ionized water was added to the sample solution for the ICP-MS measurement. The chemical treatment was performed in an ISO Class 5 area in the CLEAR Laboratory.

2.2.2. *For TIMS measurements*

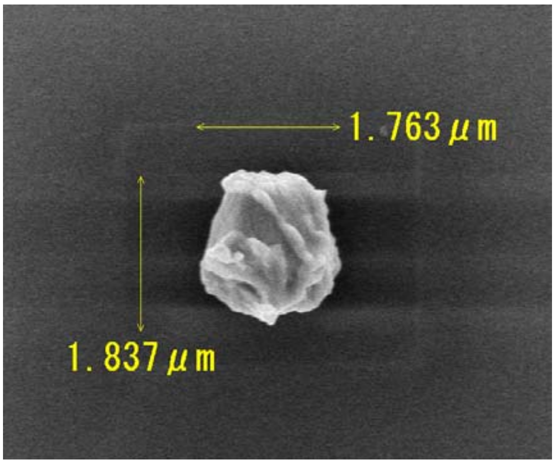
High purity nitric acid (Seastar Chemicals Inc.) and 18.2 MΩ (Emilli-Q Element) of de-ionized water were used for the chemical treatment. The particle sample on a silicon piece was put in a teflon vial and a 0.1 ml of 15M HNO_3 was added and heated until the solution was dried. The dried sample was re-dissolved and transferred onto a Re filament with 1 µl of 3M HNO_3 applied twice.

3. Results and discussion

3.1. *Results for uranium (NBS 500)*

3.1.1. *Observation and analysis of uranium particle with SEM and EDX*

Figures 2 and 3 show, respectively, the image and the EDX spectrum for particle NBS U500-02 (uranium oxide U_3O_8), which was later analyzed with TIMS (see Table 1). The particle dimensions were 1.84 x 1.76 µm. The EDX spectrum showed four M series X-ray fluorescence lines of U together with a silicon line coming from the Si substrate. No other obvious peak was observed except for carbon and oxygen.



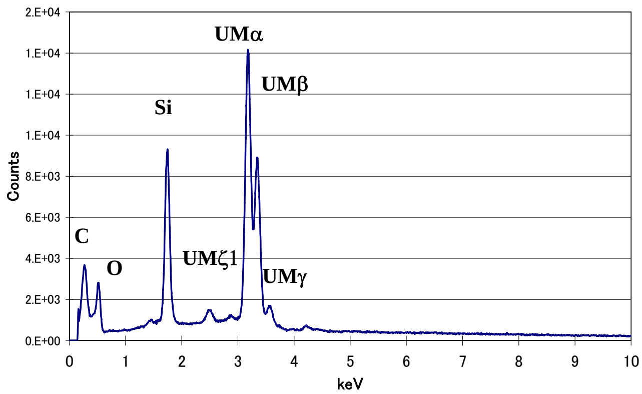


FIG. 2. SEM image of uranium particle (NBS U500 / 02; measurement by TIMS is shown in Table 1) on silicon piece.

FIG. 3. EDX spectrum of uranium particle (NBS U500-02 TIMS in Table 1) on silicon piece. The peaks of uranium appear at 2.48keV (Mζ1) 3.18keV (Mα), 3.34keV (Mβ), 3.55keV (Mγ).

3.1.2. Results of isotopic ratios of uranium measured with ICP-MS and TIMS

The results of isotopic ratios measured by the ICP-MS and TIMS are shown in Table 1. The results of the isotope ratios (236U/238U, 235U/238U and 234U/238U) from eight particles of NBS U500 with diameters of a few µm's measured by a sector field ICP-MS are in good agreement with the certified values within the stated uncertainty. The standard deviations ranged from a few percent to thirty percent in these measurements. The uncertainty improved as a function of the size of the particle. The standard deviation for the largest particle showed 0.8 %, 0.4 % and 1.3 % for the measurements of 234U/238U, 235U/238U and 236U/238U, respectively. The precision of the measurements obtained with TIMS showed almost the same performance as the ICP-MS and the results were also in good agreements with the certified values. The maximum standard deviations of 234U/238U and 236U/238U for CRM U010 measured with SIMS were reported as 13.1% and 8.9%, respectively [4]. These results indicate, for example, that more precise results can be obtained with ICP-MS and TIMS measurements than with SIMS.

Table 1. The results of isotope ratio of uranium (NBS U500) measured by ICP-MS and TIMS.

Sample name	Particle size (μm)	$^{234}U/^{238}U$	±	$^{235}U/^{238}U$	±	$^{236}U/^{238}U$	±
U500-A (ICP-MS)	2.42 × 3.63	0.0104	0.0003	0.996	0.004	0.00149	0.0002
U500-D (ICP-MS)	3.20 × 3.40	0.0105	0.0003	1.003	0.005	0.00151	0.00009
U500-E (ICP-MS)	3.00 × 3.63	0.0102	0.0002	0.997	0.004	0.00155	0.00008
U500-F (ICP-MS)	3.09 × 3.76	0.0105	0.0001	0.998	0.005	0.00151	0.00004
U500-B (ICP-MS)	3.51 × 5.38	0.0106	0.0002	1.001	0.003	0.00147	0.00004
U500-I (ICP-MS)	4.90 × 5.14	0.0105	0.0001	0.999	0.004	0.00148	0.00006
U500-H (ICP-MS)	5.21 × 5.30	0.0104	0.0001	1.000	0.003	0.00152	0.00003
U500-C (ICP-MS)	7.35 × 8.10	0.0104	0.00008	0.999	0.004	0.00151	0.00002
U500-01 (TIMS)	1.76 × 1.84	0.0105	0.0001	0.998	0.004	0.0017	0.0001
U500-02 (TIMS)	2.21 × 3.47	0.0104	0.0003	0.985	0.008	0.0015	0.0001
U500-09 (TIMS)	2.39 × 3.91	0.01046	0.00006	0.997	0.002	0.00153	0.00003
		Average	**[a]STD**	**Average**	**STD**	**Average**	**STD**
ICP-MS		**0.01044**	**0.00012**	**0.999**	**0.002**	**0.00151**	**0.00003**
TIMS		**0.01045**	**0.00005**	**0.993**	**0.007**	**0.00157**	**0.00011**
Certified value		**0.01042**	**0.00002**	**1.000**	**0.001**	**0.00152**	**0.00001**

[a] STD: Standard deviation.

3.2. Results for plutonium (NBS 947)

3.2.1. Observation and analysis of plutonium sulfate particle with SEM and EDX

Figures 4 and 5 show, respectively, the image and the EDX spectrum for particle NBS 947-1 (plutonium sulfate tetrahydrate), dimentions 5.53 x 4.05 μm, with subsequent isotope measurements by TIMS (see Table 2). The EDX spectrum showed three M series lines for Pu and no other obvious peak was observed except oxygen, silicon from the substrate, and sulfur which originates in the composition of the particles (plutonium sulfate).

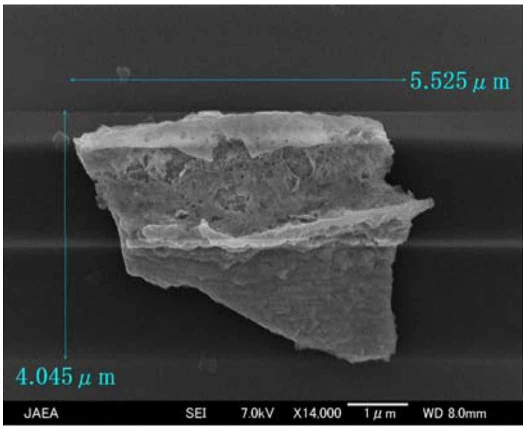


FIG. 4. SEM image of NBS 947-1 TIMS on Si tip.

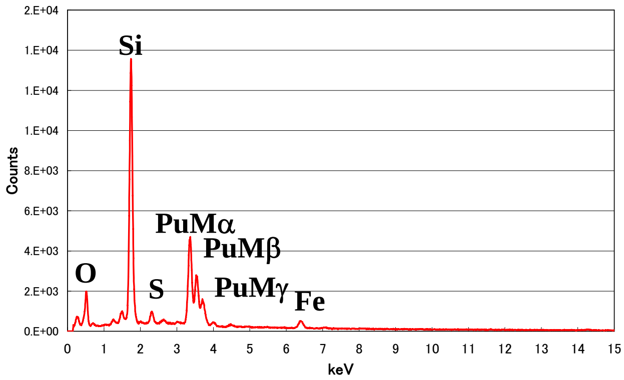


FIG. 5. EDX spectrum of NBS947-1 TIMS on Si piece. Peaks result from plutonium appear at 3.34keV (Mα), 3.37keV (Mβ), 3.53keV (Mγ).

3.2.2. Results of isotopic measurements of plutonium (NBS 947) measured with ICP-MS and TIMS

The results of Pu isotope rario measurements are shown in Table 2. Similar to the results for U particle analysis, the Pu isotope measurements show precise and accurate results. The intensity of ^{241}Pu equals the sum of ^{241}Pu and ^{241}Am, its decay product. Currently a chemical separation method is being developed which would remove Am, thus allowing an accurate measurement of the ^{241}Pu abundance and an estimation of the age of the Pu based on a measured Am/Pu ratio by EDX or WDX in the SEM.

Table 2. The results of isotope ratio of plutonium (NBS 947) measured by TIMS and ICP-MS.

Sample name	Particle size (μm)	^{240}Pu/^{239}Pu	±	^{241}Pu/^{239}Pu	±	^{242}Pu/^{239}Pu	±
NBS947-A (TIMS)	5.53 × 4.05	0.2412	0.0009	0.0113	0.0001	0.0156	0.0001
NBS947-F (TIMS)	3.34 × 1.10	0.243	0.002	0.0117	0.0001	0.0151	0.00009
NBS947-H (TIMS)	4.56 × 1.53	0.240	0.006	0.0120	0.0003	0.0159	0.0009
NBS947 (ICP-MS)	1.60 × 1.39	0.242	0.005	0.0124	0.001	0.0154	0.0008
		Average	**[a]STD**	**Average**	**STD**	**Average**	**STD**
TIMS		**0.2416**	**0.0013**	**0.0119**	**0.0005**	**0.0155**	**0.0003**
[b]Certified value		**0.2409**	**0.0003**	**0.01113**	**0.00008**	**0.0156**	**0.0005**

[a] STD: Standard deviation.

[b] The certified value was corrected to August 2006.

4. Conclusions

The applicability of particle-chemical analysis using U- and Pu particles from standard reference materials has been demonstrated. The particle pick up method was succesfully established and the agreement of the results with the certified values are encouraging. The standard deviation of mass spectrometric data based primarily on counting statistics was less than 2 % for U isotope ratios and 3 % for Pu isotope ratios. The combination of SEM-EDX, chemical treatment and mass spectrometric analysis provide reliable and detailed information on individual particles. More precise measurement of particle size, combined with quanitiative analysis of U and Pu, would allow for estimating the mass and density of particles. The authors plan to apply the methods presented in this paper to environmental samples derived from safeguards inspections.

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Importance of the impurity spectrum in nuclear materials for nuclear safeguards

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Abstract. Quantitative measurements of plutonium and uranium represent established and indispensable key measures for the assessment and verification of the nuclear material balance in nuclear materials safeguards. The required measurements are traditionally performed with high accuracy by means of conventional non destructive and destructive analytical techniques. With the introduction of new highly sensitive techniques in nuclear analytical laboratories allowing reliable measurements of trace elements in nuclear material samples, additional verification techniques now become feasible and applicable in nuclear safeguards.

The spectrum of impurities contained in any nuclear material represents a memory of its history and can be regarded as a fingerprint. For example, uranium materials at different production or reprocessing steps are in a contact with different chemical and physical media. These will inevitably leave their signature in the form of impurities in the nuclear material, thereby providing hints to their origin and sites of production.

In this paper we present results of impurity measurements in different types of uranium materials (Yellow cake, and UF_6), illustrating the significant variation in the respective spectrum of impurities.

We will illustrate the possible origin of the impurities present in the various types of uranium samples. Next a brief description is given on how the impurities are measured and finally we will show some practical examples of the analysis of the impurity spectra. The information, if adequately interpreted and analysed, can provide useful input data for more efficient and extended verification schemes in nuclear safeguards.

Introduction

With the introduction of highly sensitive techniques in nuclear analytical laboratories allowing reliable measurements of multiple trace elements in nuclear material samples, new classes of information now become routinely available. These measurements are important for progress in many R & D experiment-based projects in the nuclear field such as fuel fabrication, partitioning and transmutation, waste management, basic actinide research and nuclear safeguards.

Nuclear safeguards are presently focused on the assessment and verification of the nuclear material balance in nuclear installations. Traditionally this is done by measuring isotopic composition, concentration, physical properties and enrichment of such materials. The additional parameter, the trace element spectrum, provides additional information for samples characterisation. The combination of all characteristics can help to confirm that declared processes are used, purity levels are consistent with the processes used, compounds produced at one facility and used at another are the same and only declared materials are present at a facility.

The possible origin of the Impurities

To better understand the origin of elemental impurities is helpful to understand the steps involved in the production of uranium enrichment base material starting from uranium ore.

Uranium materials at different stages of production or reprocessing are in a contact with different chemical and physical media. These will inevitably leave their signature in the form of impurities in the nuclear material. The path from ore to enrichment base material essentially uses a three stage refining and conversion process (see Fig. 1-3) [1].

The mixed uranium ore concentrate is dissolved in nitric acid. The solution obtained is impure uranyl nitrate UO_2 $(NO_3)_2$. If necessary, the uranyl nitrate is filtered. The solution of uranium nitrate $UO_2(NO_3)_2 6H_2O$ is then fed into a counter current solvent extraction process, using tributylphosphate dissolved in kerosene or dodecane. The uranium is collected by the organic extractant, from which it can be washed out by a dilute nitric acid solution affter which it is concentrated by evaporation. The solution obtained is relatively pure uranyl nitrate. Then liquid is calcined (heated strongly) to produce UO_3 [2].

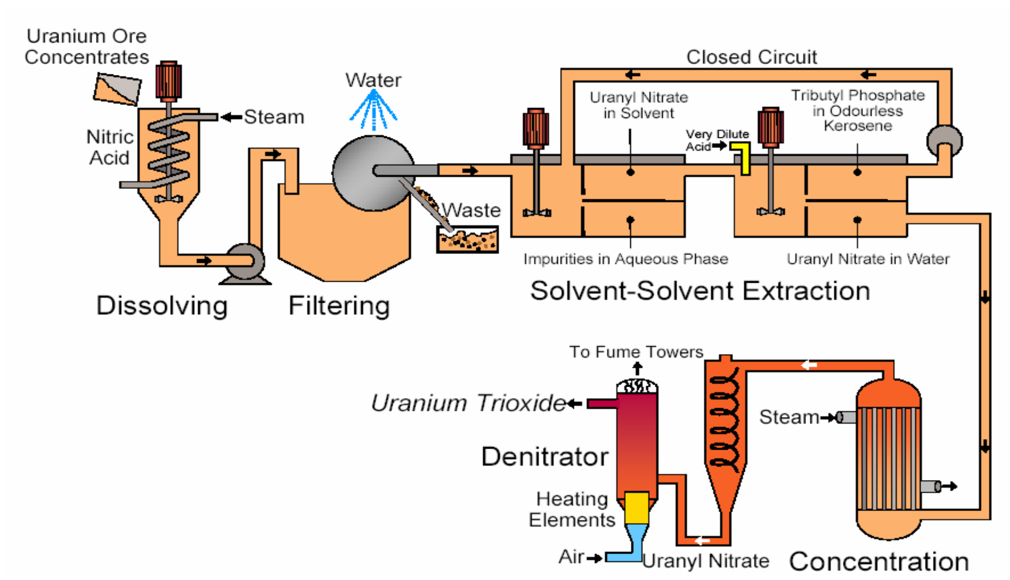

Figure 1. Conversion of uranium ore concentrates to uranium trioxide

The UO_3 is then reduced to UO_2 in a kiln using hydrogen.

$$UO_3 + H_2 \rightarrow UO_2 + H_2O$$

The reduced oxide then reacts with gaseous hydrogen fluoride in another kiln to form uranium tetrafluoride, UF_4, though in some plants this fluoridisation is performed in a wet process using aqueous HF.

$$UO_2 + 4HF \rightarrow UF_4 + 2H_2O$$

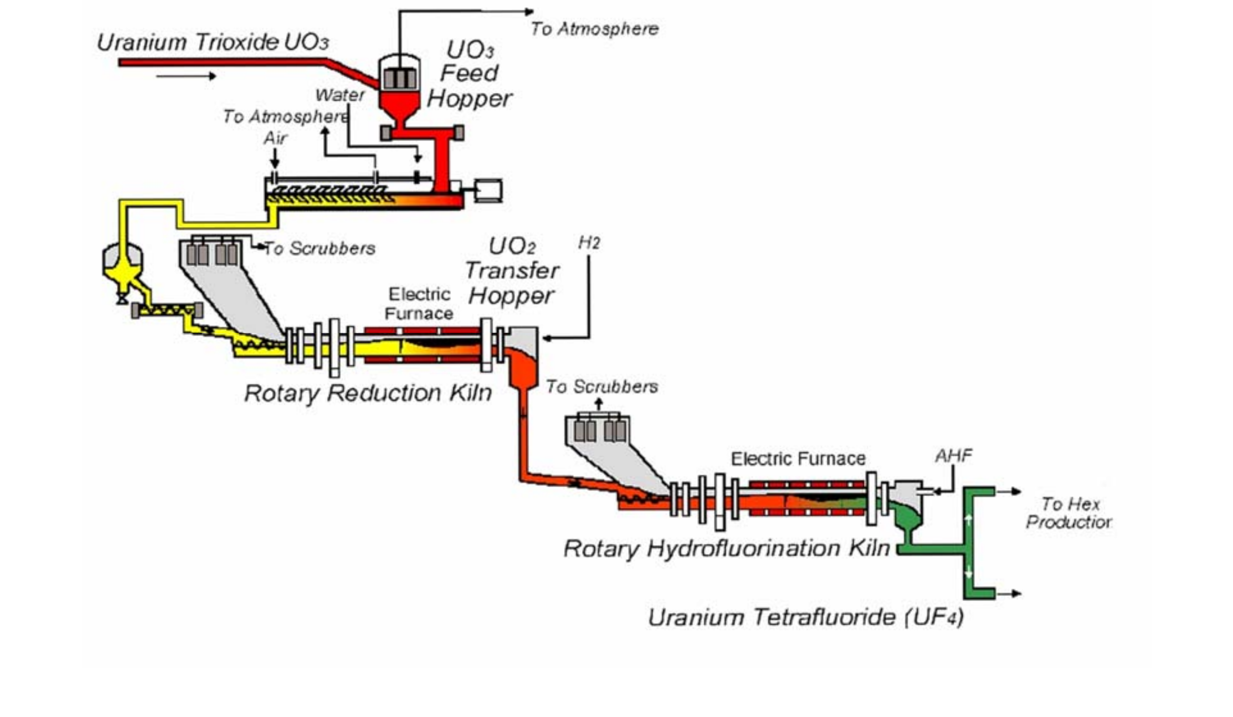

Figure 2. Conversion of uranium trioxide to uranium tetrafluoride

The tetrafluoride is then fed into a fluidised bed reactor with gaseous fluorine to produce uranium hexafluoride, UF_6. The reactor is composed of a long vertical tube in which solid UF_4 spontaneously bursts into flame on contact with gaseous fluorine.

$UF_4 + F_2 \rightarrow UF_6$

The gaseous UF_6 is cooled in crystallizers and then liquified and flows under gravity and pressure into transport containers. It is allowed to crystallize in the container for storage and transportation.

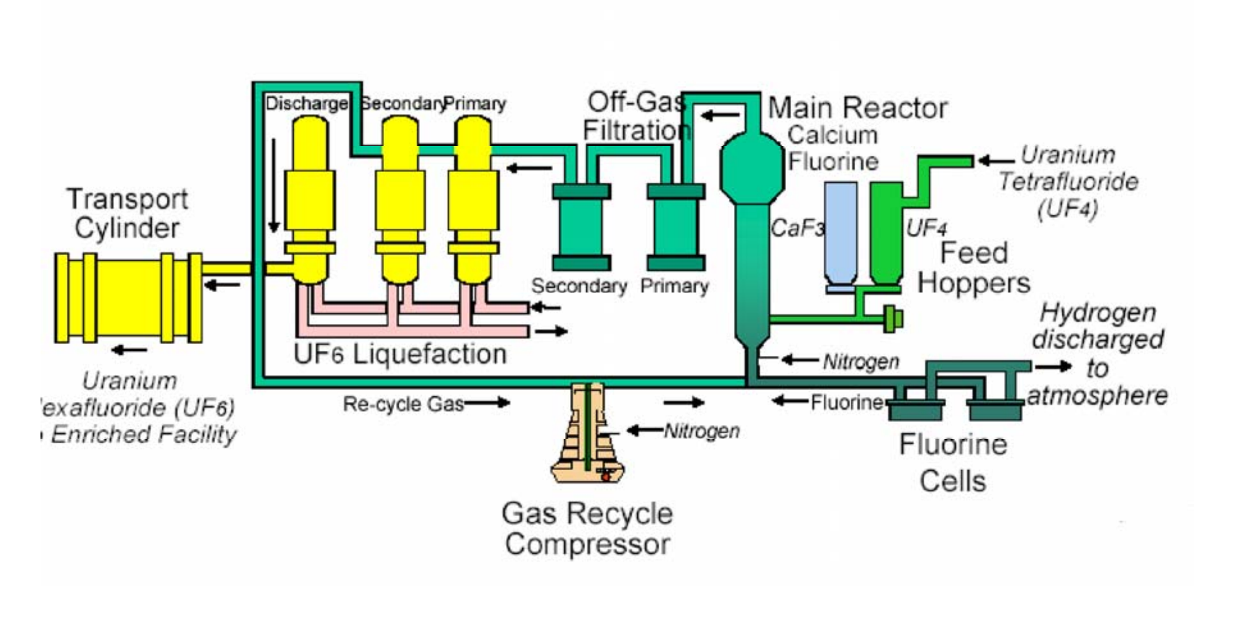

Figure 3. Conversion of uranium tetrafluoride to uranium hexafluoride

It may be apparent that, due to imperfections in the uranium extraction process, trace quantities of the impurities present in the ore may survive the chemical extraction processes to some the degree and that the aggressive conditions of the calcination and fluoridisation steps may add impurities coming from structural components such as the furnaces and from the chemical reagents used. The former impurities might contain information on the origin of the ore while the latter may shed some insight on the specific details of the conversion plant.

The measurements

The accurate determination of trace elements concentrations are performed by an inductively coupled plasma mass spectrometry (ICP-MS) [3]. The equipment used is a ThermoFinnigan Element2 (ThermoFinnigan, Bremen / Germany) installed in a glove box [4].

This ICP-MS comprises a single collector magnetic analyser which permits the entire mass range from mass 4 to mass 300 to be analysed. The machine's detection efficiency for the various elements is routinely established using commercial multi element standards and using home-made standards for the actinides. The detection limits are obviously somewhat element specific but are generally in the ppb range. Clean working practices are adopted and any measurement includes procedural blanks to monitor the impurity levels that arise from the chemical reagents used, from any chemistry steps involved or from any other sample preparation steps such as the grinding of ore samples.

Preliminary results and Discussion

For data interpretation various statistical techniques such as correlation or cluster analysis are used to better understand the multivariate data.

First of all we started with correlation analysis since this is a common form of statistical analysis. Pearson correlation coefficients are established by dividing the covariance of the impurity spectrum of two paired samples by the product of their standard deviations. This statistical technique can show whether and how strongly these sample pairs are related. Data standardization is performed before the correlation coefficient is calculated. This transformation is used to bring all values to compatible units and makes the distribution of values easier to compare across samples.

Results showed high correlation coefficients between some of UF_6 samples which indicates a strong relationship which can not be due to random factors. Although its looks very promising and correlation coefficients are simple to calculate, they are not always robust and require careful vetting of each sample pair before a conclusion can be drawn. This is a tedious process because the number of pairs becomes prohibitively large even for a modest number of samples.

Another important aspect in establishing correlations is that one should not ignore the similarity in chemical behavior between certain elements. Possibly similar chemical elements should be treated as a group for finding more meaningful correlation coefficients. Such groupings might be suggested by so-called cluster analysis. Clustering is classifying similar objects into different groups, or more precisely, the partitioning of a data set into clusters, so that the data in each subset share some common trait. The dendrogram in Fig.4 shows the result of a cluster analysis over a set of eleven yellow cake samples of various origins. A dendrogram is a tree in which elements that are present on nearby branches share more common characteristics than those on remote branches. One intrigueing finding shown in Fig. 4 is that, although the statistical cluster analysis does not possess any information on the properties of chemical analogues, it does decide that nine elements belonging to the Lanthanide group shows very similar behavior in all eleven yellow cake samples. A subsequent detailed analysis of these nine elements in the eleven samples shows that the grouping suggested by the cluster analysis is fully justified.

The analysis shown in the form of a dendogram in Fig. 5 is for eleven yellow cake samples for which the origin of the material in known to us. The analysis includes all available trace element data for all samples and no attempt was made to pregroup any elements.

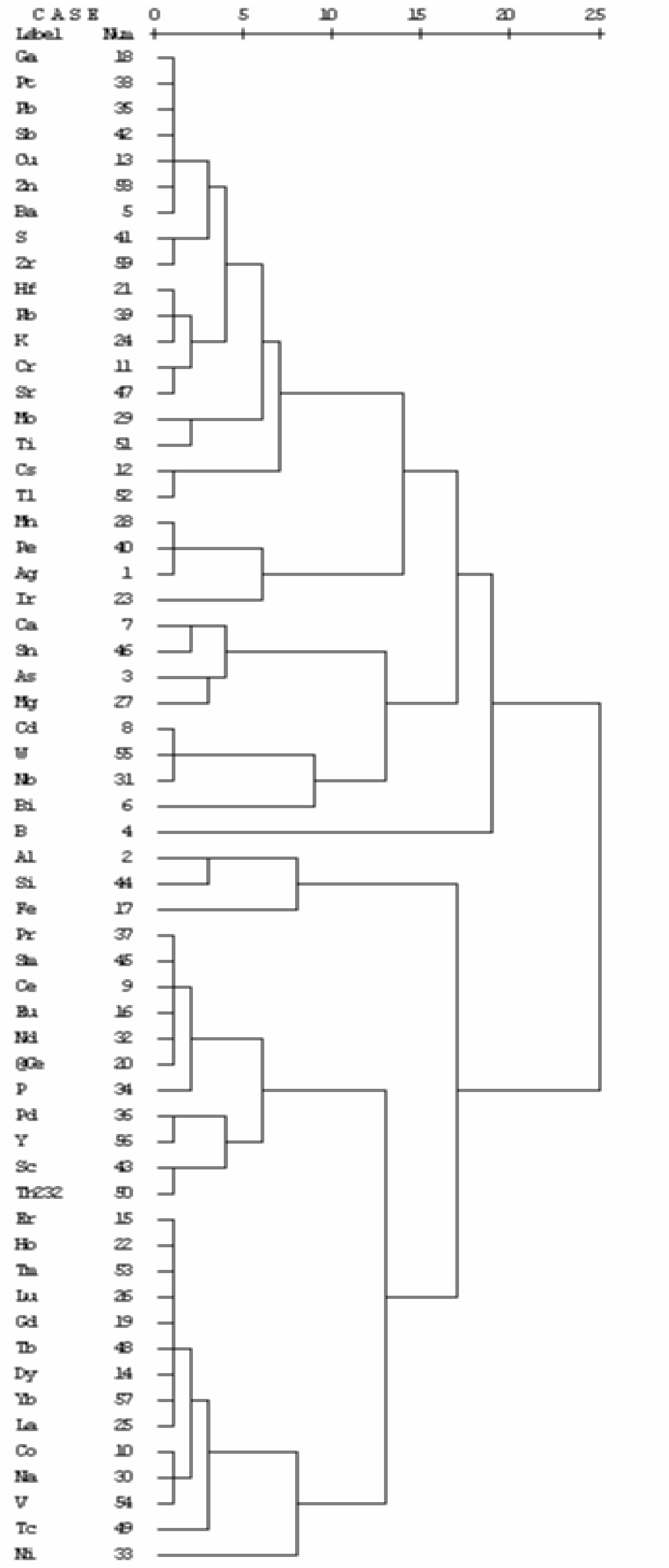

Figure 4. The dendrogram of trace elements in the yellow cake samples

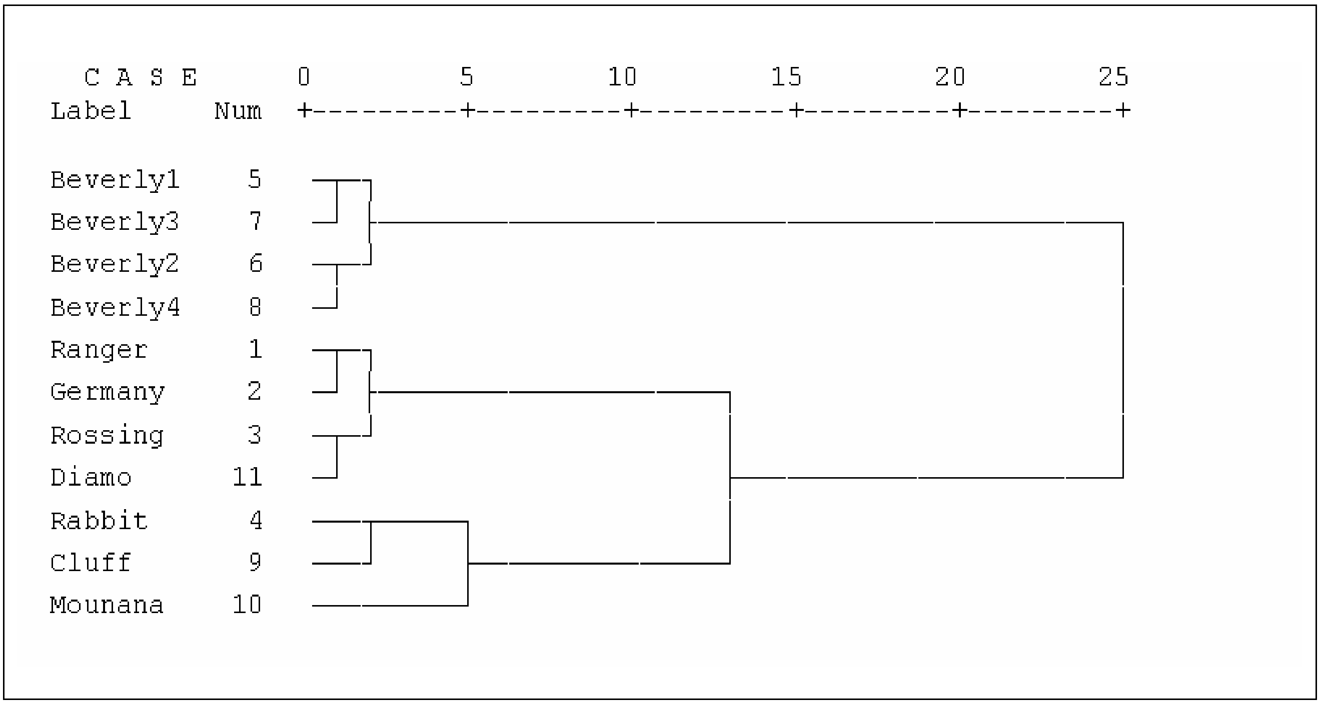

Figure 5. The dendrogram of yellow cake samples

The cluster analysis correctly identified that samples Beverly1 and Beverly3 are similar, that Beverly2 and Beverly4 are similar and that the entire Beverly grouping (1, 3, 2 and 4) are very similar amongst themselves but distinctly different from any of the samples with a different origin.

The same approach was used in UF_6 samples but here unfortunately we do not have any information on the origin of the material so that any grouping suggested by the cluster analysis can not be confirmed. The result is presented in Fig. 6. The dendrogram shows that samples 15 and 16 form a closely related group as do samples 4 and 5. This could indicate that these samples are coming from the same place or shares the same chemical process. Validation of these clusters can only be done when a data base on nuclear materials with known origin is available.

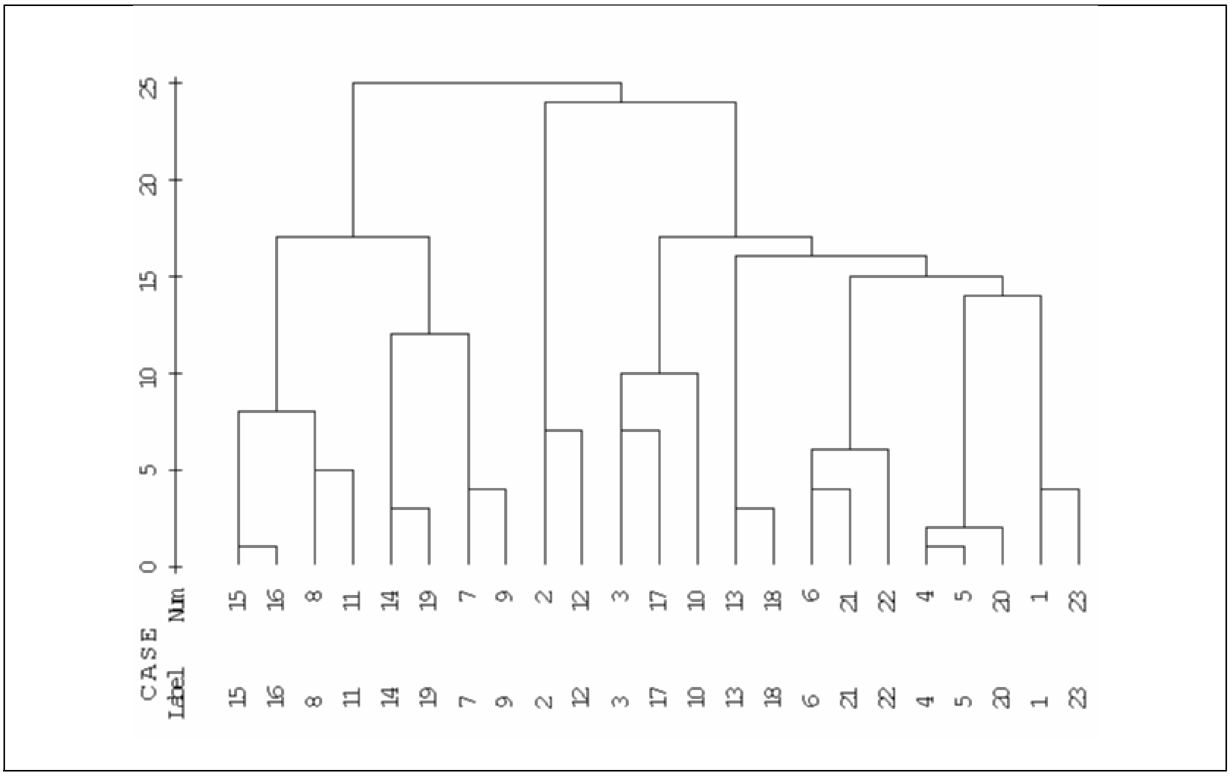

Figure 6. The dendrogram of UF_6 samples

Conclusions

We hopefully demonstrated the potential usefulness of trace impurity analysis as a means of further characterising nuclear samples material.

We are presently trying to establish statistical tests to determine whether different samples share a common origin and / or were processed in the same uranium conversion plant.

For those samples for which the origin is known we have demonstrated that cluster analysis appears to be a promising tool in identifying samples of common origin. It also became clear that more effort is needed to develop robust data analysis tools. The first results are however sufficiently encouraging to justify the additional effort.

ACKNOWLEDGEMENTS

The support of the Australian Safeguards and Non-Proliferation Office (ASNO) in providing the samples from the Australian mines is highly appreciated. We are equally grateful to the IAEA for supplying some of their yellow cake samples.

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Validation of uranium determination by ICP-SMS from QC samples from the IAEA Safeguards Analytical Laboratory

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Abstract. The IAEA Safeguards Analytical Laboratory (SAL) recently installed a high-resolution inductively coupled plasma sector-field mass spectrometer (ICP-SMS) for determination of uranium concentration down to environmental levels, with the goal of enhancing the efficiency of the analysis of quality control samples at the IAEA Clean Laboratory for Safeguards. The most challenging samples from the Clean Laboratory are room blanks, which commonly have uranium contents of 0.1 to 5 pg. The location of the instrument in the nuclear area of SAL poses challenges regarding the realistic lower working detection and quantification limits of uranium. A systematic experimental study was implemented to validate the performance of the ICP-SMS under routine operational conditions. The results demonstrate that reliable, precise determinations of uranium can be made at concentrations of ≥200 ppq with a standard uncertainty down to ~1%, assuming proper precautions against contamination are taken and the unknown sample is well bounded by fresh calibration curves. At this performance level, the ICP-SMS at SAL is capable of reliably detecting and measuring uranium for the purposes of the Clean Laboratory.

1. Introduction

The IAEA Clean Laboratory for Safeguards routinely analyses uranium and plutonium in environmental inspection samples for verification purposes. In parallel, a variety of control samples such as room blanks and chemical process blanks are measured for quality control (QC) purposes to ensure that all safeguards-related work is carried out under clean and controlled conditions. In addition, a number of reference materials are routinely measured to monitor instrument performance. Currently, all analytical work in the Clean Laboratory is performed on a single thermal ionization mass spectrometer (TIMS), located in the Clean Laboratory. The limited capacity of the TIMS (one hour to complete a single filament measurement), in combination with a steady increase of environmental sample requests and time-consuming sample preparation procedures, represents a significant impediment to satisfactory sample throughput.

To improve sample throughput from the Clean Laboratory, a high-resolution inductively coupled plasma sector-field mass spectrometer (ICP-SMS) instrument with a high-sensitivity inlet system and autosampler was installed in the nuclear chemistry area of the SAL in late 2005. The instrument is used for uranium determinations of QC room blanks, process blanks from U-Pu chemical separations and the bulk analysis of U and Pu from inspection samples. Typical content of samples transferred from the Clean Laboratory vary from a few hundred femtograms (fg) to a few hundred nanograms (ng) of uranium and a few fg to a few ng of plutonium.

The objective of this study is to demonstrate the stability and reproducibility of uranium assay measurements by the ICP-SMS across a relevant concentration range. In principle, the same approach is feasible for plutonium and will be the focus of forthcoming efforts. As an instrument with high sample throughput, adequate detection limits and good precision and accuracy, the ICP-SMS is envisioned to take a decisive role in SAL's extensive analytical programme for safeguards purposes.

2. The experiment

2.1. Analytical method set-up

Because of space limitations, the ICP-SMS is located in the nuclear chemistry area of SAL. Due to the relatively high uranium background in this area, it was necessary to determine the signal-to-noise threshold of the instrument for reliable and repeatable uranium determinations in the ppq to ppt concentration range (ppq = 1 : 10^{15}, ppt = 1 : 10^{12}). The most challenging measurements to make are from the Clean Laboratory room blank samples with a total uranium content between 0.1 to 5 picograms. Assuming the uranium content is dissolved in a volume of 0.5 mL 5% HNO_3, the target performance range for the instrument is therefore between 200 ppq and 10 ppt.

All measurements were carried out with an Element2 (ThermoElectron, Bremen, Germany) ICP-SMS equipped with a guard electrode to eliminate secondary discharge in the plasma and to enhance overall sensitivity. A microvolume autosampler and a microflow PFA nebulizer (both from ESI, Inc., Omaha, USA) were employed to transport sample solution into the plasma of the ICP-SMS. Details of the ICP-SMS operating conditions and the data acquisition parameters are listed in Table 1.

Table 1. Operational conditions of the ICP-SMS.

Feature	Condition
Forward power	1200 W
Coolant gas flow rate	16 L/min
Auxiliary gas flow rate	0.8 L/min
Sample gas flow rate	1.0 L/min
Sample uptake rate	150 mL/min
Sample cone	Ni, 1.1 mm aperture
Skimmer cone	Ni, 0.8 mm aperture
Resolution (m/Δm)	LR 400
Scan type	E-scan (electric scan over small mass ranges, constant magnetic field)
Mass window	125
Integration window	60

The experimental work reported required >300 measurements carried out over a five week period. Features that were evaluated include instrument signal stability and consistency of uranium concentration determinations over short and long time frames. The associated standard uncertainties for these low concentrations were observed around 100 ppq. In order to mitigate potential external contamination effects, solutions were always kept in a laminar flow bench and sealed in plastic bags. Exposure time of samples to the lab air was minimized to a few seconds and blanks and uranium solutions up to 100 ppq were replaced more often than solutions of higher concentrations.

Gravimetrical preparation of uranium standard solutions covering the concentration range from 50 ppq (50 pg/L) up to 100 ppt (100 ng/L) was performed in a class 100 clean area of the Clean Labortatory.

High purity water (from a Milli-Q Element system, 18.2 MΩ quality), a uranium standard of natural composition, concentrated ultra pure nitric acid (~70%, twice sub-boiled and distilled in quartz) purchased by Seastar Chemicals and a calibrated balance were utilized. All bottles used for standard preparation were made of PFA. A 2.5% nitric acid solution was used as a blank and for the dilution of the standards.

2.2. *Validation plan*

2.2.1. *Confirmation of manufacturer declared instrument performance*

After installation, acceptance testing of the instrument and set-up of the method, solutions with 50 ppq, 100 ppq, 500 ppq, 1 ppt, and 100 ppt uranium were each measured during one hour, a period of time during which 50 readings can be taken. The standard deviations of these readings, which are a measure for the instrument stability, could be compared against the manufacturer's specifications. This experiment was repeated three times over a three week period.

2.2.2. *Performance characteristics for investigation*

After verifying and confirming adherence to the technical specifications, the validation experiments were planned to demonstrate fitness for purpose with respect to the following parameters [1]:

— Lower bounds of the measurement process, i.e.:
  - Decision threshold, S:
    the minimum instrument response above which one would state to have the analyte detected (with a given error probability α of false detection).
  - Detection limit, L:
    the smallest quantity or concentration of the analyte, that – if present – would be detected (with a given error probability β of non-detection).
  - Quantification limit, Q:
    the quantity or concentration level of the analyte above which meaningful quantitative results can be obtained.

— Working and linear ranges.

— Accuracy, repeatability precision, and reproducibility precision.

Further analytical method performance characteristics, such as selectivity, specificity, recovery, ruggedness, and robustness which depending on the requirements are otherwise also subject to method validations, were not investigated during the present study. They address analytical requirements which, at this stage, are not relevant for the intended application of the method. If the application scope of the method is later extended, this needs to be addressed by further validation experiments and when sufficient experience with the current set-up has been gained.

2.2.3. *Experimental design*

Blank control solutions and uranium reference solutions with the following concentrations were prepared under clean-room conditions: 50 ppq, 100 ppq, 250 ppq, 500 ppq, 750 ppq, 1000 ppq (1 ppt), 25 ppt, 50 ppt, 75 ppt, and 100 ppt uranium. Each control solution was measured once per day. Each reference solution measurement was preceeded by a measurement of the blank control. This set of measurements was repeated on nine different days over a period of five weeks.

3. Results

3.1. *Instrument stability*

Table 2 provides the results of the 50 measurements over one hour on different concentration levels, which were repeated during three weeks. The fluctuations are well within the specified repeatability precision of the instrument.

Table 2. Stability of the ICP-SMS settings (in counts per second, cps).

Week		concentration				
		50 ppq	100 ppq	500 ppq	1 ppt	100 ppt
1	Std.dev.	34	56	69	154	9370
	average	545	715	1701	3953	273192
2	Std.dev.	65	73	120	87	6581
	average	1233	1336	2417	2386	302254
3	Std.dev.	55	79	115	141	9990
	average	944	1308	2282	3759	331527

3.2. Decision threshold, S

The decision threshold S is the lowest value C

$$C = X - B \tag{1}$$

which, when observed, would lead to state that the analyte uranium has been detected. X is the instrument reading for the sample and B is the instrument reading for the preceding blank control. The decision threshold S is calculated by equation (2):

$$\begin{gathered} S = u_{1-\alpha} \cdot \sigma_C \\ \text{or, in the absence of a known value for } \sigma_C : \\ \widehat{S} = t_{1-\alpha\,(one-sided);\nu} \cdot s_C \end{gathered} \tag{2}$$

According to equation (1), the variance of C, σ_C^2 is composed of

$$\sigma_C^2 = \sigma_X^2 + \sigma_B^2 \tag{3}$$

For low concentrations, close to zero, it is safe to assume that the fluctuation of the instrument readings hardly depends on the concentration value itself (the validity of this assumption is shown later). Hence, it is considered constant in the region of interest, and $\sigma_X = \sigma_B$. It follows that S can be estimated from the observed standard deviation s_B of blank control measurements by:

$$\widehat{S} = t_{1-\alpha\,(one-sided);\nu} \cdot s_B \cdot \sqrt{2} \tag{4}$$

The fluctuation of the blank measurements, estimated by s_B, was determined from the average of the sample variances of the 10 blank control measurements done on each day (n = 9 days). It amounts to s_B = 65 cps with ν = 81 degrees of freedom. Applying equation (4), S is then estimated as:

$$\widehat{S} = t_{1-\alpha=0.05\,(one-sided);\nu} \cdot s_B \cdot \sqrt{2} = 1.66 \cdot 65 \cdot \sqrt{2} = 153\,cps \tag{5}$$

In practical terms this means that for any observed blank corrected signal C > 153 cps we would state to have detected uranium, with an error probability α = 5% (that is, a 5% probability of a detection when there is indeed no uranium present in the sample above its level in the blank).

3.3. Detection limit, L

The detection limit L is defined as the lowest concentration of uranium that will be detected with a given probability (1-β) — that is, that concentration of uranium which will yield a blank corrected signal C > S (S = 153 cps) with (1-β)% certainty. We need to calculate, first. the 'true' blank corrected measurement result (i.e. its expected value or mean value), which, when actually measured, yields observation values higher than S in (1- β)% of all cases. This quantity in the signal domain is called L'. Assuming that the distribution of C is approximately normal:

$$L' = S + u_{1-\beta} \cdot \sigma_L \tag{6}$$

Choosing equal error probabilities α = β = 0.05 and following the same rationale as for the decision threshold, namely that the fluctuation of the measurement results is essentially constant for uranium concentrations close to blank levels, the detection limit L' is estimated by:

$$\widehat{L}' = 2 \cdot \widehat{S} = 306\, cps \tag{7}$$

To convert L' into a uranium concentration according to the definition of the detection limit L, we use the average calibration function for the working range < 1000 ppq uranium (as is explained below). The resulting detection limit is thus determined as 175 ppq uranium. The relations between decision threshold S, detection limit in the signal domain L' and detection limit L are shown in Figure 1.

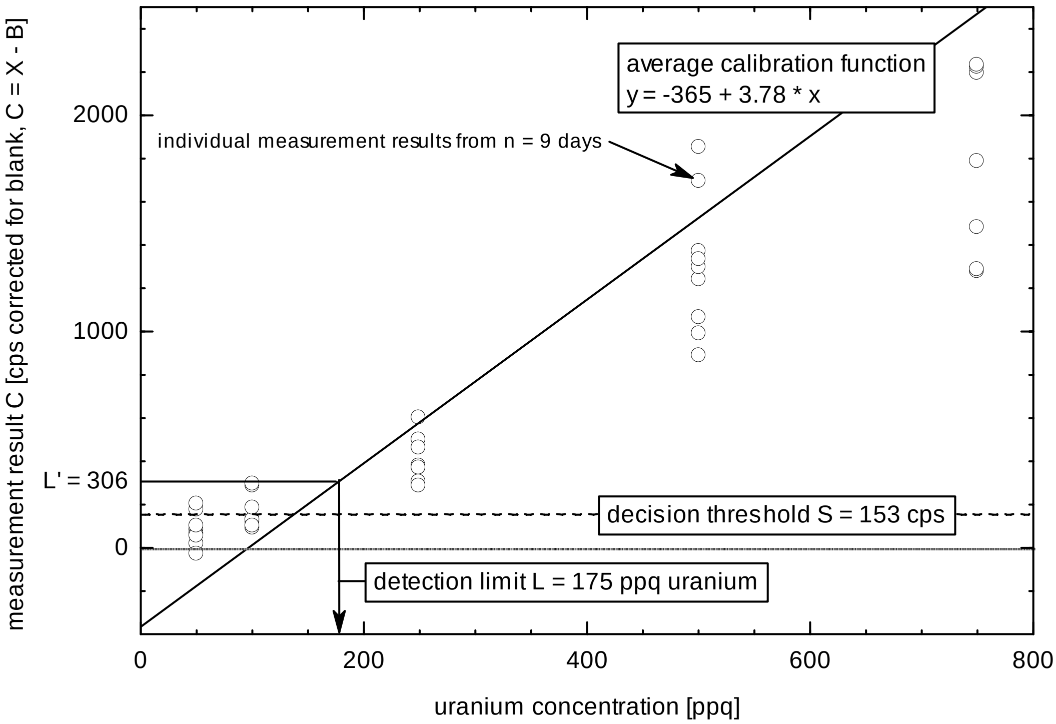

FIG. 1. Decision threshold and detection limit.

3.4. Quantification limit, Q

The quantification limit Q is determined as the lower bound on the working range above which concentrations can be measured with adequately low uncertainty. What is ‘adequate’ depends of course on the intended use of the analytical results. By convention [1], a coefficient of variation of 10% is frequently considered a reasonable threshold. However, for the present application – quality control measurements on room and reagent blanks – a much higher uncertainty is perhaps still acceptable when considering the stochastic nature of room blanks and their natural high sampling uncertainty.

Let equation (8) be the linear calibration function:

$$Y = a + b\,X \tag{8}$$

where Y denotes the observed instrument response in cps and X denotes the corresponding uranium concentration in ppq. The parameters a and b are estimated from the calibration measurements by conventional linear regression analysis.

Concentration values X are calculated from the inverse calibration function[1]:

$$X = \frac{Y - a}{b} \tag{9}$$

The standard uncertainty of X, resulting from the measurement of Y and from the uncertainty of the calibration (i.e. the uncertainty associated with the parameter estimates of a and b), expressed as coefficient of variation [%], is a function of X given by[2]:

$$U = \frac{100}{X} \cdot \frac{\sigma}{|b|} \cdot \sqrt{1 + \frac{1}{N} + \frac{(X - \bar{x})^2}{\sum_{i=1}^{N} (x_i - \bar{x})^2}} \tag{10}$$

where N denotes the number of calibration measurements, x_i denotes the concentrations of the reference solutions used for calibration, and $\bar{x}$ the arithmetic mean of the concentrations of the reference solutions.

The residual standard deviation σ in equation (10) is estimated by:

$$s^2 = \frac{\sum_{i=1}^{N} (y_i - a - b \cdot x_i)^2}{N - 2} \tag{11}$$

The uncertainty U has been calculated for each data point and plotted versus the concentration (Figure 2). It is easily visible that measurements with U < 10% coefficient of variation are possible for

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[1] There are discussions whether a calibration should determine the immediate measurement result (the instrument response) as a function of the quantity (or concentration) of analyte or vice versa. In the first case, one must first invert the calibration function for calculating analyte quantities (or concentrations) from the instrument readings (this is easy for linear functions, but may pose some mathematical difficulty, for example for higher order polynoms). In the latter case, the ‘calibration function’ could be directly used. Contributions to this discussion can be found in [2] and [3]. According to these references, the first approach was adopted as this is the IAEA safeguards standard methodology.

[2] For the case of a single measurement of the quantity Y.

concentrations > 1000 ppq uranium, if the range of concentrations of the reference solutions used for calibration encompasses a sufficient area below and above the concentration of the unknown sample.

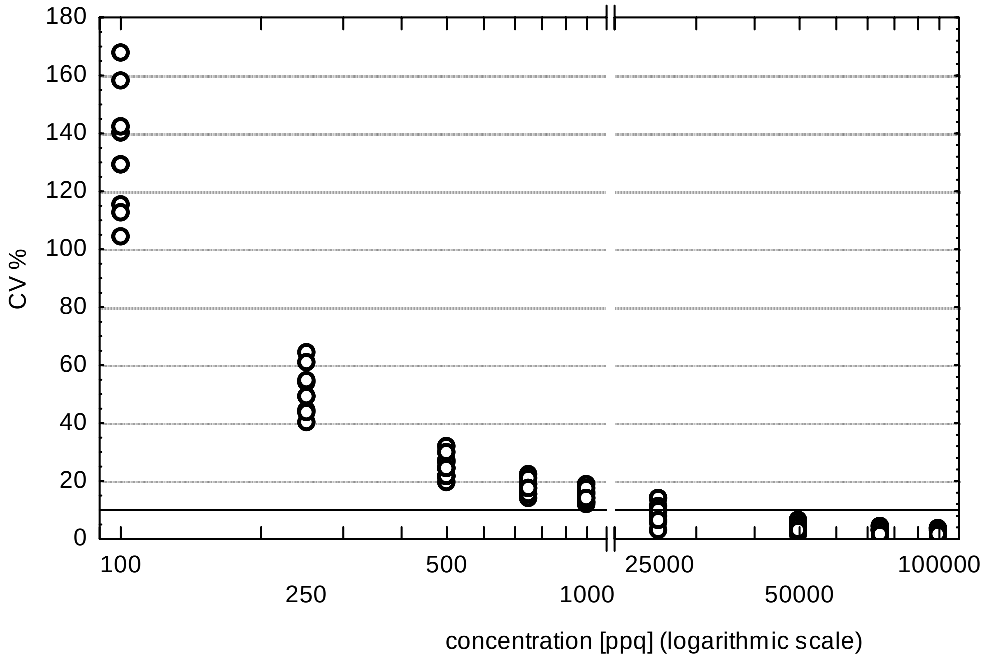

FIG. 2. Uncertainty U [CV%] as function of concentration.

3.5. Uncertainty of measurements

The uncertainty of a particular analysis can be calculated from equations (10) and (11). A visual evaluation of a rescaled version of Figure 2 indicates that the standard uncertainty is typically of the order of 1% - 4% coefficient of variation, given the precautions mentioned above.

3.5.1. Working ranges

Figure 3 shows the measurement results for concentrations in the range 50 to 1000 ppq U. Also indicated are:

— Calibration curve for day 2 obtained from reference solutions with concentrations 50-1000;
— Calibration curve for day 8 obtained from reference solutions with concentrations 50-1000;
— Calibration curve for day 2 obtained from reference solutions with concentrations 25000-100000; and
— Calibration curve for day 8 obtained from reference solutions with concentrations 25000-100000.

The figure demonstrates two immediately obvious observations:

(1) The whole working range must be broken down in separately calibrated sub-ranges: the calibration curves obtained from concentrations in the upper range are obviously unusable for measurements in low ranges.
(2) The instrument needs to be calibrated on a daily basis: slope and intercept (parameters b and a of the calibration curve) obtained on different days differ significantly from each other.

Under these precautions, measurements can be made over a range of three to four orders of magnitude with sufficient precision and accuracy.

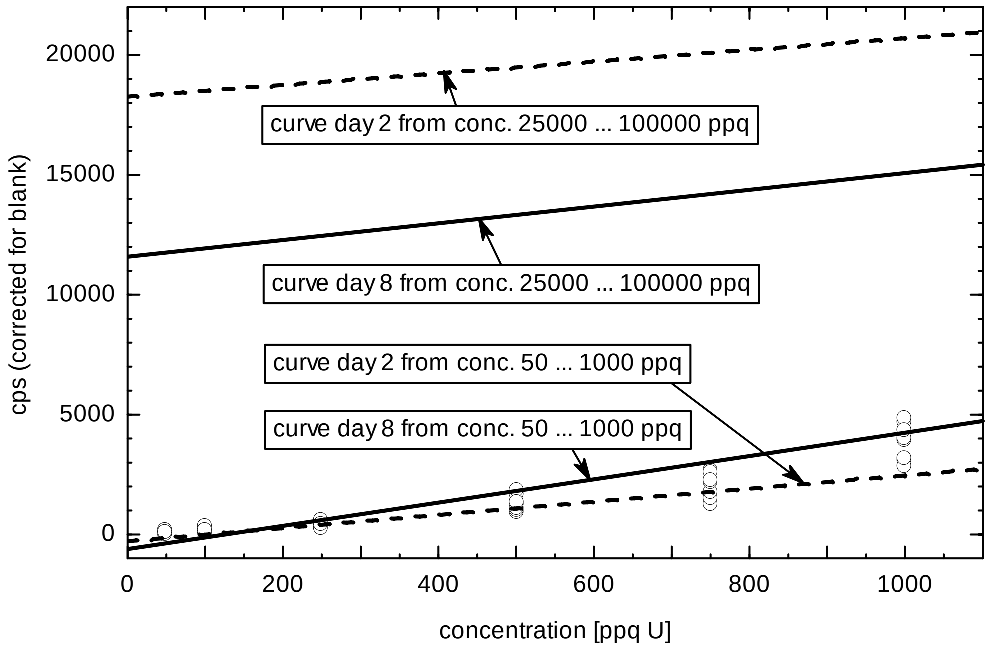

FIG. 3. Calibration measurement results with selected daily calibration curves.

4. Conclusions

The validation study presented in this paper confirms that the new high-resolution ICP-SMS instrument set up at the SAL can be used for the intended purposes. The decision threshold and detection limit for uranium concentration have been experimentally found during this exercise; the detection limit is roughly 200 ppq uranium. For any concentration at or above this limit, uranium will be detected with a probability of 95% or higher. These lower bounds have been determined under realistic routine conditions. Internal tests and instrument specifications indicate that measurements far below these levels should be possible, but this would require an unreasonable effort for guarding against cross-contamination at the current installation location. To perform even lower level measurements with justifiable effort, the whole instrument would need to be operated under clean-room conditions. However, in this case, the instrument could not be used for measurements in the higher concentration ranges.

The instrument needs to be calibrated on a daily basis; influential factors, such as reagent blanks, vary significantly over expanded periods of time and their effect must be taken into account and corrected for on a short-term basis. Sufficiently linear working ranges span over one to two orders of magnitude; the instrument needs to be calibrated separately for each such working range, but the combined working ranges comprise about four orders of magnitude. Under these precautions and with careful selection of the calibration range, uranium determinations are possible with standard uncertainties of the order of 1% (coefficient of variation).

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Optimization of the preparation of certified uranium particles by controlled hydrolysis of UF_6 for nuclear safeguards

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Abstract. Environmental sampling (ES) was implemented by the IAEA in the early 1990's as a safeguards strengthening measure for the detection of undeclared nuclear activities. Uranium particles with an isotopic composition characteristic for the process are collected by swiping surfaces in and around a uranium enrichment facility and are subsequently measured by SEM and SIMS. Since its implementation, ES has proven to be a very sensitive and effective technique. However, there is still a need for reference particles that have characteristics similar to the particles found in swipes taken at uranium enrichment plants.

A new method for the production of uranium particles with certified abundances has been set up at EC-JRC-IRMM. The goal is to realistically reproduce the particles found in an enrichment facility. As the morphology and composition of these particles are most likely influenced by environmental conditions such as temperature, relative humidity and exposure to UV-light during and after formation, the conditions in which the reference particles are produced need to be carefully controlled as well.

In an aerosol deposition chamber, designed and built for the purpose, solid particles were formed by the reaction of a small amount of certified UF_6 with the water in the chamber's atmosphere. The temperature and relative humidity of the air in the chamber were varied to determine the effect on particle composition and morphology. SEM measurements showed that the relative humidity of the air strongly affects the particle morphology.

Improvements were made on the design of the aerosol deposition chamber, based on our experience with the first model, and an improved model was developed and tested recently. The new aerosol deposition chamber allows an easy adjustment of the temperature as well as the humidity of the air inside the chamber.

1. Introduction

Safeguards inspectors of the International Atomic Energy Agency (IAEA) use environmental sampling (ES) for the detection of undeclared nuclear activities. The technique is based on swiping surfaces in and around the enrichment facility and thereby collecting dust material [1]. This dust will contain uranium particles and their isotopic composition is assumed to be characteristic for the process in the facility. The swipes are sent for analysis to the Network of Analytical Laboratories (NWAL). The isotopic composition of the particles is verified by different techniques such as fission track-thermal ionisation mass spectrometry (FT-TIMS) and secondary ion mass spectrometry (SIMS). The procedure for particle analysis using these techniques has been developed by research groups [2][3][4][5][6].

In addition, scanning electron microscopy combined with energy dispersive X-ray spectrometry (SEM-EDX) can be applied to characterize particle sizes and shapes [7] giving complementary information on their history and source [5][8].

It is clear that these particle measurements for safeguards and nuclear non-proliferation control may have major legal or political implications and for this reason they need to be subjected to a careful quality management system.

Starting with a UF_6 material certified for isotopic composition, uranium oxyfluoride particles are produced in an aerosol deposition chamber developed at the Institute for Reference Materials and Measurements (EC-JRC-IRMM). With this instrument UF_6 leaks are simulated on a small scale and uranium oxyfluoride particles can be produced with customized uranium isotopic abundances. The aerosol deposition chamber is a very flexible instrument with a number of adjustable process parameters such as air temperature and relative humidity.

In this work, an overview is given of the results obtained with the first aerosol deposition chamber and the changes that were made to this model to optimize the control of process parameters such as relative humidity and temperature.

2. Materials and Methods

A small amount of a certified UF_6 material, cooled to a temperature between 6 °C and 8 °C, was sampled into a glass vial. This vial was flame-sealed and subsequently placed into the aerosol deposition chamber (Fig. 1). On breaking the vial, the uranium hexafluoride was released and hydrolyzed by the water vapour in the chamber's atmosphere. The reaction between the released uranium hexafluoride and the atmospheric moisture proceeded very rapidly to form uranium oxyfluoride particles and hydrogen fluoride:

$UF_6 + 2H_2O \rightarrow UO_2F_2 + 4HF$

The particles accumulated by aggregation and were finally deposited on polished carbon planchets (Schunk, Belgium) at the base of the chamber. The relative humidity of the air in the chamber was varied between 10 % and 70 %. For the experiments with the new aerosol deposition chamber, both the temperature and humidity of the atmosphere in the chamber were adjusted: the relative humidity varied between 10 % and 70 %; the temperature was set to values between 10 °C and 40 °C (section 3.2).

FIG. 1. Second aerosol deposition chamber

After a collection time of several hours, the carbon planchets were removed from the reaction chamber and were either treated in a furnace at 350 ºC under normal atmosphere for a minimum of 5 hours for stabilization to U_xO_y and removal of inorganic volatile elements and excess of water; or stored in a desiccator without further heat-treatment to avoid water uptake after preparation.

The morphology and composition of the particles produced were measured using SEM-EDX. Most of the SEM-EDX measurements were either carried out using a JEOL-6310 with a LaB_6 filament and Si(Li) EDX detector at the Studie Centrum voor Kernenergie/Centre d'Etude de l'énergie Nucléaire (SCK•CEN, Mol, Belgium) or a JEOL-6300 with a tungsten filament at Antwerp University (Belgium), both at 20 kV electron beam energy. For the detection of light elements such as fluorine, a voltage of 10 kV was applied. Images were recorded in secondary electron mode (SEI) for topography or in backscattered electron mode (BEI) for compositional contrast. To estimate the size distribution of the particles, an automated particle analysis programme developed at Antwerp University combined with Feature Analysis software was used to record and analyze the BEI images and EDX spectra of a 1000 uranium particles. The identification of the particles in the EDX spectrum was based on the presence of the uranium M_α X-ray line at 3.17 keV, the M_β line at 3.34 keV, the L_α line at 13.6 keV, and in some cases the $L_{\beta 2}$ line at 16.4 keV. The images were recorded at a magnification of 1500×, therefore the detection limit (i.e. a minimum of 7 pixels per particle) was 0.46 µm.

3. Results and Discussion

3.1. Effect of humidity

Previous studies indicated that the humidity of the air in the reaction chamber has an effect on the particle morphology [9][10]. To study this effect, small amounts of UF_6 were released in atmospheres having a different relative humidity: 15 %, 43 % and 65 % respectively.

A dry atmosphere in the chamber produced clusters of particles of a few 100 µm composed of many small particles, with an area equivalent diameter of less than 1 µm. These clusters appeared on the planchet together with a background consisting of many small single particles.

An intermediate relative humidity of 43 % led to particles with an irregular shape, with no significant agglomeration.

When the UF_6 was released in air with a high relative humidity (63%), the particles were almost spherical. Moreover, the particles were well separated by at least a few micrometers and the degree of agglomeration was low. The SEM-EDX measurements on almost 1000 uranium particles showed an area equivalent diameter between 0.1 µm and 2.25 µm [11]. The particles might have absorbed water during the process which could explain their spherical shape. A heat treatment of 350 °C for 5 hours did not appear to affect the particle morphology.

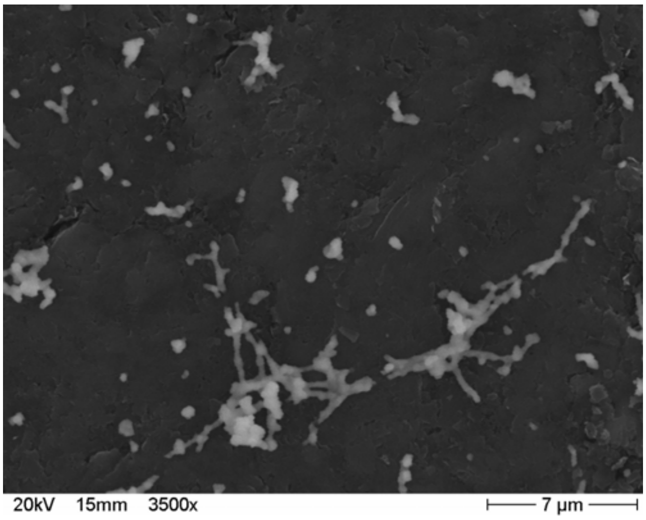

FIG. 2. U particles formed in a relative humidity < 20 %

3.2. Effect of temperature

The temperature in the new model of the aerosol deposition chamber can be set by passing water from a thermostatically controlled water bath through a copper coil on the outside of the chamber. A new improved mechanism to crack the vial has been introduced in which the vial is now cracked by a pin from the top of the chamber. In addition, the chamber has a retractable platform for the planchets, which avoids pieces of glass falling down on the planchets when breaking the vial. Another advantage of this platform is that the planchets can be retracted from the reaction chamber at various time intervals, giving information on the settling rate of the particles after release of UF_6.

Our first experiments with the new aerosol deposition chamber at a relative humidity of 68 % and a temperature of 11 °C resulted in particles with a rather spherical shape, as expected from the high relative humidity, yet the size distribution was noticeably wider than for the experiment with a high relative humidity at ambient air temperature in the original reaction chamber. The largest particles (> 5 µm) showed signs of severe charge-up on the secondary electron images and this was explained by the EDX measurements demonstrating the presence of silicon, probably originating from the vial, in addition to uranium.

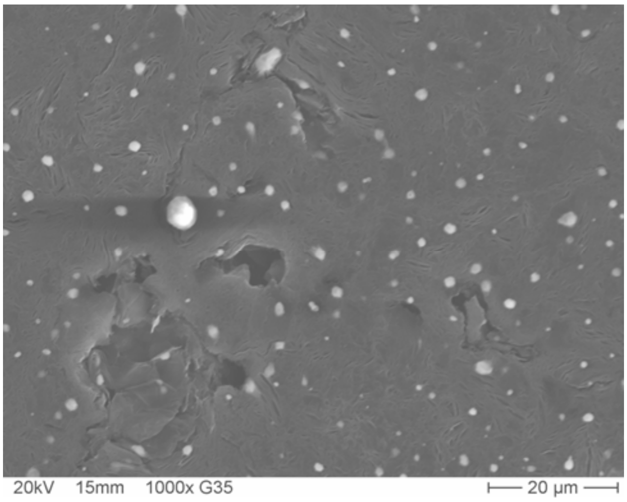

FIG. 3. U particles from the new aerosol deposition chamber

3.3. Effect of heat treatment after preparation

For this experiment the particles were either heat-treated at 350 °C for several hours or stored in a desiccator at room temperature where the relative humidity was less than 10 %. The heat-treatment was previously used to stabilize the particles and to remove excess of water. However, this also removes most of the volatile components, including fluorine, which is a potential indicator of uranium enrichment activities.

The spectra and elemental distribution maps by SEM-EDX on the particles that did not undergo the heat-treatment demonstrated that fluorine can be detected and that its signal is related to the uranium signal in single particles. The fluorine signal could still be measured in a batch of particles that was stored in a plastic box in a fume hood for one year. On the other hand, no fluorine was detected by SEM-EDX in the annealed particles [11].

The particles found in swipes taken at enrichment facilities are also generally not stable and the amount of fluorine present in these uranium oxyfluoride particles is assumed to be dependent on environmental parameters such as temperature, humidity and UV light exposure. In this way, measuring the amount of fluorine that is still retained in the particles holds important information on the history of the particles and the environment to which the particles were exposed. More results on the measurement of fluorine in the particles produced by the aerosol deposition chamber will be presented in a separate publication.

3.4. Effect of swiping

To check for morphology changes caused by swipe sampling, particles of different morphology were transferred by a 4 x 4 cm piece of cotton fabric (Texwipe-304) [12]. After swiping, this piece of cotton was cut into smaller pieces and put into a vial containing 10 mL of n-heptane. The vial was ultrasonified for a few minutes before extracting the particles from the suspension by a 10 µL pipette. In total, 150 µL was redeposited on the planchet. The transfer efficiency and the uranium intensity on the backscattered electron images and EDX spectra were sufficiently high to allow a quick localization of the swiped particles by SEM-EDX.

The morphology of the particles that were formed under high relative humidity conditions was changed by the transfer from almost spherical to a more irregular form, and agglomerates were formed during the process (Fig. 4). On the original planchets smears of uranium were visible after swiping (Fig. 5). This could be a confirmation of the high water content due to absorption as described in section 3.1.

For the particles formed in a dry atmosphere (< 20 % relative humidity), the typical clusters found previously still existed after swiping but with a shorter chain length. Many smaller single particles were also detected on the carbon planchet, and their morphology resembled the morphology of the spherical particles that were swiped.

In both cases, particle agglomeration during the swiping procedure was observed.

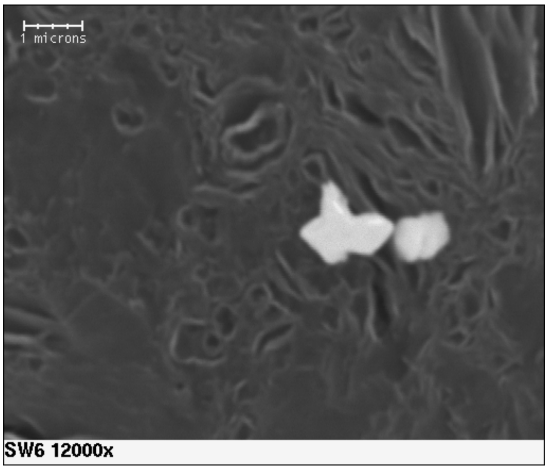

FIG. 4. U particles deposited on a planchet after swiping

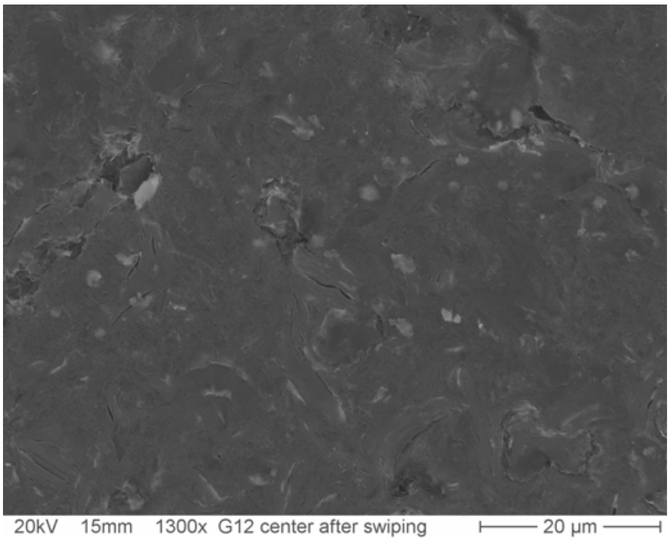

FIG. 5. Smears of U left on the planchet after swiping

4. Conclusions and Outlook

The aerosol deposition chamber developed at IRMM simulates small leaks of UF_6 to produce uranium oxyfluoride particles with a certified isotopic composition. The method is simple and flexible. By changing the humidity of the air in the chamber particles of different morphologies were produced. A humid atmosphere generates particles with an almost spherical shape and a size range of 0.1 µm to 2.25 µm whereas a dry atmosphere leads to particle clusters of a few 100 µm composed of many small particles. The SEM-EDX measurements showed the presence of fluorine in single uranium particles, even when they were stored in the lab for one year. This could be a valuable tool for the timely detection of clandestine enrichment activities. To estimate the decrease of the fluorine signal over time and to determine the chemical structure of these particles, additional experiments on µ-Raman and SIMS will be carried out. The influence of UV-light could also be an important factor here and will be taken into account as well.

The swiping and redeposition from a heptane suspension was successful as particles could be detected and measured after swiping, although the procedure partially breaks up the clusters.

Improvements on the design of the deposition chamber include an improved mechanism for breaking the glass vial, a platform for the carbon planchets and a coil to control the temperature of the air inside the chamber. This improved model is more flexible and allows us to carry out experiments on settling rate and the effect of temperature on particle morphology and composition.

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Non-proliferation and the Global Nuclear Energy Partnership (GNEP)

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Abstract. The Bush Administration announced in 2006 a new Global Nuclear Energy Partnership (GNEP) that seeks to increase U.S. and global energy security and promote non-proliferation through the expanded use of proliferation-resistant nuclear energy to meet growing electricity demand. The key elements of GNEP include the expansion of domestic use of nuclear power; demonstration of proliferation-resistant recycling; the minimization of nuclear waste; the development of advanced burner reactors; the establishment of reliable global fuel services; the demonstration of small- and medium-scale, proliferation-resistant reactors; and the revitalization of programs for advanced nuclear safeguards. If it succeeds, the partnership would demonstrate the critical technologies needed to change the way spent nuclear fuel is managed. National and international acceptance of GNEP, as well as its success, will depend in large part on early demonstration of the technical non-proliferation elements, including advanced safeguards and proliferation resistance, of all GNEP fuel cycle elements. An enhanced, systematic, defense-in-depth approach to non-proliferation that acknowledges the changing threat space, along with the new technological possibilities opened by GNEP, is essential. The prospects for GNEP are also dependent on developments in the wider non-proliferation arena, including initiatives to strengthen the non-proliferation regime being pursued by the United States, the International Atomic Energy Agency, the Nuclear Suppliers Group and the Group of Eight. This paper will assess the importance of the non-proliferation elements of GNEP, the non-proliferation challenges that might arise during efforts to transition to the GNEP vision and the technological requirements for meeting those challenges.

Introduction/Background

As part of President Bush's Advanced Energy Initiative, the Bush Administration has announced a new Global Nuclear Energy Partnership (GNEP) that seeks to increase U.S. and global energy security and promote non-proliferation through the expanded use of proliferation-resistant nuclear energy to meet growing electricity demand. The key elements of GNEP include the expansion of domestic use of nuclear power; demonstration of proliferation-resistant recycling; the minimization of nuclear waste; the development of advanced burner reactors; the establishment of reliable global fuel services; the demonstration of small- and medium-scale, proliferation-resistant reactors; and the revitalization of programs for advanced nuclear safeguards.

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[1] The views expressed are the author's own and not those of the Los Alamos National Laboratory, the National Nuclear Security Administration or the Department of Energy.

The closed fuel cycle model envisioned by this partnership requires development and deployment of technologies that enable recycling and consumption of long-lived radionuclides in radioactive waste. More specifically, GNEP would achieve its goals by:

- having nations with secure, advanced nuclear capabilities provide fuel services — assured supply of fresh fuel and the disposition of spent fuel — to other nations who agree to forgo enrichment and reprocessing activities;
- demonstrating the critical technologies needed to change the way spent nuclear fuel is managed; and
- building recycling technologies that enhance energy security in a safe and environmentally responsible manner, while promoting non-proliferation.

This paper will assess the importance of the non-proliferation elements of GNEP, the non-proliferation challenges that might arise during efforts to transition to the GNEP vision and the technological requirements for meeting those challenges.

The GNEP Non-proliferation Vision

It might be argued that because recycling technology and advanced burner reactors will be limited to either nuclear-weapon states (NWSs) or to other states with advanced fuel cycles, that non-proliferation and safeguards are irrelevant and that those GNEP elements that referred to non-proliferation were unnecessary. This would be erroneous, as it does not take into consideration such factors as domestic and international public acceptance, the importance of transparency for the states with these facilities, the long-term risks posed by states that might not accept GNEP and by nonstate actors, etc.

Non-proliferation is important to GNEP. The partnership offers a bold, comprehensive vision of the future of nuclear energy that seeks to address the challenges posed by a number of the most pressing of today’s proliferation problems. It attempts to address the spread of sensitive nuclear technology and the concerns posed by vast stockpiles of separated plutonium, as well as to meet the non-proliferation demands of a global nuclear energy renaissance.

If GNEP succeeds as planned, significant non-proliferation benefits could be expected, including:

- Slowing, if not halting, the spread of enrichment and reprocessing (ENR) technologies;
- Creating a fully functioning, effective and nondiscriminatory assured fuel supply/takeback regime that should facilitate the political acceptance of ENR limitations;
- Limiting inventories of separated weapon-usable material and ensuring that they are rigorously safeguarded, protected and accounted for;
- Slowing, if not halting, further production of separated plutonium, as new recycling technologies will allow the burning of plutonium in fast spectrum reactors without ever having separated it from other actinides; and

- Minimizing and disposing of waste, reducing potentially attractive targets for terrorists.

In this world, even if there were near-universal buy in for GNEP by states, there would continue to be proliferation problems and risks. There would be growing requirements as a result of take back to move spent fuel around the world, increasing transportation risks to some degree. At least some states could be expected to develop or expand virtual capabilities through their fuel-cycle choices, creating the prospect of a breakout. Finally, states with clandestine programs will remain a possible threat, as will nonstate actors seeking nuclear and radiological weapons. These issues they must be seen in perspective. They will appear and need to be addressed to some degree with or without GNEP, but they cannot be ignored and must be considered in the GNEP calculus.

Beyond any such risks, it must be recognized that GNEP technology, and the non-proliferation approaches surrounding it, including advanced safeguards and proliferation resistance, will need to be fully demonstrated.

Non-proliferation Elements in GNEP

Given the importance of non-proliferation to GNEP, a key lynchpin for realizing the GNEP is development of a next-generation non-proliferation system, including advanced safeguards and proliferation resistance.

Advanced Safeguards for GNEP Facilities

There is little doubt safeguards will need to evolve in the future, as they largely have over the last decades. GNEP can be a critical impetus for the evolutionary path to proceed. GNEP offers an opportunity to design an effective, integrated global safeguards system.

Advanced safeguards development and implementation is a key component of the partnership. A central need, recognized in GNEP, is cooperation with partners and the International Atomic Energy Agency (IAEA) a defense-in-depth safeguards approach to GNEP. This will involve, inter alia:

- threat assessments;
- integrated, advanced safeguards systems;
- integration of safeguards and security systems in the early design phase;
- tailored fuel-cycle development to facilitate safeguardability (e.g., fuel design); and
- modeling and simulation.

More specifically, elements of defense-in-depth could include, inter alia:

- state-of-the-art instrumentation and methodologies for materials measurement and accounting, including sensor platform integration;
- enhanced containment and surveillance, including portal and area radiation monitoring;
- integration of access denial and transparency elements of physical protection and safeguards; and
- integration of traditional process monitoring with non-traditional indicators, such as detection of radiation signals where they should not be, questionable movement of equipment and people, etc.

The advanced safeguards technologies for GNEP and other future needs will require a serious investment in research and development and advanced safeguards concept demonstrations, including the following:

- state-of-the-art instrumentation and methodologies for materials measurement;
- advanced process monitoring, including in-line process monitoring to detect diversion;
- use of operational and safety information through sharing and authentication of the operator's data;
- near real-time accounting and in-line evaluation of the effectiveness of safeguards measures;
- process simulation and modeling to optimize safeguards; and
- intrinsic transparency in facility operations.

Each of the GNEP construction projects represents an opportunity to demonstrate not only advanced fuel cycle techniques and processes but also new safeguards elements and approaches. As the experience at Rokkosho shows, early consideration of safeguards in all stages of facility design is critical. Recycling spent fuel and fabrication of transuranics require automated processes; it will be difficult and costly to retrofit safeguards after the fact.

The effort outlined above should meet emerging needs and represents a possible point-of-departure for process changes and safeguards improvements as nuclear power expands. If they succeed, they will show that GNEP can foster non-proliferation, and substantially improve the future non-proliferation environment. It will be important to understand this to the extent possible during the demonstration phase; it is not too early to begin thinking about future authorities as well as technologie**s** needed to implement them.

GNEP and Proliferation Resistance

Although there is no consensus on the meaning of proliferation resistance, it is seen to involve both intrinsic (technological) and extrinsic (institutional) factors. No separations process can be inherently proliferation resistant, and ultimately safeguardability along with extrinsic factors are central to achieving proliferation resistance. Foe GNEP, an important aspect of proliferation resistance involves its approach to dealing with plutonium.

GNEP looks to reduce the risks of a closed fuel cycle by developing and deploying new technologies to recycle spent nuclear fuel without separating pure plutonium. It envisions development and deployment of advanced burner reactors to minimize nuclear waste as well as produce energy from recycled nuclear fuel. If widely adopted, GNEP would limit the generation of stocks of separated plutonium around the world and, in the long term, offer a path to reducing existing stocks. To the extent these goals are realized, GNEP would offer improvements over the PUREX fuel cycle as utilized around the world.

More specifically, GNEP demonstration plans include UREX + processing of spent fuel with new fuel fabrication and actinide burning. UREX + would do group separation of actinides, keeping the long-lived actinides with the plutonium and burn them in fast reactors. The UREX + intermediate product, i.e., plutonium along with higher actinides is not as self-protecting as spent fuel. However, it will be hotter than separated reactor-grade plutonium, under tight safeguards and security and less attractive and practical for a proliferant state seeking its own nuclear weapons than separated plutonium. The vulnerability to possible terrorist theft is limited, and can be further reduced by modifications to the material or to the facility design.

Strengthening the Non-proliferation/Security Environment

Beyond these direct non-proliferation needs outlined, in part, in the GNEP announcement and subsequent plans, other elements of US and international non-proliferation efforts will reinforce and be reinforced by GNEP, including increased support for the Treaty on the Non-proliferation of Nuclear Weapons (NPT), the Nuclear Suppliers Group (NSG), the International Atomic Energy Agency and other non-proliferation initiatives.

There are technological challenges in these areas, as in the area of safeguards at declared facilities. In this context, several relevant areas are discussed below.

Improved Detection of Clandestine Facilities/Activities

Although GNEP calls for advanced safeguards at declared fuel cycle facilities, it does not explicitly address undeclared facilities and activities. Nonetheless, widespread undeclared capabilities would undermine the vision and needs to be addressed.

The IAEA is charged with detection of clandestine facilities and activities, a mission previously limited in practice to a few national intelligence agencies. For the Agency, the tools in the Additional Protocol, including enhances information analysis and complementary access, are central. But there is a need to develop new technical detection capabilities. This will require further work to identify technological gaps, especially with respect to detecting clandestine enrichment, and ensure they are filled to the extent possible. To do so will require in turn a significant R&D investment and strategic planning and cooperation between the Agency and member states.

Nuclear Material Control, Tracking and Forensics

The ability to accurately control, to track and to make definitive identification of material in use within facilities, in storage/disposal or in transit would greatly facilitate GNEP's objectives as well as other US and international non-proliferation and counterterrorism goals. To begin, pursuing global best practices on security, Material Protection, Control & Accountancy (MPC&A), etc., will be critical. This is an effort that can build off of US-Russian cooperation, US-IAEA interactions and other ongoing relationships. But new capabilities are needed. Some of these capabilities would be captured through advanced safeguards R&D, including enhanced process monitoring. However, there are needs that go beyond safeguards and they must be addressed as well.

NSG/Export Control Technical Support

The importance of export controls remains central even though the UREX + plants and advanced burner reactors envisioned initially under GNEP would not be widely exported, if at all. Key export control considerations include enhanced trigger list and dual-use controls in the context of requiring more stringent conditions for exports. One concrete activity for the NSG and the Zangger Committee to engage in could be technical trigger list clarification exercises on advanced burner reactors, related fuel fabrication technologies and UREX+ and related reprocessing technologies.

Development of Proliferation-Resistant Reactors

A key element of GNEP—one being pursued outside of GNEP as well—is the development of proliferation-resistant, safeguardable reactors for export. More rigorous assessments of the features of next-generation reactors and analyses of the costs and benefits of retrofitting existing reactors where feasible are necessary. A fundamental need is the development of international standards.

As part of this activity, it will be essential to understand better and to address the safeguards challenges of new reactor concepts, including assessing whether small- and medium-scale reactors designed to meet the energy needs of developing states are sufficiently proliferation resistant and, if needed, developing approaches to safeguarding these reactors.

Summary/Conclusions

Not only GNEP's acceptance, but its success, depends on early demonstration of the non-proliferation elements of, and approaches to, all GNEP closed fuel cycle elements. Moreover, it is clear that the broader non-proliferation/counterterrorism environment needs to be addressed if the objectives of GNEP are to be realized fully. In this context, many of the steps that should be taken during the transition to GNEP—which will likely last decades—are already being proposed or pursued as part of the current non-proliferation initiatives of the United States, the International Atomic Energy Agency and others. These efforts are critical to the success of GNEP; they also remain highly valuable in their own right. If GNEP is to be successful, it will build on and advance these and other US and international non-proliferation objectives.

A demonstration of advanced nuclear fuel cycle transparency concepts

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Abstract. A framework that continuously monitors facility process information can lead to greater transparency of nuclear fuel cycle activities and can demonstrate the ability of proliferation resistance associated with these activities. Sandia National Laboratories (SNL) and the Japan Nuclear Cycle Development Institute (JNC), a forerunner of the current Japan Atomic Energy Agency (JAEA), have been developing the "Nuclear Transparency Framework" concept as advanced nuclear fuel cycle transparency concept since 2003. This concept can provide an environment for the exchange of pertinent scientific and technological information to ensure the safe and legitimate use of nuclear material and technology in the future.

In 2005, SNL and JNC/JAEA entered into a cooperative program to demonstrate the methodology that has developed by those two organizations earlier, capable assessing proliferation risk in support of overall nuclear energy plant transparency. In essence, the method evaluates "expected" versus "observed" operational parameters and alerts monitoring systems if any abnormalities occur.
This advanced fuel cycle Transparency Framework Concept is being applied and demonstrated at the Fuel Handling Training Model designed for "Monju", prototype Fast Breeder Reactor (FBR). This physical model of the "Monju" fuel handling operations is located closed to "Monju" site, within the International Nuclear Information and Training Center (INITC) of the JAEA Tsuruga Head Office.

This paper describes the concept, implementation plan, status of the demonstration project, and progress toward the first demonstration of the transparency system.

1. Introduction

Future nuclear energy systems are currently under active study and discussion within several proposed international frameworks, such as the Generation IV International Forum (GIF), the Innovative Nuclear Reactors and Fuel Cycles (INPRO) initiative, and the Global Nuclear Energy Partnership (GNEP) proposed by USA. In order to be accepted by international communities, the proposed nuclear energy systems should be safe and proliferation resistant. In other words, the deployed nuclear facilities should keep both the accident risk and the proliferation risk at low, acceptable levels during their entire operational periods. In these regards, maintaining safety and transparency among nation-states and international communities is a critical key for the successful future global deployment of nuclear fuel cycles.

A key objective to successful global deployment of nuclear technology is maintaining transparency among nation-states and international communities. Proliferation resistance features that can prevent theft of nuclear material and diversion of nuclear material and technology from civilian nuclear energy systems are critical for a global nuclear future. Highly transparent nuclear systems are also highly proliferation resistant. A methodology that continuously monitors the operational status during whole life of the nuclear energy plant can demonstrate the ability of the plant to resist proliferation, thereby ensuring the legitimate use of the nuclear material and technology.

The Japan Nuclear Cycle Development Institute (JNC) and Sandia National Laboratories (SNL) started a co-study to demonstrate a "Transparency Framework" concept using the Fuel Handling Training Model (a physical mock-up) of the FBR "Monju" located in Tsuruga, Japan. The cooperative program has continued after the Japan Atomic Energy Agency (JAEA) was established by the merger of JNC and the Japan Atomic Energy Research Institute (JAERI) in 2005. Currently, this project is in its second phase, Phase II. Phase I developed the concept of the "Transparency Framework" and formulated the plans for Phase II. In Phase II, the partners are demonstrating transparency concepts and functionality using the "Monju" training model as an analog to an actual reactor facility.

2. "Monju" Fuel Handling Training Model

"Monju" is a prototype Fast Breeder Reactor (FBR), and is fully automated its operation. Its first criticality began in April 1994. However, the operation was shut down after a sodium leak accident in December 1995. Subsequently, a training model of "Monju" was developed to train operators and to help them understand how the refueling process is carried out, because the fuel handling process of "Monju" is very complicated.. The model is a physical mock-up built to scale, and is remotely controlled. The model simulates a refueling operation, carries a dummy fuel assembly from the fresh fuel storage to the reactor core via the ex-vessel storage tank (EVST), exchanges the fresh fuel and the spent fuel, and then carry the spent fuel assembly to EVST. In another operational mode, the operators can observe that the model takes a dummy spent fuel assembly from EVST and carries it to a washing pit, canning station, and finally to the spent fuel pool. This physical model helps the operators to develop and maintain their skills during the long suspension of the reactor's operation. In addition, JAEA promotes the use of "Monju" and its training facilities for international study. This increases the cooperative atmosphere for an international program and transparency.

The "Monju" training model provides an ideal and convenient opportunity to help develop the concept for an operational transparency framework, and to demonstrate and test the concept on a physical model. Because it copies the refueling movement of "Monju" precisely, no safety reviews, physical access, or other concerns are needed that would otherwise be required if a real facility was used.

3. "Transparency Framework" for Advanced Fuel Cycle

Nuclear fuel cycle tranaparency is defined here as a confidence building approach among political entities to ensure civilian nuclear facilites are not being used for the development of a nuclear weapon. Transparency involves the cooperative sharing of relevant nuclear material, process, and facility information among all authorized parties to ensure the safe and legitimate use of nuclear material and technology. A system is considered transparent when each of the parties involved can evaluate for themselves whether or not the proliferation risk of another party is at an acceptable level. The advanced transparency framework incorporates current efforts to quantify nuclear fuel-cycle proliferation risk in order to achieve transparency.

Future nuclear facilities, such as the "Monju" fuel handling facility, are expected to be more automated than current facilities. The automation of those facilities requiring minimal manual operation makes possible the generation of real-time system data that can be used to track and measure the status of the process and material at any given point in time.

Figure 1 shows the proposed framework of nuclear fuel-cycle transparency. The framework is designed to support and maintain an acceptable level of proliferation risk at an operational facility. The data that describes the normal or expected operational status, measurement data, and other types of pertinent information are collected on a real-time basis from the nuclear facility and sent to a database through a secure transmission. An analysis is conducted on the information to identify real-time inconsistencies in the operation of a nuclear facility in comparison to expected, normal, or design parameters. Rapid analysis of plant process and monitoring information to detect any abnormalities can indicate possible diversion or other illicit activities. Following the analysis, the results obtained can also be used to provide feedback to the facility operator or other authorized parties and to help recommend changes to reduce proliferation risk. This transparency system may be "owned" by an international organization such as the IAEA, a host state, or an exporter; and can be shared with the operators.

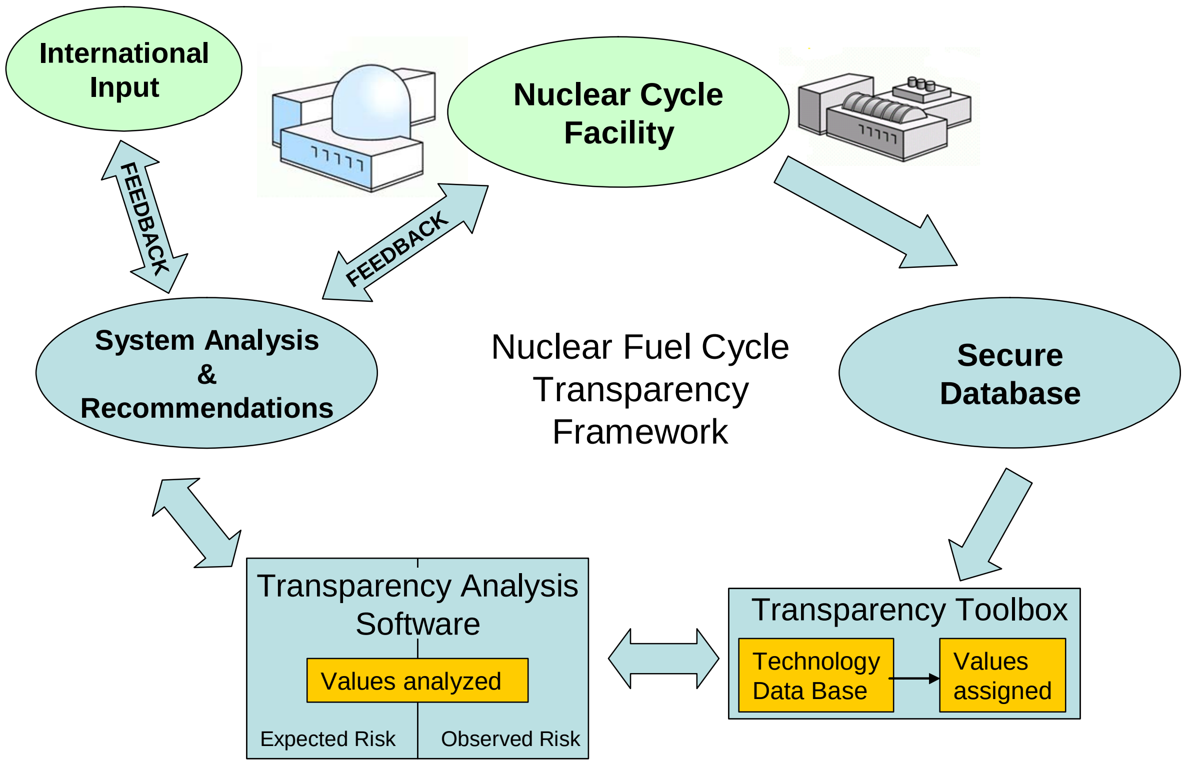

FIG. 1. Nuclear Transparency Framework

3.1. Transparency Toolbox

The "transparency toolbox" contains the system used to support the proliferation risk analysis. Real-time monitoring data from the facility and design information for the plant and related equipment are collected and aggregated within the toolbox. The facility and design information includes details unique to the site, for example, information on the crane that carries fuel assemblies, communication technologies, facility layout, operation plans, etc. The real-time data may include information from monitors and data from sensors, such as door sensors and radiation measurements, plant process data, communication status, etc. The transparency toolbox resides at the plant where the data are generated. This "local transparency toolbox" is mirrored and synchronized with a "remote transparency toolbox" that sits in a remote location where the data are analyzed. These toolboxes convert the raw signals obtained from the system into data that are usable by the transparency analyst.

3.2. *Transparency Analysis Software*

A timely and accurate analysis is necessary to design, support, and maintain transparent systems. The real-time data obtained from the facility, along with the design information for the facility and related equipment are voluminous and requires a very high level of expertise to accurately interpret. It is difficult to manually analyze those data/information to detect anomalous activity in a timely fashion. The “transparency analysis software” provides easier and more systematic detection of abnormalities.

The transparency analysis software contains the system by which the operational verification and proliferation risk of a nuclear facility is calculated. The software incorporates analysis tools in the form of a risk model to quantitatively evaluate the process data. The risk model decomposes information from the transparency toolbox to calculate the risk parameters, and compares the expected risk to the observed the risk; resulting in a calculation of the proliferation risk. The continuous stream of information provided by the software can yield a timely assessment of proliferation risk under global conditions, which can change in a short period and without warning.

3.3. *Secure Database*

The “secure database” provides the secure transmission and storage of transparency system information. A client-server database serves as the intermediary for the transmission of information and data between the nuclear fuel-cycle facility and the organization conducting the transparency analysis. The client-server collects pertinent information from the nuclear fuel-cycle facility via an electronic operations interface, and then packages and encrypts the information for secure transmission through a virtual private network (VPN). Figure 2 summarizes the flow of process data from the facility, through the client-server, and to the remote site for analysis.

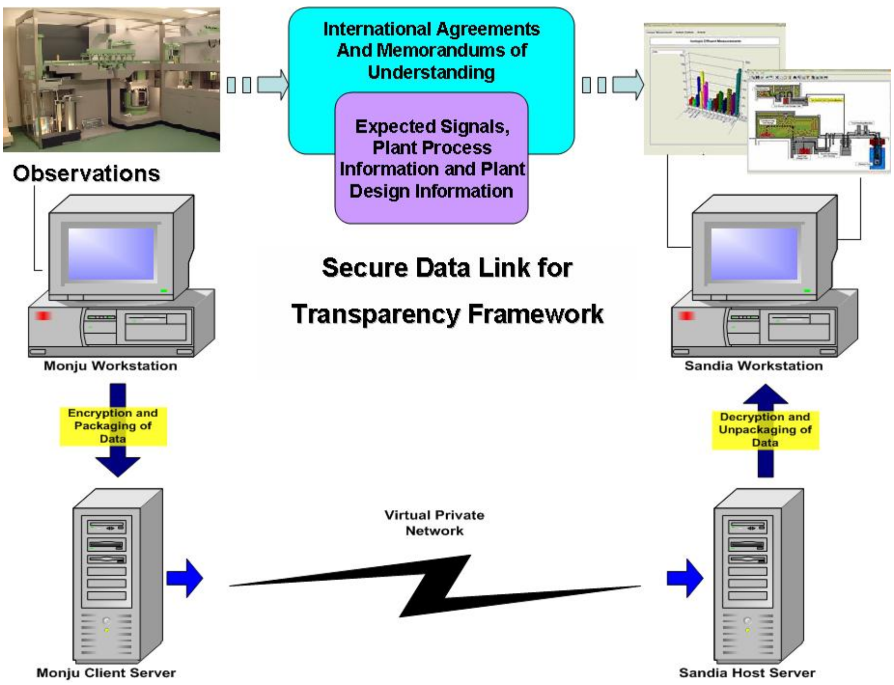

FIG. 2. Secure Information Transmission

4. Demonstration and Status of Transparency Framework

The concept of the nuclear fuel-cycle transparency framework requires an operational demonstration on a realistic test-bed to evaluate and establish its potential contribution for reducing the proliferation of nuclear weapons, and support the evolution of nuclear fuel cycles. Phase I of the project developed the "Transparency Framework" concept and formulated the plans for the subsequent phase. Phase II is to demonstrate the transparency framework concept using the "Monju" training model as an analog for an actual facility. Phase II of the project began in August 2005.

The project goal of Phase II is to apply the transparency framework to a working demonstration at the Fuel Handling Model developed for the "Monju" Fast Breeder Reactor. During Phase II of this project, the training model will be used to generate and transmit information via the secure data link previously described. Training model operations will be collected via an electronic interface connected to the Program Logic Controllers (PLC) within the model. This operational process data will be collected within the Local Technology Toolbox, which is then synchronized with the Remote Technology Toolbox. This provides analysts at SNL with real-time process data that can be assessed for anomalous activity within the physical training model using the Monju Transparency Software. The proliferation risk then is calculated using the observed process data from the PLC compared to the expected process data.

The Phase II demonstration consists of two key aspects:

1. Collection and transmission of data from the "Monju" Training Model to SNL.

2. Data analyses made to determine the validity and importance of the data collected, proliferation risk, and transparency.

Work is currently being performed to install a computer that is synchronized to the visible movements in the physical model and to mimic facility operations. The demonstration will supply process data from the model, transmit this data to a secure database at INITC, and then transmit this data to a secure database operated by analysts at SNL. A demonstration of secure communications between the location generating the data and the location analyzing the data will evaluate the integrity of the data.

The "Monju" model system has been modified to collect the data from the PLC and transmit it to the on-site database computer. This work was already completed and tested in 2006. Currently, SNL and JAEA are actively working to establish the data transmission link from JAEA to SNL, and development of the Transparency Toolbox and Analysis Software.

Once secure communications are established, a demonstration of the quantitative risk concept and its qualitative interpretation will be documented for specific operations within the fuel-cycle model. A real-time analysis of simulated proliferation risk will be demonstrated using the analysis software.

According to the current plan, Phase II will continue through August 2008. Activities will include testing, validation, review and documentation of the system. Subsequent steps for a possible demonstration at an operational facility will be discussed during this time.

5. Conclusion

Successful future nuclear energy systems will require maintaining safety and proliferation transparency among nation-states and international communities as a critical key for the successful global deployment of nuclear fuel cycles. Technologies and methods to support nuclear facility operational transparency are under active development by JAEA and SNL using the "Monju" Fast Breeder Reactor training model. After refinement and testing, application of the transparency framework for other facilities can be developed using a similar concept. Calculating the proliferation risk will depend on each individual facility and their specific data and information type. Applying the transparency framework and technologies is one factor for the successful development and deployment of future global nuclear fuel cycles.

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Integrating proliferation resistance into the design of nuclear facilities and improving safeguards approaches at the facility and State levels

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Abstract. Developing new approaches for integrating proliferation resistance into the design of nuclear facilities could lead to improved safeguards applications at both the facility and state levels. This paper provides a practical approach to integrating into the design of nuclear facilities intrinsic and extrinsic proliferation-resistance features that will provide an effective and efficient method to help prevent the diversion, production, theft, and misuse of nuclear material and sensitive technologies for use in development of nuclear or radiological weapons (i.e., “proliferation”). To facilitate this approach at the facility level, an integrated proliferation risk analysis (IPRA) process is proposed to systematically analyze a facility’s processes, systems, equipment, structures, and management controls to ensure that all relevant proliferation scenarios that could result in unacceptable consequences have been identified, evaluated, and mitigated. This approach, which can be institutionalized into the country’s regulatory structure, is similar to how facilities are licensed to operate safely and to how they are monitored through audits and incident reporting to ensure continued safe operation.

1. Background

Over the years, a significant effort has been devoted to the issue of reducing the proliferation risks associated with nuclear fuel cycles. Some major efforts to date are listed below:

- INFCE (International Fuel Cycle Evaluation) effort between 1977 and 1980: Examination of proliferation resistance to ensure that benefits of nuclear power are available.
- TOPS (Task Force on Technical Opportunities for Increasing the Proliferation Resistance of Global Civilian Nuclear Power Systems) 1999–2001 by the DOE Nuclear Energy Research Advisory Committee [1]: Identified areas in which technical contributions could be useful to increase proliferation resistance of civilian nuclear energy systems.
- INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycles) ongoing by the International Atomic Energy Agency (IAEA) since 2000: Creating an innovative nuclear power technology to further reduce nuclear proliferation risks and fulfilling the needs in the 21st century.
- GIF (Generation IV International Forum) ongoing since 2001, including the Proliferation Resistance and Physical Protection (PR&PP) Working Group: Developing Generation IV (GenIV) nuclear energy systems for meeting challenges of safety, economics, waste, and proliferation resistance.
- AFCI (Advanced Fuel Cycle Initiative): Launched in 2003 to address nuclear energy, waste management, and nonproliferation concerns [2].
- GNEP (Global Nuclear Energy Partnership), a new U.S. administration initiative announced in 2006: Seeking to redesign the nuclear fuel cycle to minimize waste, maximize benefits and availability of nuclear energy, and develop proliferation-resistant recycling technologies.

These efforts all include attempts to define and measure proliferation resistance and implement controls to maximize the proliferation resistance of the fuel cycle. This paper provides further development of those efforts by suggesting an approach to "institutionalize" the design of proliferation resistance into nuclear facilities. This concept is often called "safeguards by design."* A clear distinction must be made between the proliferation resistance of a nuclear fuel cycle and the proliferation resistance of an individual nuclear facility. The approach proposed by the authors concentrates on the proliferation resistance that can be designed into individual nuclear facilities. To build proliferation resistance into nuclear facilities, the following prerequisites must exist:

A. Ability to define and quantify proliferation resistance.
B. Ability to measure the degree of proliferation resistance of specific facility and system designs (i.e., through risk analysis techniques, modeling, simulation, or other analytical methods).
C. Institutional mechanisms used by nuclear facility operators to incorporate proliferation resistance into facilities and systems. These mechanisms will be used for licensing application, construction authorization, and justification for continued operation of the facility.

2. Defining and Quantifying Proliferation Resistance

The IAEA has defined ***proliferation resistance*** as that characteristic of a nuclear energy system that impedes the diversion or undeclared production of nuclear material, or misuse of technology, by states intent upon acquiring nuclear weapons or other nuclear explosive devices [3]. The degree of proliferation resistance results from a combination of, inter alia, technical design features, operational modalities, institutional arrangements, and safeguards measures. These can be classified as *intrinsic features* and *extrinsic measures.*

Intrinsic proliferation-resistance features are those features that result from the technical design of nuclear energy systems, including those that facilitate the implementation of extrinsic measures. Types of intrinsic features are those that reduce the attractiveness for nuclear weapons programs of nuclear material during production, use, transport, storage and disposal; prevent or inhibit the diversion of nuclear material; prevent or inhibit the undeclared production of direct-use material; and facilitate verification, including continuity of knowledge. Intrinsic features can be further broken down into material barriers and technical barriers. ***Extrinsic proliferation-resistance measures*** are those that result from states' decisions and undertakings related to nuclear energy systems.

Safeguards is an extrinsic measure comprising legal agreements between the party having authority over the nuclear energy system and a verification of control authority, binding obligations on both parties, and verification using, inter alia, on-site inspections. Safeguards are realized chiefly through the application of technology and management controls for accounting and control of nuclear material and technology. Safeguards have both national and international components that play complementary roles in proliferation resistance. National safeguards provide protection against theft or diversion by subnational groups with malicious intent to develop nuclear weapons or to sell nuclear material on the black market to another nation that is intent on proliferation. International safeguards deter proliferation at the nation-state level and permit response by other member states of the IAEA and other international organizations by detecting proliferation attempts.

3. Measuring the Degree of Proliferation Resistance of a Facility

The second prerequisite of achieving proliferation-resistant designs is the ability to measure the degree of proliferation resistance of specific facility and system designs. The PR&PP Working Group is an international group that was organized in December 2002 [4] under the GIF. The group is developing measures for defining proliferation resistance and physical protection separately, with the intent to develop a comprehensive methodology for evaluation of the proliferation resistance and physical protection of GenIV nuclear energy systems. The results of this effort will address this prerequisite. The remainder of this paper will concentrate on how to develop design approaches for proliferation resistance using the assumption that the degree of proliferation resistance in the designs of nuclear facilities can be measured.

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*The term "safeguards by design" is somewhat misleading since proliferation resistance is broader and includes both physical protection and management controls. However, physical protection is usually differentiated from safeguards, and management controls are not usually associated with the design of a facility.

In designing facilities or systems, process flow sheets are developed. It is inherently more efficient and effective to incorporate proliferation resistance and requirements and plan for the application of state and international safeguards at this stage than try to retrofit the requirements after the design has been completed. This is a very simple but crucial concept that must be adequately defined and developed. Therefore, the basic approach to integrating proliferation resistance into facility designs should be modeled on the approach that has been successfully used by regulators and industry to integrate safety into the design of nuclear facilities. We can define this as an "integrated proliferation risk analysis (IPRA)," which is a systematic examination of a facility's processes, systems, equipment, structures, and management controls to ensure that all relevant proliferation scenarios that could result in unacceptable consequences have been adequately evaluated and the appropriate measures identified and implemented to eliminate or mitigate such scenarios.

4. A Regulatory Approach to Proliferation Resistance and Design

Two basic principles were put forth by INPRO to provide high-level guidance regarding innovative nuclear energy systems. Basic Principle BP1: *Proliferation-resistance features and measures shall be implemented throughout the full life cycle for innovative nuclear energy systems to help ensure that innovative nuclear energy systems will continue to be an unattractive means to acquire fissile material for a nuclear weapons program.* Basic Principle BP2: *Both intrinsic features and extrinsic measures are essential, and neither shall be considered sufficient by itself* [3]. This is the basic premise behind the concept of safeguards by design. However, the INPRO guidance is high level only and, as is the case for the other efforts in proliferation resistance, no clear methodology has been provided to date for incorporating this principle into actual facility designs. Therefore, to satisfy this third prerequisite for building proliferation resistance into facility designs, a methodology to implement these concepts must be developed.

4.1 Current National Safeguards Approach

For purposes of this paper, <u>the U.S. national safeguards program is used as an example.</u> The U.S. civilian national safeguards program supports the implementation of international safeguards for U.S. nuclear facilities and activities, excluding only those facilities having activities with direct national security significance to the United States. Administratively, U.S. national-level safeguards are overseen by the U.S. Nuclear Regulatory Commission (NRC) for commercial nuclear facilities and activities. The U.S. national safeguards program is codified in NRC regulations that take into account a wide spectrum of domestic and international subnational threats.

These regulations and directives establish minimum performance standards and mandate certain safeguards measures for nuclear facilities and activities using a graded approach that addresses safeguards risks posed by the nuclear materials. The NRC licensees responsible for operating nuclear facilities develop and implement the safeguards measures that meet the NRC regulations. These programs are described in submittals to NRC. NRC reviews these submittals; develops specific details with the licensees; and approves the revised submittals, which serve as the bases for NRC oversight of the licensee safeguards measures. The NRC regulations mandate that licensees provide physical protection, material control, and material accountability measures for nuclear material. The regulations also establish requirements for export control of equipment and materials; classification and protection of information; personnel security, including background investigation; and a program to ensure continued reliability.

The NRC conducts inspections and investigations to ensure that safeguards requirements are being met and may impose civil penalties and sanctions for violations. In addition, U.S. law imposes criminal penalties and sanctions for violation of certain safeguards requirements and certain activities related to nuclear materials (e.g., theft, diversion, or unauthorized possession). Incidents of this type are investigated by the Federal Bureau of Investigation and, where warranted, prosecuted by the U.S. Department of Justice. The NRC safeguards regulations are contained in Title 10, "Energy," of the Code of Federal Regulations (CFR).

The existing regulatory requirements and structure mandated in 10 CFR for nuclear material control and accounting (MC&A) and physical protection requirements can be used as the basis for creating a comprehensive approach that covers all aspects of proliferation resistance: facility and systems design, current and advanced safeguards technologies, physical protection, MC&A, process monitoring, and

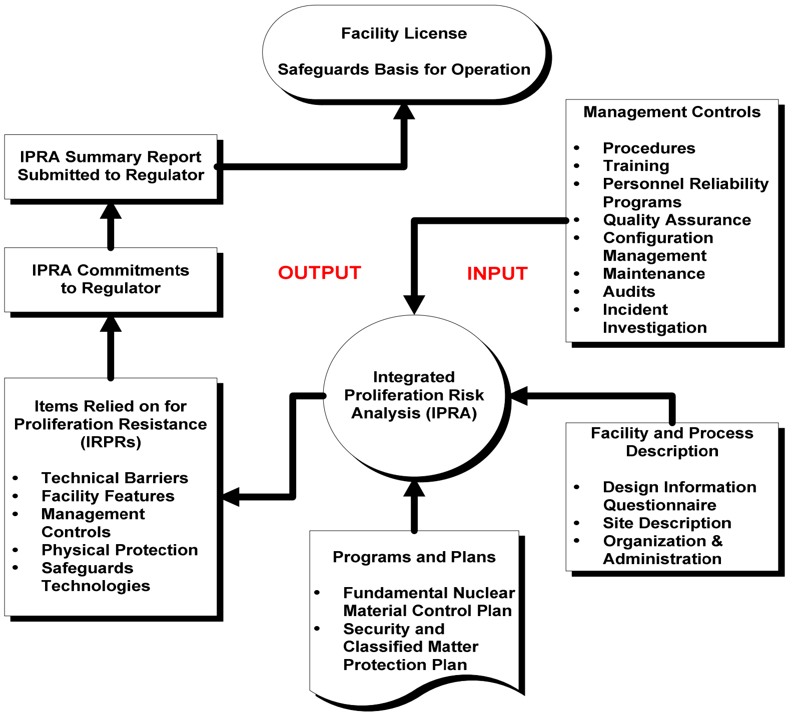

Figure 1. IPRA process—"institutionalizing" proliferation resistance.

integration with plant operations. What is required is an IPRA process modeled after the NRC integrated safety analysis (ISA) process. The ISA process identifies potential accident sequences in the facility's operations, designates items relied on for safety (IROFS) to either prevent such accidents or mitigate their consequences to an acceptable level, and describes management measures to provide reasonable assurance of the availability and reliability of IROFS. All applicants for new licenses must perform an ISA and submit a summary to the NRC for approval prior to receiving the NRC license to construct and operate. The proposed IPRA process is modeled in Figure 1.

4.2 Integrated Proliferation Risk Analysis at the Facility and State Level

The IPRA approach proposed by the authors can be defined as a systematic examination of a facility's processes, systems, equipment, structures, and management controls to ensure that all relevant proliferation scenarios that could result in unacceptable consequences have been adequately identified, evaluated and mitigated. This approach will require development of regulatory guidance for proliferation resistance, which would be similar in concept to NUREG 1513 [5] and NUREG 1520 [6] (used for the ISA process in the United States), needed to assist the nuclear facility designers in demonstrating that the criteria for proliferation resistance have been met. The requisite information can be included in the licensing submittals currently required for approval of a nuclear facility construction and operating license. Similarly, additional guidance will need to be developed for regulatory staff to verify compliance with the design requirements. In the U.S. regulatory system, these items would be developed as part of a package of proposed rule changes under the Administrative Procedures Act. Although different nations have different regulatory approaches, most, if not all, modifications to national safeguards requirements would include similar rule change and guidance development activities.

The development and implementation of the IPRA process is a creative and aggressive mechanism for institutionalizing proliferation resistance. The IAEA could encourage the international implementation of the IPRA concept through the development of conventions and documents. One historically consistent approach would be for the IAEA to convene a working group of member states that are nuclear technology holders to prepare a draft similar to the IAEA Safety Standard, *Draft Safety Guide (DS 344), Safety of Conversion and Enrichment Facilities* [7]. Alternatively, a similar approach could be taken to that of IAEA Convention of Nuclear Safety [8]. The Convention was adopted in Vienna on June 17, 1994. The Convention was drawn up during a series of expert-level meetings from 1992 to 1994 and was the result of considerable work by governments, national nuclear safety authorities, and the IAEA Secretariat. Its aim is to legally commit participating states operating land-based nuclear power plants to maintain a high level of safety by setting international benchmarks to which states would subscribe. It should be noted that the design information questionnaire that is required by the IAEA for all civilian nuclear facilities already contains the facility and process description and the site description and organization, which are the bases of the proposed IPRA process.

If this standard were implemented through an appropriate international effort, many nations might be willing to adopt it. If the Nuclear Suppliers Group were willing to make compliance with the IAEA requirements for application of the proliferation-resistance design criteria a condition sale of trigger-list components, the standard would become mandatory for many nations.

The IPRA process will require that controls and systems are in place to ensure adequate proliferation resistance and secure operation of the facility. IPRA techniques applied to nuclear facilities must address the special vulnerability paths that are present at such facilities and their potential for theft, diversion, or undeclared production of nuclear material and must identify items relied on for proliferation resistance (IRPRs). The IRPRs are analogous to the IROFS required in the ISA process. The IPRAs must provide the following elements:

- a description of the structures, systems, components, and processes at the facility;
- an identification and systematic analysis of proliferation vulnerabilities at the facility and identification of proliferation sequences that would result in unacceptable consequences, as well as the expected likelihood of those sequences;
- an identification and description of controls (i.e., structures, systems, equipment, components, processes, and management controls) that are relied upon to limit or prevent potential proliferation activities or mitigate their consequences; and
- an identification of measures taken to ensure the availability and reliability of identified IRPRs.

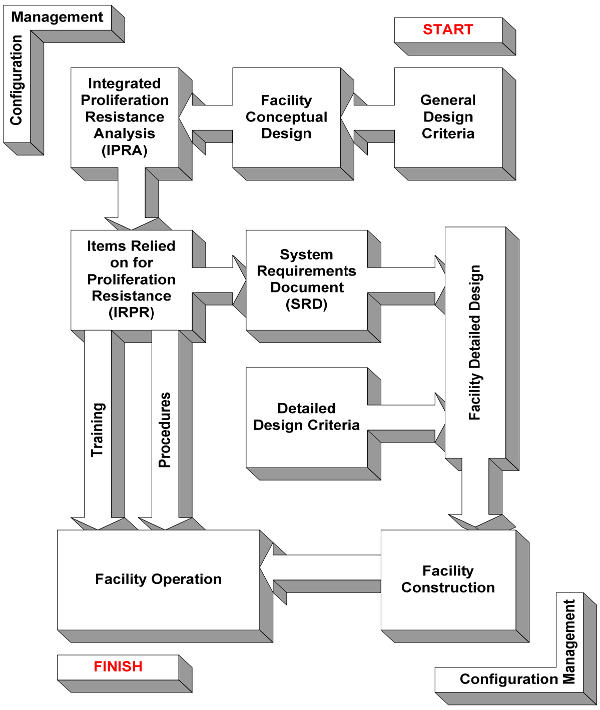

Figure 2. Design criteria flow-down.

Management controls require special mention here and can include the following programmatic elements: policies and procedures; training and qualifications; maintenance, calibration, and surveillance; document and records control; configuration management; quality assurance; audits and self-assessments; human reliability programs; and incident investigation.

Proliferation resistance is implemented in the facility design through the use of design criteria. The flow-down process from the general design criteria to the detailed design criteria and to the construction and operation of the facility is represented in Figure 2. The IPRA process supports the identification of the complete suite of IRPRs and the performance of safeguards and security operations at the facility and establishes the roles and importance of individual design and programmatic elements. It is important to recognize that the detailed design criteria and system performance requirements evolve as the facility design matures. This design and evaluation process, as described in the next section, is the core of the design process mandated by the IPRA approach.

4.3 Design and Evaluation Process at the Facility Level

The methodology by which proliferation resistance is incorporated into the design of facilities and systems is made possible by the development of design criteria and can be categorized as the following four types: (1) the design process itself, (2) the intrinsic proliferation resistance features, (3) features that facilitate extrinsic proliferation-resistance measures, and (4) extrinsic proliferation-resistance measures. **Design process criteria** are those concerned with the overall process of ensuring that the facility design is performed in a manner that affords appropriate consideration to proliferation resistance from the initial phases of design through the construction and operation of the facility. Design criteria for **intrinsic proliferation-resistance features** require that the design include features that impede theft, diversion, or undeclared production of nuclear material or misuse of technology. Design criteria for **features to support extrinsic proliferation-resistance measures** require that a nuclear facility design be conducive to the cost-effective application of the physical protection, MC&A, and inspection and verification measures needed to meet international- and national-level safeguards requirements. Design criteria for **extrinsic proliferation-resistance features** use actual safeguards technologies and management controls. These categories are depicted in Figure 3.

The proliferation-resistance design criteria for nuclear facilities must be implemented by using a structured process. The traditional systems engineering process, which uses the development of system requirements documents (SRDs), can satisfy this need [9] [10] [11]. The systems engineering process covers a broad range of activities that involve the design and operation of a facility. These include the following:

- identifying and integrating proliferation-resistance requirements;
- identifying and evaluating proliferation vulnerabilities and conducting an analysis of the effectiveness of the potential mitigation measures;
- coordinating multidisciplinary teamwork in implementing proliferation-resistance requirements;
- providing safeguards-related interface management;
- providing configuration management to include the establishment of baseline configuration; and
- coordinating technical reviews of the facility proliferation-resistance features.

The application of systems engineering activities to the proliferation-resistance aspects of facility design should utilize a graded approach and should be commensurate with the facility hazards and complexity. The goal is to ensure that the systems engineering activities incorporate the appropriate proliferation-resistance features. When developing the safeguards aspects of the facility design, there is a logical sequence of design considerations to follow, as shown in Figure 3.

First, the facility/system design will be reviewed to identify opportunities to incorporate intrinsic proliferation resistance features. These features are used to impede and detect theft, diversion, or undeclared production of nuclear material or misuse of technology.

Second, the facility design should be analyzed to identify proliferation vulnerabilities, using analysis techniques developed by one of the international proliferation-resistance programs/efforts. A comprehensive suite of intrinsic and extrinsic design solutions will be identified to eliminate or mitigate the identified vulnerabilities. As a general rule, technical barriers incorporated into the facility design are preferred to extrinsic measures, and measures or features that eliminate vulnerabilities are preferred to those that mitigate vulnerabilities. Additionally, intrinsic proliferation-resistant design features will be considered that will facilitate the application of extrinsic proliferation-resistant measures, such as the efficient application of international safeguards and the associated implementing technologies. Some examples are as follows:

- design of the physical layout that facilitates efficient access control and physical protection features for vehicles, people, nuclear materials, and process equipment;
- design of a plant layout that facilitates containment and surveillance;
- design of the process areas, including piping and valving, that protect proliferation-sensitive information while facilitating visual verification of piping configuration and valve alignment;

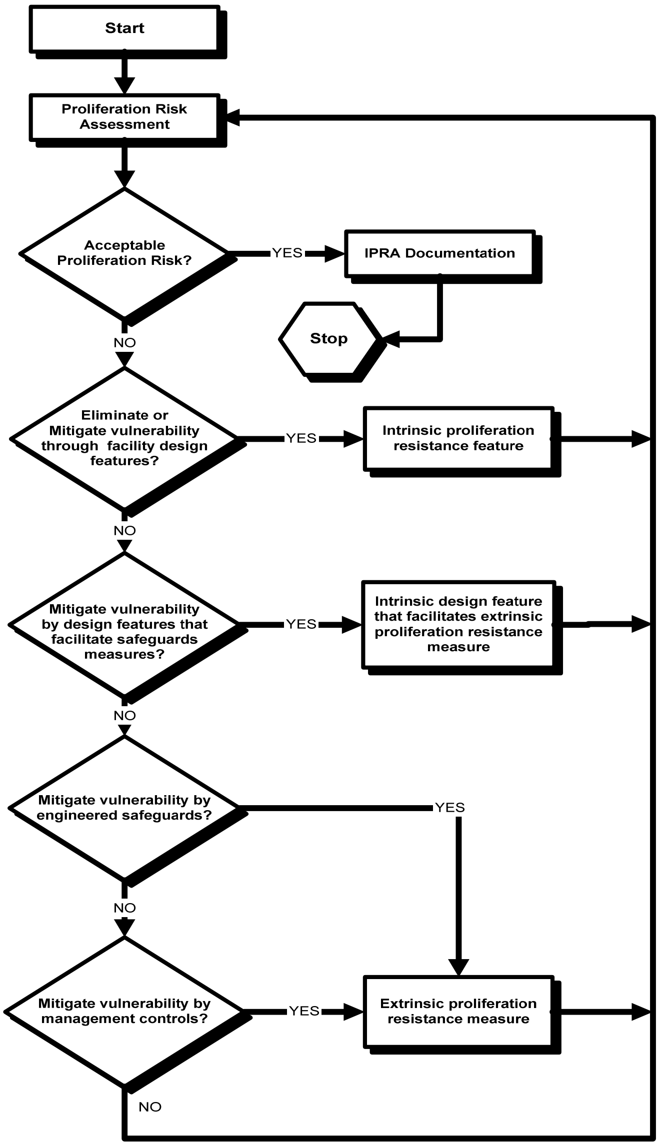

Figure 3. Design and evaluation process.

- design of sampling ports to facilitate monitoring; and
- design of piping to facilitate continuous material monitoring.

<u>Third</u>, the potential vulnerabilities will be addressed by the extrinsic features such as safeguards and management controls. As the facility design matures, this design and evaluation process will be repeated as appropriate to identify possible new vulnerabilities or opportunities for safeguard improvement created during the design evolution.

A useful feature of this approach is that it encompasses the entire safeguards "envelope" during the design, licensing, construction, operation, and decommissioning phases. Additionally, it can facilitate the efficient integration of safeguards, safety, and normal plant operations. During the IPRA process, the IRPRs can be the same as those relied on for safety or normal operations, provided that the proper instrumentation robustness and data authentication are achieved. In other words, existing systems that are already designed into the facility are "credited" during the IPRA process. In this way, the impact of implementing proliferation resistance into the facility design and operation is minimized. Furthermore, this approach recognizes that there are cost trade-offs between intrinsic features and extrinsic measures and encourages their optimization for cost-effectiveness. It further recognizes that the features and measures must be compatible with other design considerations, such as safety and economics, and that the proliferation-resistance costs must be reasonable. Some technical features that provide proliferation resistance may be incorporated into the design of a facility primarily for other reasons such as safety or functionality. In evaluating the total cost of implementation, it is important to realize that the actual cost of a proliferation-resistant design feature is only incremental to the cost that has already been allocated for safe and/or efficient operations.

5. Concluding Remarks

From this discussion, it can readily be seen that there exists a significant need to develop an approach to incorporate the concepts of proliferation resistance into the design process for nuclear facilities. This could lead to improved safeguards application at both the facility and state levels. This has been achieved in the nuclear safety arena for quite some time, with very positive results. Nuclear power stations are more efficiently and safely operated, and this accomplishment has led to an increased public acceptance of nuclear power. As is normally the case in human endeavors, the emphasis on safety increased significantly in the United States after the accident at the Three Mile Island nuclear plant in 1979. That event highlighted both the success of the safety culture in the United States and the need for improvement. The resulting regulations and safety approaches stemming from the accident considerably improved the safety awareness in the nuclear industry and the approach to "safety by design." There are important lessons that should be learned from the history of the development of safety culture in the United States and internationally, and it is hoped that a similar incident involving the proliferation of nuclear material or technology is not necessary to spur the movement toward "safeguards by design." As put forth in this paper, the IPRA process can be an effective component in supporting the integration of proliferation resistance into the design of nuclear facilities.

To implement this "safeguards by design" approach, there must be a concerted effort by national and international safeguards communities to develop methodologies to implement this vision. The near-term efforts of the active programs, such as GNEP and INPRO, offer an excellent opportunity to begin mapping out a process of translating the proliferation resistance knowledge possessed by the safeguards community into the designs of the facilities. Also, the involvement of the IAEA in developing international conventions, INFCIRCs (information circulars), and guidelines will be necessary in gaining international approval and effective implementation worldwide. This can also help meet some of the key requirements of the United Nations Security Council Resolution 1540, *Non-proliferation of Weapons of Mass Destruction* [12], which mandates that all states adopt and enforce effective laws to prevent theft of nuclear material and enforce effective measures to establish domestic controls to prevent proliferation of nuclear material and technology. This will likely be an iterative process because the concept of designing a system to eliminate or mitigate proliferation vulnerabilities, or to select the optimal safeguards technologies, is new, and there are no existing guidelines for implementation. However, the effort must begin now to develop a formalized process whereby proliferation-resistance requirements are documented and provided to the designers of the next generation of nuclear facilities as the basic set of requirements that must be met before these new facilities are designed, constructed, and operated around the world.

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International safeguards and the Global Nuclear Energy Partnership

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Abstract. A fundamental principle of the Global Nuclear Energy Partnership (GNEP) is that success in expanding the role of nuclear power will require continuing improvements in international safeguards effectiveness and efficiency. The United States is committed to working with the IAEA and other international partners to address the safeguards challenges of GNEP. The U.S. approach includes an assessment of GNEP's proliferation risk reduction potential, encompassing the overall GNEP architecture, as well as specific facility types and transportation. The United States intends to make GNEP-related facilities eligible for IAEA safeguards and will stress the early incorporation of international safeguards requirements in facility and process designs. Advancements in safeguards methods and technology will be required, resulting in an immediate need to increase investment in R&D for international safeguards.

Introduction

One of the fundamental principles of the Global Nuclear Energy Partnership (GNEP) is that success in expanding the role of nuclear power will depend on continuing improvements in the effectiveness and efficiency of international safeguards. As envisioned by the Bush Administration, GNEP will encourage the development and deployment of a new generation of more proliferation-resistant fuel cycle systems that avoid the separation of pure plutonium, and, in the long term, reduce global inventories of separated plutonium through the use of new fuel cycle facilities and the development and deployment of advanced, fast burner reactors. GNEP will encourage the development and deployment of small and medium size proliferation-resistant power reactors. These are among the nonproliferation benefits of GNEP. In order to achieve these goals, however, we will need to find ways to safeguard effectively and efficiently the new fuel cycle systems – the nuclear materials and the facilities in which those materials will be produced, used, stored, and disposed of – envisioned under GNEP.

The United States is taking a comprehensive approach to international safeguards for GNEP. This approach includes a thorough assessment of the proliferation risk reduction potential of the various fuel cycle architectures that may be possible under GNEP. For GNEP facilities in the United States, we intend to stress the early incorporation of safeguards concepts and technology in facility designs – the philosophy of "safeguards by design." Advanced technology will be a key factor in continued improvements to safeguards effectiveness and efficiency. In the United States, our national laboratories are examining advanced safeguards approaches that may be applicable to GNEP, and we anticipate the need for significant increased investment in safeguards technology development. Last but not least, we expect that international collaboration will play a major role in developing and gaining acceptance of the new safeguards measures necessary to support GNEP.

International Collaboration

The United States is committed to working with the International Atomic Energy Agency (IAEA) and other international partners to address the safeguards challenges posed by GNEP. While it may present challenges, GNEP also represents an opportunity to further strengthen the international safeguards system. In the U.S. vision for GNEP, technical advancements in safeguards for new processes, facility designs and fuel types will lead directly to improvements in the effectiveness and efficiency of international safeguards wherever they are applied.

Many cutting edge developments are already occurring that may be relevant to GNEP, in such areas as unattended and remote monitoring (UNARM), process monitoring, and safeguards data integration and analysis. Much of that ongoing work is happening outside the United States. In addition, the IAEA, with assistance from Member States, continues to seek novel solutions for the verification of declared nuclear materials and, especially, for the detection of undeclared nuclear activities. One of the strengths of the international safeguards regime is the network of bilateral arrangements and Member State Support Programs that enable and support work on safeguards technology challenges. The United States currently has ten bilateral safeguards cooperation arrangements with foreign partners and the U.S. Program of Technical Assistance to IAEA Safeguards is a major provider of direct technical support to the Agency. We expect consultations now underway will lead to further collaborations on international safeguards issues related to GNEP. In the near term, our existing bilateral safeguards cooperation arrangements and possibly Member State Support Programs provide mechanisms for collaboration.

Proliferation Resistance

GNEP is intended to take advantage of both intrinsic (e.g., design) features of advanced fuels and recycle technologies, and extrinsic factors, including institutional measures like international safeguards and fuel leasing, to reduce the proliferation risks associated with the expansion of nuclear power. GNEP's promise of proliferation risk reduction is not a function of any one institutional or technical measure, but rather the combination of measures within the entire GNEP fuel cycle architecture. The advanced recycle and fast reactor technologies envisioned under GNEP may reduce the proliferation attractiveness of nuclear materials and processes, but they will be only part of the picture. How intrinsic and extrinsic factors can be combined to reduce the potential for proliferation, through diversion of material or other undeclared activities, is of fundamental importance to GNEP.

Understanding how intrinsic and extrinsic factors can be synergistically integrated to reduce the proliferation risks associated with a given facility or process will require a broad-ranging assessment. The United States has begun an effort to conduct such an assessment, encompassing the overall GNEP architecture, as well as specific facility types and transportation. This proliferation risk reduction assessment will build on the methodology developed by the Generation IV International Forum's Proliferation Resistance and Physical Protection Working Group [1]. International safeguards considerations will be an important element in the analysis. Questions addressed in the assessment will include how facilities can be designed to enhance their "safeguardability." The assessment will examine how significantly new process flowsheets and facility designs may be used to limit opportunities and increase the resources and time a proliferator would need in order to obtain weapons-usable nuclear materials (particulary for a non-state actor). It will consider how readily detectable the misuse of a facility might be, given the application of IAEA safeguards at that facility. In addition, the assessment will look at potential ways, for instance through technology, to increase the efficiency of safeguards, so that IAEA resource constraints do not limit safeguards effectiveness.

GNEP Implementation in the United States

In implementing GNEP domestically, the United States will emphasize the early incorporation of advanced international safeguards technology and approaches in the design of new fuel cycle facilities, reactors, and associated materials storage and transportation systems. This will require close coordination with facility designers, an approach referred to as "safeguards by design."

The United States plans to develop and deploy a commercial-scale Consolidated Fuel Treatment Facilty (CFTF), using the UREX+1a or comparable recycle technology, and an Advanced Burner Reactor (ABR). GNEP also calls for construction of an Advanced Fuel Cycle Facility (AFCF) to support long-term, advanced fuel cycle research and development. The Administration has sought expressions of interest from industry to accelerate delivery of the CFTF and the ABR. The AFCF is envisioned to serve as a state-of-the-art fuel cycle research and development laboratory, which would include the development, design and testing of advanced instrumentation, controls, and monitoring equipment in support of advanced safeguards technologies. The AFCF would be designed and operated through the DOE national laboratory system.

The United States intends to make its GNEP facilities eligible for international safeguards, consistent with the terms of our safeguards agreement with the Agency. We will also seek to ensure that appropriate safeguards measures will be available should the IAEA choose to select any of the U.S. GNEP facilities that are eligible for safeguards. The "safeguards by design" approach will be critical for achieving that goal. In order to proceed past initial conceptual design, it will be necessary to show that the enabling technologies for GNEP (materials separation, fast reactors, and recycling) can be implemented in a manner that satisfies IAEA safeguards requirements. The early incorporation of those requirements into the facility design could greatly facilitate the application of Agency safeguards, leading to improved safeguards results while minimizing resource impacts both to the facility operator and to the Agency. Regardless of whether any of the U.S. GNEP facilities were selected for Agency safeguards, these facilities could serve as platforms for supporting collaborations with the IAEA and other partners to demonstrate international safeguards concepts and technologies in support of GNEP.

Safeguards by Design.

The early incorporation of safeguards in facility designs, or safeguards by design, is intended to produce a more robust safeguards approach at less cost, to both the IAEA and facility operators, than would be the case if safeguards had to be factored in late in the design process. In other words, safeguards would be a goal of the facility or process design, rather than an external requirement imposed after the design has been completed.

The nuclear safety community has long recognized the benefits of integrating passive safety features into nuclear power plant designs, and therefore may offer lessons for nuclear safeguards integration. Recent discussions on proliferation resistance have also highlighted the advantages of the early incorporation of safeguards concepts into facility designs [2].

As plans for GNEP facilities progress, the United States will encourage close collaborations between engineers, facility operations specialists, others responsible for facility design and safeguards experts who will be involved in the design process. These collaborations will build on safeguards work that has already been done at U.S. national laboratories, especially in the context of the Department of Energy's Advanced Fuel Cycle Initiative (AFCI) program, a precursor to GNEP [3]. As participants on design teams, safeguards experts would be involved in gathering information on planned processes and equipment, and determining potential safeguards issues. They would recommend material balance areas, key measurement points and process-monitoring activities, containment and surveillance, and other measures relevant to the application of international safeguards. They would also provide advice on additional research and development that may be required to meet safeguards needs of the new GNEP flowsheets and facilities.

For commercial facilities, the U.S. Nuclear Regulatory Commission (NRC) is taking steps to include "security by design" as a regulatory requirement for certification of advanced reactor designs, and may extend this to include a requirement for safeguards by design [4][5][6]. As a result, if GNEP facilities are subject to NRC licensing, then the licensing process would likely require technical documentation of a safeguards-by-design approach.

In addition to collaboration between facility designers and safeguards experts in the United States, it will be equally important for U.S. design teams to work closely with the IAEA. In the early 1990's, the participants in the Large-Scale Reprocessing (LASCAR) project concluded that among the benefits of such consultations, begun early in design and continued through construction and commissioning, "would be the implementation of a more efficient and less intrusive safeguards approach" [7]. The IAEA Board of Governors' decision in 1992 concerning the provision of design information establishes a framework within which early engagement with the Agency on safeguards and facility design issues can occur [8]. Compared to States with advanced nuclear power programs subject to comprehensive safeguards agreements, the United States has relatively limited experience working with the IAEA to incorporate international safeguards in our own facility and process designs. We expect the lack of practical experience to be offset, however, by the technical expertise of our national laboratories, which have provided extensive support to the Agency for the application of safeguards, and by the commitment of policy makers to proactive engagement with the Agency on international safeguards issues related to GNEP [9].

Safeguards Technology and GNEP

While the expansion of nuclear power under GNEP will result in increased demands on the IAEA, it should also serve as a catalyst for strengthening the effectiveness and efficiency of international safeguards. The idea that technology will play a key role in strenthening safeguards is not new, of course. As proliferation threats evolve and place still greater demands on safeguards, the IAEA is asking Member States for continued support in identifying novel technological solutions to further improve its ability to verify declared materials and, especially, to detect undeclared nuclear actitivities. However, while important new tools have been made available to the Agency in recent years, the overall technology development picture has been one of contraction rather than growth [10]. In order to realize the nonproliferation potential of GNEP, it will be necessary to vigorously reinvest in technology development for international safeguards.

There are several areas where GNEP may help to promote – and indeed will likely require – technical improvements in the existing safeguards system. For instance, we expect a need for greater investment in R&D concerning methods and instrumentation for nuclear material measurements and accountancy, process monitoring, containment and surveillance, and robust, highly reliable UNARM systems.

Facility designers and the IAEA may both need enhanced capabilities in systems modeling and simulation in order to evaluate the trade-offs and synergies between safeguards and operational requirements of facilities. Current facility simulation capabilities offer the potential to improve both the incorporation of safeguards in facility and process designs and safeguards implementation. For example, design teams and the IAEA could consider diversion scenarios of interest in a "virtual" facility. Safeguards systems could then be designed and evaluated in terms of their ability to detect such scenarios.

The early incorporation of safeguards requirements in facility design should facilitate the implementation of independently verifiable process monitoring systems. Continuing developments in the areas of process control and in-line instrumentation represent new opportunities for safeguards. Plant operators from chemical and other industries continue to develop more sophisticated means of automated control, monitoring and diagnostics to manage their facilties. While these developments have been driven by economic, environmental or safety-related reasons, they may also offer lessons for how to achieve similar improvements for safeguards.

Advanced techniques for the collection and analysis of safeguards and process control data will be needed for the large, complex fuel cycle facilities contemplated under GNEP. Facility and process monitoring systems could generate an enormous amount of data of potential safeguards significance. To be useful for inspectors, however, such data will need to be authenticated. Once the data taken from facilities are validated, new tools will likely be needed to analyze the data and to integrate it with other information used by the Agency in drawing conclusions and evaluating safeguards effectiveness.

Finally, GNEP technology development may lead to improvements related to new safeguards technologies and other methods for detection, tracking, and forensic analysis. Such improvements could strengthen the Agency's capabilities for detecting undeclared nuclear activities and for investigating possible cases of safeguards noncompliance.

Conclusion

As envisioned by the Bush Administration, GNEP holds the promise of expanding the use of nuclear power in the United States and supporting the projected growth of nuclear energy use worldwide, while also reducing proliferation risks. In order to fulfill the nonproliferation goals of GNEP, however, it will be necessary to further strengthen the international safeguards system while reducing resource demands on the IAEA. The United States is taking a comprehensive approach to GNEP and international safeguards. This approach includes a thorough assessment of the proliferation risk reduction potential of GNEP, looking at a range of implementation scenarios. In our domestic implementation of GNEP, the United States intends to make facilities eligible for IAEA safeguards and will stress the early incorporation of international safeguards requirements in facility and process designs. Our technical experts are investigating advanced safeguards approaches that may be applicable to GNEP flowsheets and facilities and we anticipate that a significantly expanded investment in R&D for international safeguards will be needed. The United States looks forward to close collaboration, with the IAEA and other partners, on the international safeguards issues that will need to be solved in order to realize the nonproliferation potential of GNEP.

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Future interactions between IAEA safeguards and trade in sensitive nuclear technologies

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This paper represents only its author's opinions and do not reflect either French positions, or the French ministry of defense position.

Abstract. Recent events have clearly shown that the non proliferation regime is currently under threat. One obvious cause is that some states – and clearly not only those outside the NPT! – have elected to develop non civilian nuclear programs. A less common reason is that, while control lists have already been subject to a lot of accuracy improvements and attempts to be exhaustive, trade in nuclear sensitive goods and technology has concurrently expanded, making the control always more difficult to enforce, partly for judicial and financial reasons, and partly because of technical issues.

Thus, the non-proliferation regime is at a crossroads. The main axis for improvement is to strengthen the link between trade control procedures and the Agency's safeguards and verification system. There are a few options to explore: one would be to subject the exportation to an IAEA-related condition, such as the ratification of an Additional Protocol. Another one would be to force the universality of the Additional Protocol, which includes for any country the obligation to declare any exportation of trigger-list and some dual-use list items. The last – and most polemic – one is the creation of international centers for all sensitive steps of the fuel cycle, with the crucial question of how to guaranty the access rights.

The immediate quandary to such changes is that it might be perceived as a restriction, imposed by the nuclear-weapon States, of the access right to the peaceful use of nuclear energy conferred by NPT's article IV. It is therefore important to weight carefully the pros and cons of the different options and try and convince both NPT and non-NPT States of the necessity of further enforcing their non-proliferation efforts.

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1. What are the fundamentals of the non-proliferation regime?

I would like first to apologize for writing the obvious, but the non-proliferation regime relies on two separate concepts.

1) On the upside end of any trade in nuclear sensitive technologies, the exporting part has to comply with certain guidelines (namely the NSG guidelines) which prevent them from exporting too sensitive technologies to non-reliable counterparts;

2) On the downside end, the receiving country has its installations under IAEA safeguards: therefore, the Agency should be able to assert whether the country is trying to acquire or develop military nuclear capabilities.

The first part of this assertion/verification system is highly dependant on the integrity of the exporting companies, on the ability of the licensing officers and on the quality of the classified information gathered by secret service. More specifically, let's imagine country A is developing a military nuclear program and needs, per say, specific filament winding machines. Those winding machines are produced by company X in country B. Country A cannot and will not directly ask company X to deliver the machine to A's Ministry of Defense. Therefore, A will use either an existing civil company Y, whose activity might include the use of winding machines, or create a factice company Z. The contract and the exportation license become thus an inter-company issue.

For this matter to be properly resolved, the following steps have to be respected:

1. a Company X refers to its administration (country B) and asks for an exporting license;

Or

1. b Country B's Customs stops the exportation if X has forgotten (conveniently or not) to ask for the license;

2. Country B's secret service notices that Z is a fake company

3. Country B's licensing officers and experts try and understand how this winding machine could be of any use to country A's specific nuclear program – which supposes country B's secret services have warned the administration against the existence of such a program – and if the declared use, quantities and so forth are plausible. For instance, are the activities of real company Y compatible with the specific characteristics of the winding machine? Is the machine price realistic if Y's activities are low-margins?

Answering all these questions and trying to find something that seems odd or out of picture implies a high level of expertise in very specific, sometimes weapon-related, topics.

Also please note that, as it is an interaction between information gathering, scientific, technical and financial expertise, this is definitely not a perfect science: this means that some exportations can pass through the whole export control system without being noticed.

The second part of the control lies within the status of the importing country towards the Agency. As the NSG guidelines specifically precise, NSG members shall not export nuclear goods or technologies to countries which have not enforced a safeguard agreement with the IAEA. If all nuclear installation of a non nuclear weapon state are covered by a safeguard agreement, risks that one installation shelters an unauthorized nuclear program are strongly minimized.

2. What are the issues this regime is currently facing?

Unfortunately, the situation is never as simple as it seems: recent events have proven that countries that are not outside the NPT have developed nuclear weapons ambitions. Those attempts, whatever their results have been or will be, have revealed a few weaknesses in the system. In short, before listing these weaknesses more precisely, let us recognize that the non-proliferation community has to deal with multiple aspects of the problem. In addition to the well-known diplomatic issues, any restriction in trade has financial and sometimes jurisdictional impacts, and we have already named a few of the technical difficulties – for instance, how to develop in every country an expertise capability without giving too many proliferating details on the composition of a weapon or for instance, technical procedures towards an enrichment plant?

More specifically, new issues have spread out recently:

- The AQ-Khan network case has proved to the world that NSG-states do not detain the monopoly of sensitive technologies and of the ability to export them; an obvious step has been taken as some of the countries outside the NPT have elected to adhere to the NSG guidelines, but this clearly doesn't tackle the issue of non-state actors.

- As the listings (trigger list or dual-use list) become always more accurate and exhaustive, the discrepancy between civil items and items that are specifically designed and prepared for a forbidden use is always thinner. Moreover, it is nearly impossible to provide more specific definitions or to widen the listings without blocking huge pans of the industrial activity and dramatically increasing the number of files to be examined by the licensing officers.

    - Take the UF-6 resistant valves as an example: we will never find a valve "specifically designed or prepared for a UF-6 environment use"; it will merely be labelled (if we are lucky): "resistant to extreme corrosion". So, without any further due diligence, it is impossible to know whether the valve will indeed be used in an enrichment plant or in any pharmaceutical, agro-alimentary or petrochemical plant. And it is impossible to screen any exportation file that includes a "resistant to corrosion" valve.

- Concurrently, a massive legal issue is blossoming: an increasing number of trials are taking place in NSG member States: some industrial do not understand nor accept the restrictions on trade of sensitive technologies to NPT States. They feel, and they are right to feel so, that as long as international law authorizes such transfers, as long as the importing country has its installations under safeguards agreements, they do not have to wear alone the burden of non-proliferation.

    - So let's take an example: a company wants to export good grade alumina to a country known for its willingness to build centrifuge plants. If there were only one dual-use item, then alumina would be it! The importing country pretends it needs the top-grade alumina for its automotive industry and there is absolutely no way to prove it wrong. If the NSG state in which the company is settled refuses the exportation license, the company will attack this refusal, and under any probability, win.

- It is also essential to recall that decisions to export any sensitive good or technology is a sovereign right. In the case of dual-use items, if some countries have already refused the exportation, based on suspicions or doubts about the final user, another country who doesn't share the same views can allow the very same exportation of the same product.

Therefore, in many cases, the international community has to rely on IAEA safeguards in the destination country to make sure there will be no improper use of the transferred technology or equipment. This explains why there is a trend towards more integrated safeguards and a renewed pressure on all the States that have not signed an Additional Protocol yet. However, issues remains, even with States under full scope safeguards:

- As IAEA Director General M. El-Baradei observed in *Towards a Safer World* [1], any state with a fully-developed fuel cycle capability could, if it were to break away from its non-proliferation commitments, produce a nuclear weapon in a few months. This comes from NPT's article IV: in short, no activity, no matter how close it could bring a country to the knowledge of nuclear weapon use, can be forbidden, as long as the activity takes place under the frame of IAEA safeguards;

- There always is a possibility for a country to conceal from IAEA some installations, which can be duplications of civilian facilities that are under IAEA safeguards;

- as recent events have shown, the IAEA inspections can be somehow insufficient to assess with certainty that a State is not, in a facility currently placed under safeguards, working for military purposes;

While restrictions on trade of nuclear sensitive technologies provide but little help to prevent the scenario expressed by M. El-Baradei, they could prove to be of great use in the two other cases.

3. *What are the most commonly discussed options?*

Taking into account all these issues, the international community has to further strengthen the interaction between the IAEA safeguards and the trade for nuclear sensitive technologies, especially the ones linked to the fuel cycle most sensitive parts – enrichment and reprocessing. Yet the method is still subject to debate.

Three kinds of options have been proposed. We are going to present these options, with our perception of their pros and cons.

1) <u>Conditional options</u>

This range of options has already retained the interest of many countries. It basically comes down to submit to IAEA-related legal conditions the transfer of any sensitive technologies. Then any kind of criteria could be considered: the ratification of an Additional Protocol, the full-scope safeguards system, etc.

One major idea behind this is to reconsider the “all for nothing” rule that applies when dealing with the access to civilian nuclear energy. Thus this access would not be automatic any more and would very much depend upon the behavior of the country. Certain violations of the safeguards for instance would be punished by a restriction of the fuel cycle access rights.

Obviously, this method has three huge advantages:

- First, it does not invalid the non-proliferation regime as it is. The export license system remains into force and the integrated safeguards system powers into implementation.

- Second, new international rules clarify the situation towards companies that are willing to trade with other countries: if the stricter conditions are not there, it then becomes useless to ask for a license and moreover to file against the refusal.

- It avoids most of the technical and somewhat hazardous questioning about the plausible and real use of the material or the technology.

On the other hand, four major obstacles remain:

- First, this kind of clause does not tackle the issue of outside-NSG states or organizations.

- Second, as the process now avoids most of the technical issues and questions, it relies mostly on the efficiency of the IAEA safeguards and Additional Protocol. If a country with nuclear weapons ambitions succeeds in concealing its installations from the Agency, those will remain unseen.

- Third, this option obviously raises the issue of States that fall into non-compliance after having acquired fuel cycle-related facilities;

- Fourth, and more fundamentally, some countries will see this option as an unacceptable restriction of their right to access civilian nuclear technology, as stated in the Article IV of the NPT.

2) <u>Control-related options</u>

This range of options tends to reinforce the link between the trading control procedures and the activities of the Agency. One simple way of doing this – simple in terms of explanation, not in terms of acceptability! – would be communicate to the IAEA a list of all technologies and goods transferred. This is already the case for some items of the trigger and dual-use lists for States that have implemented their Additional Protocol.

In short, for the exporting country, the procedure remains the same, with a comprehensive analysis of all exportation licenses that deal either with sensitive technologies or sensitive destinations. But, exactly as the refusals are communicated to the IAEA (on an individual and voluntary basis), the information related to any transfer of a technology or a good of the INFCIRC 254 (parts 1 & 2) would be communicated to the Agency – excluding the price, but including the name of acquirer, quantity, declared destination and use.

Therefore, whenever the IAEA has also signed an Additional Protocol with the destination country, the Agency would be in position to inspect without delay or restriction the facility to which the equipment has been shipped, to ensure the declared use is the real one.

This solution adds a couple advantages to the first option:

- First and foremost, it creates the missing link between the two aspects of the control;

- Secondly, the IAEA gets the complete list of the sensitive materials transferred to a specific country, therefore becoming able to better figure out the existence of a forbidden program, if relevant.

But this option also comes with massive quandaries:

- It is only applicable in countries which have enforced an Additional Protocol;

- It doesn't solve the problem of parallel acquisitions – through clandestine networks or through non-NSG states;

- To reach the desired efficiency, the number of IAEA's inspections is bound to explode and create therefore a financial problem;

- IAEA's inspections would also become excessively intrusive, from the point of view of a sovereign state;

- Last but not least, a lot of countries are not ready to share with the IAEA their decisions about exportation licenses – an agreement has merely been reached about the refusals; a similar agreement on the acceptations is extremely unlikely to take place.

3) <u>Strategic options</u>

This third range of options deals with international consortiums for fuel supplies; the trade of sensitive technologies would then be towards a multilateral commercial consortium or towards an international organization such as IAEA.

Therefore, any transfer of a trigger list item would be restricted to a registered international consortium or to the Agency. Every State would have an unlimited access to the civilian nuclear energy, but no access to any of the sensitive parts of the fuel cycle – enrichment and reprocessing.

This option seems appalling first, as it allows the international community to make sure that the non-proliferation regime is enforced without multiplying the inspections or the trading procedures. It would also give to a lot of countries the possibility to access the use of civilian nuclear technology without having to develop all the infrastructures for the fuel cycle, infrastructures a lot of States cannot handle nor afford. Lastly, this option would also simplify some considerations about terrorism, as the number of eligible targets would remain limited and easier to defend.

But the cons are obvious:

- It substantially transforms Article IV of the NPT. Obviously, there would be no possibility to deny individual States the right to master some fuel cycle steps; this could only happen on a voluntary basis. But there could be an interdiction to export goods related with these sensitive technologies, interdiction that would be assimilated as a restriction of Article IV.

- This option deals with a strategic issue: the guaranties of supply. How to create a system that makes everyone confident that they will have access to fuel, no matter what happens geopolitically or financially? How to convince – and on which ground could they be convinced – certain States, which feel that they could develop their own fuel cycle and become alternative industrial players?

- It does not prevent some States from developing clandestine facilities;

- On an industrial basis, it creates an oligopoly of nuclear actors, which is beneficial neither to the investments and developing of better technologies, nor to the final customers.

4. Discussion – Prospective

Seeing the different aspects of the question and the pros and cons of each option proposed, it is still possible to draw a few conclusions.

Firstly, any improvement in terms of controlling the trade of nuclear sensitive technology will translate into an increase of the practical difficulties for a country to acquire some sensitive technologies and will be felt by some States as a restriction to the "Atom for peace" equation. This might not be a deal-breaker but will require a lot of diplomatic discussions to be acceptable to every State.

A second – sadder – conclusion is that, no matter which enforcement of trade control is chosen, it will not completely settle the issue of clandestine facilities. The non-proliferation will thus still be highly dependant on intelligence information.

A third conclusion, more positive, is that recent events have now (or should have) convinced everyone of the utility of increasing the quantity and improving the quality of the controls of sensitive trade of goods and technologies. While always more accurate lists are published and exchange group meetings take place to ensure that all the experts are able to better acknowledge the risks of certain exportations, renewed pressure is put on countries that haven't ratified yet an Additional Protocol. The universality of such an Additional Protocol, including the non-NPT States would be a huge step forward in terms of non-proliferation.

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Analysis of proliferation networks*

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Abstract. The analysis of all available relevant information constitutes an essential part of the evaluation of the correctness and completeness of a State's declarations. In response to the proliferation risks from covert networks that trade in sensitive nuclear technology, the IAEA has strengthened its focus on clandestine nuclear related supply and procurement. Actions include the creation of an interdisciplinary nuclear trade analysis unit whose specific task is to coordinate the analysis of all relevant information. New work procedures and software tools have been developed to support the processing, analysis and storage of nuclear trade-related data. The activities and progress of this initiative are described in this paper, as well as its role in the safeguards State evaluation process. The process by which Member States can support the IAEA by providing relevant information and expertise on a voluntary basis is also described.

1. Introduction

The recent investigations into clandestine nuclear programmes have revealed the existence of covert nuclear trade networks that have assisted a number of countries in developing their nuclear capabilities. Over the last couple of decades, these networks have provided sophisticated nuclear equipment and technology to their customers. Their existence constitutes a serious threat to the nuclear non-proliferation regime and a new verification challenge.

Responding to this challenge, the IAEA Department of Safeguards is stepping up its efforts to understand the nature of such covert operations by analysing information on nuclear trade activities.

The results of these analyses are part of the safeguards State evaluation process, whereby the IAEA monitors overall consistency of a State's declarations with its past, present and planned nuclear programme, verifies the non-diversion of declared nuclear material and looks for indications of undeclared nuclear material and activities.

2. Clandestine nuclear trade-related proliferation risks

The challenge posed by clandestine procurement networks requires a new type of effort by the IAEA. The IAEA must perform an analysis at the trans-national level, noting in particular the entrepreneurial role of entities involved in covert nuclear trade.

This involves, above all, the analysis of information already available to the IAEA through States' declarations under their safeguards agreements and additional protocols. Voluntary undertakings, such as the Voluntary Reporting Scheme, provide the IAEA with useful information related to the export and import of nuclear material and specified equipment, and non-nuclear material. But the analysis at

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the trans-national level is also predicated on the assumption that not all nuclear-trade related activities within or passing through the territory of a State may be known to the State’s authorities. Actors involved in ‘black market’ activities may deliberately evade national export controls and other regulations, and effectively conceal clandestine shipments within the enormous and growing volume of legitimate global trade. The IAEA is continuing the process of developing the conceptual basis and information requirements for this type of analysis. A central dilemma is distinguishing the miniscule fraction of safeguards-relevant covert nuclear trade from the vast quantity of legitimate international commerce resulting from globalization.

3. Contributing to State evaluations

A number of analytical processes have been developed to contribute to the nuclear trade analysis. These include, among others, ‘micro’ studies of specific procurements, ‘mid-level’ assessments of potential suppliers in key technology sectors and ‘macro’ reviews of the evolution of clandestine nuclear networks.

Clandestine transactions which have come to the attention of the IAEA, whether from news reports, legal documents regarding criminal violations of export control laws or other sources, are analysed in depth. This analysis covers both technical dimensions and network relationships, assessing their potential relevance to possible undeclared activities in States under safeguards. Suspected clandestine attempts to procure or to supply nuclear-related goods are analysed for consistency with a State’s declared nuclear programme. If a possible inconsistency is identified, then clarification by the State may be required. Where such transactions involve private middlemen in procurement or supply activities, these can be considered as possible sources of supply from, or to, other States. The reported role of some private actors in supplying the undeclared programme in one State as well as nuclear programmes in other States is a significant example that underscores the importance of such tightly focused, micro-level analysis.

Analysis of open-source information on a State’s nuclear-related industrial capabilities is part of the normal State evaluation process, both in the routine evaluation of a State’s declarations and in reaching broader conclusions on undeclared nuclear material and activities. While the primary purpose for evaluating information regarding industrial sectors is to check for consistency with a State’s declared nuclear programme, a benefit of increasing significance is in assessing potential export of nuclear-related items. Such mid-level assessments can aid in providing context for the analysis of nuclear trade in such commodities as maraging steel.

Publicly available information regarding the nuclear network involved with one State’s undeclared nuclear programme indicates a substantial evolution in its activities over time. Fundamentally, a network created for clandestine nuclear procurement was transformed into one engaged in clandestine supply. Other changes have occurred in the context of relevant international trade, most importantly in the development and implementation of nuclear-related export controls. Cataloguing the range of tactics that have been employed is also important in considering how national export controls may be circumvented. In addition, a State’s declarations may not be accurate or complete due to a lack of awareness on the part of State authorities. Through macro-level analysis of the historical, regional and functional transformation of this network, an assessment can be made of its potential involvement for clandestine acquisition in undeclared nuclear programmes in all States where the network is known or suspected to have had contacts.

4. Responding to the threat

The Nuclear Trade Analysis Unit (NUTRAN) was established, in November 2004, within the IAEA Department of Safeguards to fill a gap in comprehensive analysis of information related to covert nuclear trade. The unit has currently seven members with broad experience covering nuclear technology, nuclear trade, custom controls and safeguards. The unit members also have in-field experience in nuclear trade analysis related activities conducted previously by the IAEA’s Iraq Nuclear Verification Office.

The unit centralizes the analysis of information related to trade in sensitive nuclear technology as well as of data on known and potentially undetected networks, using all procurement network-related information available to the IAEA.

Open source research and analysis on procurement network activities is carried out by the Information Analysis Unit (IAU) of the Department of Safeguards. The IAU monitors a wide range of information sources for new reports related to nuclear trade and conducts in-depth research and analysis on selected issues. This research complements the work conducted by NUTRAN, and since it is based on open sources it can be more widely distributed within the Department.

NUTRAN also maintains an institutional memory on activities related to covert nuclear trade. This includes collecting and storing source documents, extracting and storing relevant information for analysis, and providing complementary information to support on-going State evaluations.

Figure 1 schematically shows how the IAEA's institutional memory on nuclear trade analysis is built up. It also shows how common terminology is used in this context.

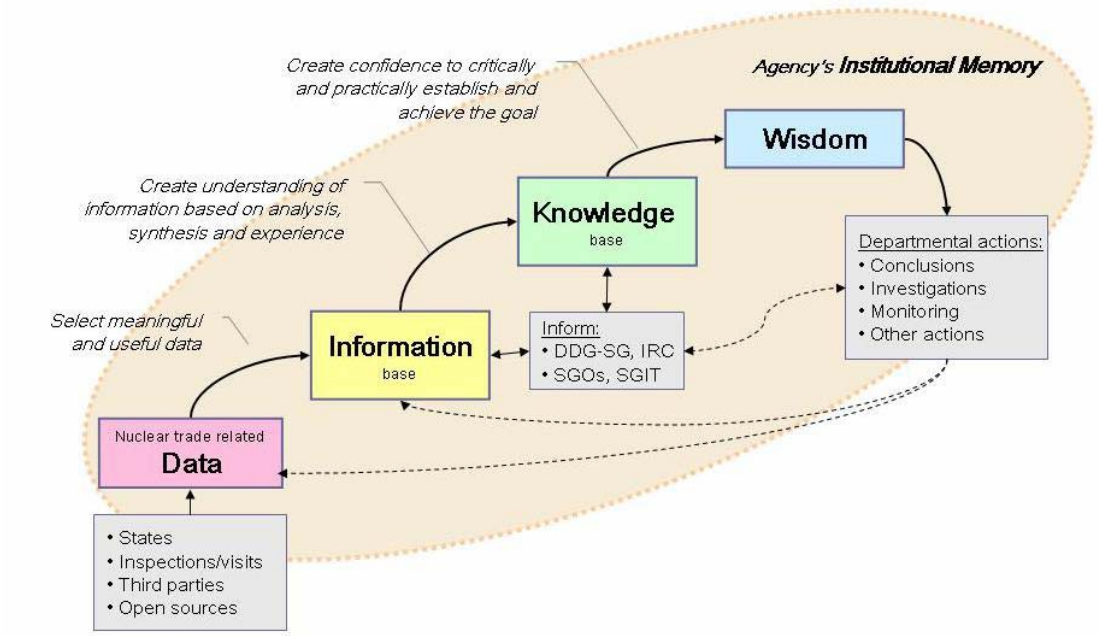

FIG. 1. IAEA's institutional memory on nuclear trade analysis.

Raw nuclear trade-related data become available in different forms and through different channels. A number of data processing steps are required to convert raw data into a form that can be used for analysis, either currently or in the future, including registration for reference and traceability, conversion to a standard format electronic file or indexing its content.

Two mechanisms are used to store indexed files. All files are stored in an information base, which is a folder structure based on topical or geographical content. The information base allows text-based searching and retrieval of documents. The second mechanism is a specially created Procurement Tracking System (PTS) in which suspicious nuclear trade-related transactions and events are stored. It holds analytical data, cross-references source material, combines new data and historical knowledge on covert nuclear trade and reveals any links or procurement patterns between old and new data.

Nuclear trade analysis aims at recognizing the presence of covert trade activities related to nuclear goods or services. A primary target of the analysis is to identify traders and middlemen involved in undeclared nuclear trade related activities. These may be linked to one or more States, either as end-users or through transport or transhipment.

5. Procurement outreach to nuclear related industry

In 2005, and again in 2006, the IAEA General Conference invited all States to cooperate with the IAEA in its efforts to verify and analyse information provided by Member States on nuclear supply and procurement[1] The IAEA Secretariat is reaching out to States that might be willing to help the Secretariat in this regard, with a view to gaining access to information that could strengthen the IAEA safeguards system. The provision of such information to the Secretariat is done entirely on a voluntary basis and is handled with the highest level of confidentiality.

The IAEA procurement outreach initiative is based on the premise that entities related to covert networks are likely to leave visible traces, as they try to acquire nuclear-related goods and services on the open market. Procurement outreach is designed to acquire access, with the agreement of the States concerned, to such traces.

On this basis, the IAEA intends to agree on appropriate modalities with Member States for access to safeguards relevant trade-related data that is not normally available to the IAEA. It should be underlined that such information is sought in order to support the IAEA's nuclear verification mandate, which does not encompass export control per se.

Companies in selected business sectors would be encouraged to watch for procurement enquiries received from entities seeking to acquire goods that might be included in a nuclear programme. It is a combination of several features, each innocuous in itself, that identifies a suspicious enquiry - not all features are related to the goods being sought.

Information received, combined with appropriate analysis, can provide early indications of attempts to circumvent States' safeguards and other nuclear non-proliferation undertakings.

6. Conclusion

The response to the requirements expressed by the IAEA Board of Governors and the General Conference concerning analysis of covert nuclear trade has resulted in further strengthening of the safeguards system. Due to the large volume of nuclear trade-related data and the understanding that most of such trade is legal commercial activity, the IAEA has focused its activities on the analysis of information related to the trade in sensitive nuclear nuclear technology.

Over the last year, the IAEA has endeavoured to put into place an infrastructure to respond to the challenges caused by trans-national procurement/supply networks. But work remains in progress. The IAEA increasingly needs the voluntary support of Member States in the form of information and collaboration in developing tools and methods. This is essential to address the safeguards challenges caused by what the IAEA Director General has characterized as the most dangerous phenomenon we have seen in the non-proliferation area for many years.

[1] In paragraph 21 of resolution GC (49)/RES/13, adopted on 30 September 2005, and paragraph 24 of resolution GC(50)/RES/14, adopted on 22 September 2006, respectively, the General Conference "welcomes efforts to strengthen safeguards, including the Secretariat's activities in verifying and analysing information provided by Member States on nuclear supply and procurement, taking into account the need for efficiency, and invites all States to cooperate with the IAEA in this regard".

The role of quality management in delivering safeguards conclusions

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Abstract. Drawing safeguards conclusions regarding the absence of undeclared nuclear material and activities is intrinsically more challenging than verifying the non-diversion of declared nuclear material. Important new measures are available to the IAEA to help in this respect. However, the Secretariat can only have confidence in the soundness of its conclusions, and thus provide credible assurances of States' compliance with their safeguards obligations, if the necessary and sufficient processes and tools are in place, if the processes are carried out correctly and adequately monitored and if appropriate feedback loops are used. It is simply not possible to carry out quality control activities to verify the absence of undeclared nuclear material and activities. Part of the approach being taken by the Department of Safeguards to meet this challenge is to implement a Departmental quality management system (QMS) based on the ISO 9001:2000 standard. This paper describes the steps being taken by the Department to implement such a system.

1. Introduction

How does an organisation assure the quality of its 'products' when such products are conclusions regarding the absence of undeclared nuclear material and activities or the resulting assurances of States' compliance with their safeguards obligations? Such conclusions cannot be quality controlled in a traditional sense, like manufactured components, and unlike a modern service industry the 'customer' is not able simply to choose another service provider if it perceives that the products are of poor quality, however it may define quality.

This is the kind of challenge that the IAEA has always faced and never more so than now, as an increasing number of States have concluded additional protocols to their comprehensive safeguards agreements (CSAs). The safeguards measures provided for in additional protocols equip the IAEA with supplementary measures that enhance its ability to detect undeclared nuclear material and activities. Safeguards conclusions are no longer drawn solely at the facility level, based on assessments against the well-defined safeguards criteria, but are drawn at the level of a State as a whole. The framework in which this takes place is the State evaluation process. State evaluations take into account a much wider range of safeguards-relevant information than previously available to the IAEA and reflect the increasing move away from purely quantitative to more qualitative assessments.

Drawing conclusions regarding the absence of undeclared nuclear material and activities is intrinsically more challenging than verifying the non-diversion of declared nuclear material. The requirement from Member States that the broader conclusion — that all nuclear material remained in peaceful activities — should be soundly based and thus credible obliges the IAEA to study very carefully how it is derived. The familiar evaluation methods used in traditional safeguards when dealing with declared material are no longer capable of delivering the necessary confidence in this broader conclusion.

Against the background of a fundamental change in the way safeguards conclusions are drawn, the Secretariat needs to ensure that it has the necessary and sufficient processes in place and that the processes are well understood, carried out correctly by staff with the right skills and competencies and provide appropriate feedback loops. All of these are important if the conclusions drawn under strengthened safeguards are to be soundly based. To help meet this challenge, the IAEA is implementing a quality management system (QMS) in the Department of Safeguards, which is based on the ISO 9001:2000 standard[1].

2. Why a quality management system?

The need for a fully-fledged and comprehensive Departmental QMS was recognised internally by the IAEA in light of the changing safeguards environment and the increasing move away from purely quantitative to more qualitative assessments. It was also recognised by a number of other 'stakeholders' such as Member States and the Director General's Standing Advisory Group on Safeguards Implementation (SAGSI). More recently, the need for implementation of IAEA-wide quality management practices was reflected, *inter alia*, in the IAEA Medium Term Strategy for 2006-2011.

The ISO 9001:2000 standard was chosen because it is an international system, it is widely accepted and used and, more importantly, it is sufficiently flexible to enable the Department to introduce a system appropriate to its needs.

The ISO 9000 series of standards is based on eight quality management principles that can be used to lead any organisation towards improved performance. The principles are: customer focus, leadership, the involvement of people, a process approach, a system approach to management, continual improvement, a factual approach to decision-making and mutually-beneficial supplier relationships.

The ISO 9001:2000 standard fits into a process of Plan-Do-Check-Act (PDCA) in order to deliver the product, whatever that product might be. The essence of the PDCA approach is: **plan** what you will do, **do** what you planned, **check** that you actually accomplished it and **act** on any perceived gaps between what was planned and what was accomplished and then restart the cycle at the plan phase.

The adoption of the ISO standard and its use as the framework for the Departmental QMS are expected to yield many benefits. Key outcomes envisaged by the Department include improved efficiency and effectiveness, more widespread sharing of knowledge and experience and a greater ability to manage change and continually improve performance.

3. Beginnings

Preparatory work to create awareness of the need for a Departmental QMS and to establish a quality culture was carried out through training courses funded by the United States Support Programme to IAEA Safeguards (USSP). The Departmental QMS was formally initiated on 23 November 2004 when the Departmental Quality Policy Statement was issued. Its broad aim is "to enhance the efficiency and effectiveness of the IAEA's verification and evaluation activities through continual improvement".

The first step towards implementing the Departmental QMS involved a gap analysis to identify areas where the Department did not fully meet the requirements of the ISO 9001:2000 standard. The analysis was carried out in the first part of 2005. It involved the review of available documentation and interviews with staff members throughout the Department. The analysis also took into account previous work to establish localised management systems within individual areas of the Department. Completion of the gap analysis led to the development of a project plan designed to address the areas of non-conformance with the ISO 9001:2000 standard. Priorities for the plan were established by evaluating the required tasks against criteria such as: (a) the task's impact on increasing the effectiveness of the IAEA verification and evaluation activities, (b) whether the task was a pre-requisite for carrying out other activities, or (c) whether it addressed a critical or particularly important issue.

Organisational support for the Departmental QMS was also established. It included the formation of a team within the Safeguards Division of Concepts and Planning to lead the project and the appointment of a Departmental quality manager and of quality managers from each of the Divisions in the Department. The responsibilities of the quality managers include: (a) ensuring that processes needed for the QMS are established at the Division level, (b) acting as a focal point for quality management activities, (c) providing appropriate two-way communication between the Division and the wider Departmental QMS project, and (d) ensuring that activities undertaken within a Division are consistent with the implementation of the Departmental QMS. The quality managers meet on a monthly basis to review implementation of the QMS.

4. Moving forward

Achievements during the early stages of QMS implementation include: (a) the delivery of training and the development of an internal website as a communication tool, (b) mapping Departmental processes, (c) agreed arrangements for continual process improvement (CPI), (d) development of a document control system, and (e) development of an internal quality audit system.

An important aspect of implementing the QMS is ensuring that staff members are aware of their role in the system. To achieve this, a number of communication media have been developed or are planned. The QMS website is accessible to all Safeguards staff members via the Department's Intranet. It is updated regularly and provides information on the status of the QMS implementation as well as background and reference information.

A more interactive forum is provided by the 90-minute presentation "Introduction to quality management in the Department of Safeguards". The training was originally planned for new staff members, but it was realized that all staff members would benefit from an understanding of the QMS. Attendance is now mandatory for all staff members and a series of 30 sessions is currently underway to ensure that all staff members have the opportunity to attend before the end of 2006. One of the benefits of this training is that it provides direct feedback on the implementation of the QMS and on staff members' perceptions regarding the Quality Policy.

Information for new staff members will continue to be made available in this training, but during 2007 it will run at a reduced frequency as determined by recruitment rates. To complement this, a computer-based learning package is being developed with assistance from the Canadian Support Programme. In addition, the Department offers a five-day QMS Workshop and a two-day QMS seminar for safeguards managers. These are provided with assistance from the USSP.

The ISO 9000 series of standards considers an organisation as a system of interlinked processes that need to be managed, rather than viewing the organisation in a strict hierarchical manner and managing its individual functions. This change in emphasis is an important aspect that will need to be embedded in the Department's quality culture. Departmental processes have been identified and process mapping has commenced. The process maps and descriptions derived will be used as the basis for continually improving processes as well as for ensuring that across the Department there is uniform understanding of how the Department carries out its core activities and what knowledge and skills are required for those activities. Process maps and descriptions will also provide the basis for gathering user requirements for the IAEA Safeguards Information System Re-engineering Project (IRP), and close collaboration is being maintained with the IRP team to align needs. At a later stage in the implementation of the QMS, appropriate measures will be put in place to monitor performance of key processes.

Departmental arrangements for CPI have been developed, based on the DMAIC (Define-Measure-Analyse-Improve-Control) approach. This approach is a well-known business improvement method and forms the basis of the six-sigma methodology used by many organisations worldwide. The development of these Departmental arrangements also made use of the recently issued IAEA publication on the management of continual improvement[2]. The aim of the CPI activities is to increase the Department's ability to meet the requirements of internal and external customers and to adapt when those requirements change.

A document control system has been developed to ensure that correct information on how to carry out specific tasks is available to those staff members that need it, when it is required. Clear rules have been defined for determining whether a document needs to be controlled, thus ensuring that the number of documents is commensurate with the needs. In addition, the introduction of a standard document template allows users to quickly identify the status of a document. A simple approval process for the creation and use of documents has also been introduced with four defined steps: (a) writing documents, (b) verifying that the described task/procedure is fit for purpose, (c) confirming that the document complies with QMS guidelines, and (d) authorising implementation.

Within the ISO 9001:2000 PDCA cycle, one activity within the check phase is carrying out internal quality audits to determine whether the working practices agree with the planned arrangements, to assess compliance with the ISO 9001:2000 standard and to identify the strengths and weaknesses of an QMS process or activity. During 2006, internal quality audits have been carried out in a number of support areas, including seals, the provision of instrumentation and software development. Experience gained during these audits has been reviewed and arrangements for audits have been improved. Furthermore, training is being developed to enable more staff members to participate in the audit process. The internal quality audit process will now be deployed across main Departmental process areas such as inspections and design information verification.

In addition to the monthly review of implementation by the Division quality managers, the implementation of the QMS is also reviewed by the Department's management team on a regular basis. The Deputy Director General of Safeguards leads this formal management review of the QMS.

5. The way ahead

The introduction of the QMS within the Department of Safeguards is not a one-off event but represents a new way of thinking and working. Constructive first steps have been taken and many activities remain to be accomplished. The early stages of implementing the QMS require that the necessary processes and systems be put in place, and work throughout 2006 and 2007 will continue with this. Important areas include: (a) incorporating existing documents into the document control system, (b) developing a corrective action system, (c) implementing process measurement, (d) aligning staff competence and training systems with ISO 9001:2000 requirements, and (e) putting in place a Department-wide record control system. This work will be pursued, in addition to the maintenance of those activities already implemented.

The implementation of a corrective action system is already underway. This system provides a methodology, tools and techniques for identifying nonconformities, determining the causes and implementing corrective action to avoid recurrence. Included within the system are requirements for the related recording, tracking and reviewing of the effectiveness of actions. The intent of the system is to improve the quality of the Department's activities by engaging the participation of all Safeguards staff members.

6. Conclusions

Drawing conclusions regarding the absence of undeclared nuclear material and activities is intrinsically more challenging than verifying the non-diversion of declared nuclear material. A wide range of new tools is available to the IAEA to contribute to this process. However, the Secretariat can only have confidence in the soundness of its conclusions, and thus provide credible assurances of States' compliance with their safeguards obligations, if the necessary and sufficient processes are in place, if they are carried out correctly and adequately monitored and if the appropriate feedback loops are used. It is simply not possible to carry out quality control activities to verify the absence of undeclared nuclear material and activities.

Part of the approach being taken by the Department of Safeguards to meet this challenge is to implement a Departmental QMS based on the requirements of the ISO 9001:2000 standard. Many interlinked activities are underway to achieve this. The transformation to working within an QMS is not simply a mechanistic activity of documenting the activities of an organisation but requires a fundamental change in the organisational culture and the adoption of quality management principles. Implementing the QMS will better enable the Department to understand what it does, why it does it and how it is done in a sustainable manner that allows it to adapt to the continually changing safeguards environment.

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Hexapartite safeguards project: A retrospective

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Abstract. The Hexapartite Safeguards Project (HSP) was created in November 1980 as an international forum consisting of technology holders of gas-centrifuge enrichment and the international safeguards inspectorates in order to develop a strategy for applying effective and efficient safeguards to a commercial centrifuge enrichment plant without compromising the sensitive information related to centrifuge enrichment technology [1]. The participants were Australia, Japan, the so-called 'Troika' states, i.e. the Federal Republic of Germany (now Germany), the Netherlands and the United Kingdom, the United States, EURATOM and the IAEA. They met for some two years, producing in February 1983 a final report with an agreed strategy of the "LFUA" (Limited Frequency Unannounced Access) Model [2].

Its basic feature is to allow safeguards inspectors routine access to cascade halls with short notice and an agreed average frequency. This model has served as the basis for enrichment plant safeguards in HSP countries with necessary modifications and adaptations reflecting the specific features of individual plants. However, more than two decades have passed since the conclusion of the HSP and it has become necessary to develop new safeguards approaches reflecting the recent developments in enrichment and safeguards technology, particularly the advent of large-scale facilities. SAGSI (Standing Advisory Group on Safeguards Implementation) recently reviewed and approved the Agency's proposed approach.

The author, served as the Secretariat of the HSP in charge of drafting its final report. This paper provides an outline of the outcome of the HSP: (i) in a historical perspective with an emphasis on the issues that helped define the HSP deliberations; (ii) how they were addressed in the LFUA model; (iii) the significance of the HSP in the context of international safeguards; and (iv) its limitations due to prevailing legal interpretations at that time.

Disclaimer: The views expressed in this paper represent those of the author and should not be construed as representing those of NMCC, nor the Japanese Government.

1. Formulation of HSP

International Nuclear Fuel Cycle Evaluation (INFCE), initiated by US President Jimmy Carter in October 1977, reaffirmed that the peaceful uses of nuclear energy including enrichment and reprocessing can proceed if effective measures are taken to minimize the danger of nuclear proliferation. The communique of the Final INFCE Plenary Conference contains, among other things, the following:

> " (T)he findings of INFCE have strengthened the view that ……. effective measures can and should be taken to minimize the danger of the proliferation of nuclear weapons without jeopardizing energy supplies or the development of nuclear energy for peaceful purposes."

In this context, "effective measures" meant, *inter alia* safeguards methods and technologies as well as legal and institutional arrangements. A different forum was required to consider how to safeguard reprocessing, uranium enrichment, and other sensitive plants. Prior to and after the INFCE studies, the IAEA organized a series of technical meetings such as the International Working Group on Reprocessing Plant Safeguards (IWG-RPS), a consultants meeting on safeguarding uranium enrichment plants, and an Advisory Group meeting on MOX fuel fabrication plants in order to consider how to effectively and efficiently safeguard these sensitive facilities.

Meanwhile, activities were initiated in early 1970s for the construction and operation of commercial gas-centrifuge enrichment plants. In August 1971, the Federal Republic of Germany (FRG; now Germany), the Netherlands and the United Kingdom established a trilateral consortium, URENCO, for commercial uranium enrichment using gas-centrifuge technology. However, there was no international consensus on the issue whether inspector access to cascade halls was indispensable for the effective and efficient safeguards on such facilities because centrifuge enrichment technology was sensitive from both a commercial and a nuclear proliferation perspective. Technology holders that received safeguards inspectors into their facilities maintained that inspector access to cascade halls was not acceptable because there was a risk of disclosure of technical information regarding their centrifuges. Their concern was the disclosure of proliferation sensitive and proprietary information. This was a head-on crash with the Inspectorates, IAEA and EURATOM, which asserted that inspector access was essential.

Under such circumstances, there was no inspector access to cascade halls and safeguards on the then operating commercial gas-centrifuge enrichment plants included an ad hoc inspection regime provided for in paragraph 71, INFCIRC/153, with the understanding that this would not prejudice the future safeguards system for such plants.

By the later half of 1970s, plans for commercial scale gas-centrifuge enrichment plants were contemplated or announced in the USA, Japan and Australia. Under the strong initiative from the USA, technology holders and the Inspectorates were urged to formulate an international consensus on how to safeguards these facilities. After extensive informal consultations among the six parties, i.e. Australia, Japan, the URENCO states, the United States, EURATOM and the IAEA, there emerged a consensus on the form of collaboration. The HSP was established during the initial meeting convened at URENCO headquarters in Marlow, UK, in November 1980.

2. Organization of HSP

The word “Hexapartite” stems from the fact that the Project consists of the six parties as described above, i.e. four technology holder parties and two international safeguards inspectorates. The prefix “hex(a)” is of Greek origin meaning “six”.

The HSP was formulated as a project among the six parties, not as one of the IAEA sponsored meetings. This was because any interested Member States can participate in an IAEA sponsored meeting, which cannot exclude the possibility of the disclosure of sensitive information outside the six parties. This was one of the major concerns of the HSP technology holders.

The Marlow initial meeting agreed, among other things, on the establishment of the HSP and its objective and organization as follows:

- Both the technology holders and the Inspectorates will work together to establish within the target time frame of two years the technical basis for facilitating the application of effective and efficient safeguards to commercial gas-centrifuge enrichment plants;

- The project aims for the practical application to actual plants, with a design capacity of up to 2,200t SWU/y, not a conceptual exercise for model plants;

- The project will consider equally the case of inspections without access to cascade halls and the case of inspector access with various frequencies and scope;

- The project is not the place for formulating and negotiating a legal framework; and
- The following four teams will be set up under the Plenary to review specific issues:
  - Team 1: Facility Characteristics (Japan);
  - Team 2: Containment and Surveillance, or C/S (UK);
  - Team 3: Nuclear Materials Accountancy (Australia); and
  - Team 4: Safeguards Strategies Including Different Degrees of Access to Cascade Areas (US).

  Their progress was to be monitored by the Plenary. The country in the parenthesis assumed the chair of the respective Team. Mr. F. Brown, then UK Department of Energy, was appointed as the Plenary Chair.

The subsequent Plenary meetings, six in total, were hosted alternately by each of participating parties as follows:

- Boekelo, the Netherlands, March 1981;
- Germantown, USA, July 1981;
- Tokyo, Japan, November 1981;
- Sydney, Australia, March 1982;
- Aachen, FRG, November 1982; and
- EURATOM Headquarters, Luxembourg, January 1983.

3. Major Issues and Findings

At an early stage of the project, there was an extensive "theological" argument on how to describe one of the diversion scenarios being addressed, i.e. undeclared production of HEU. Some of the technology holders were reluctant to use the word "undeclared". They maintained that such an expression would imply that the State concerned admits the possibility of clandestinely producing HEU or expresses its intention to breach the NPT. This was totally unacceptable to them. On the other side, the Inspectorates argued that, in formulating a safeguards approach, it was a common practice to assume the existence and use of "undeclared" facilities in considering certain diversion scenarios. Retrospectively, it is hard to understand why such arguments occurred but these were days prior to the 1991 discoveries of Iraqi clandestine nuclear programmes, Programme "93+2" and the adoption of the model Additional Protocol. In the end, it was settled by the expression, "to verify that a facility is operating as declared" instead of using the word "undeclared".

Further, it became clear that there were substantial design and operational differences among facilities being considered that were important from a safeguards point of view:

- The facility maintenance policy, e.g. no maintenance and failed machines are not repaired vs. routine maintenance of centrifuge machines;
- The size of an individual centrifuge machine;
- The configuration of centrifuges, i.e. a block mounting consisting several centrifuges as a unit vs. a single mounting method;

- The ease of detection by visual observation of a rearrangement of cascade piping, i.e. whether or not cascade areas are “transparent”;
- Existence of sampling points, isolation valves or valves for the rearrangement of existing piping.

These differences would substantially impact the safeguards methods to be applied.

There was a proposal that inspector access to cascade areas be avoided by placing an extensive number of C/S equipment on the periphery of these areas. When HEU was produced clandestinely, it was argued that the C/S system could detect its removal from the cascade areas, thus not requiring inspector access.

With regard to nuclear materials accountancy, it was shown that PIT/PIVs could be conducted without stopping plant operation, that the amount of gaseous uranium in a cascade is small for those with small-sized centrifuge machines, and that an accurate material accountancy is feasible with conventional methods.

A review of the reports of the four teams at the Tokyo Plenary meeting resulted in the creation of an Assessment Sub-Group, chaired by the FRG, with the mandate to define, assess and evaluate the advantages and disadvantages of the “non-access” and “LFUA” safeguards models.

After a detailed review, the Assessment Sub-Group reached the following conclusions:

- The LFUA model will achieve the safeguards objectives;
- When compared with the “non-access” model, the risk of revealing sensitive information is greater for the LFUA model, but the latter is less intrusive into plant operations and requires less equipment and personnel for both plant operators and the Inspectorates;
- The implementation of the LFUA model is simpler, especially in existing plants or those already under construction; and
- The LFUA model is more credible in demonstrating its effectiveness and the availability of required equipment within the time frame of the HSP.

Further, the Assessment Sub-Group concluded that, in the application of LFUA, its acceptance by all HSP parties was essential as well as its universal application to all technology holder parties of the HSP. It became apparent that the detailed modalities of verification activities were to be defined and that the issue of protecting sensitive information required a solution. The Sub-Group recommended that a detailed review be made on how the safeguards methods based on the LFUA model were to be applied to each individual facility. Its findings and recommendation were reported and endorsed at the Sydney Plenary meeting.

In response to this, each technology holder submitted a document describing its views on the inspection activities and design information that would be useful and acceptable in applying the LFUA model to its specific plant. Based on the review of these documents, the Inspectorates presented to the Aachen Plenary meeting a report discussing the possible inspection activities in cascade areas. After reviewing the report, it was agreed to produce an HSP report acceptable to all the parties, incorporating also the verification activities outside cascade areas.

In parallel, each technology holder offered its facility for the demonstration of various safeguards methods and technologies, which included the integral test of the LFUA model at Capenhurst, UK. A drafting sub-group meeting was held in London, in December 1982, to draft a report reflecting the results of the integral test. The Sub-Group eventually agreed on the content of the report regarding the inspection activities inside cascade halls.

At the Luxembourg Plenary meeting, a report tabled by the Inspectorates, discussing the inspection activities outside cascade areas, was reviewed and approved to be integrated with the above-mentioned Sub-Group report. Thus, the final report was produced in February 1983, with a general description of inspection frequencies, duration, and verification activities that could be implemented inside and outside cascade areas. The final report included descriptions of specific access routes and verification activities to be implemented at each facility. The HSP successfully completed its task on the technical level at the Luxembourg Plenary meeting after approximately two years and three months of deliberations. The HSP Chair presented the above final report together with the "Overview" document describing how the experts carried out their work through the HSP to the Director General of the IAEA [3]. The document excluding the section of the final report containing plant specific descriptions was made public as the HSP report [2].

4. Outline of LFUA Model

In the LFUA model, the following scenarios are postulated for clandestine HEU production:

- Rearrangement of the enrichment equipment; or
- Modifying the operational mode, e.g. by recycling flows or parts of them using alternative feed and take-off points.

The indications of these activities or anomalies that might be observed or detected by inspectors are:

1) Significant variations in UF6 flow or concentration at feed and withdrawal stations. (This includes significant MUF or systematic data falsification);
2) Changes in declared UF6 piping arrangement;
3) Existence of additional storage, feed and withdrawal station/ facilities; and
4) A radiation field indicating the presence of HEU.

For detection of the indications/ anomalies identified in 2) to 4) above, inspectors access to cascade areas is required and the corresponding verification activities are listed.

According to the agreed LFUA model, an average LFUA frequency of 4 to 12 times per year is appropriate for facilities up to about 1,000t SWU/y capacity. The actual number is plant specific, depending *inter alia* on the ease of modifying the facility for HEU production, the time required for the production of a significant quantity of HEU and the time required to remove the resulting anomalies. For facilities with high transparency, verification activities inside cascade areas are mainly visual observations to detect rearrangement of piping. For those with less transparency, the main activities are radiation measurements with portable equipment. The average frequency of routine inspections outside cascade areas was expected to be in the range of 12 to 15 times per year for facilities with capacities up to 1,000t SWU/y.

With regard to inspection activities inside cascade areas, the following restrictions/ requirements were suggested to protect sensitive information:

- The number of inspectors participating in each inspection should be limited;
- The inspectors should be escorted;
- The inspectors may not depart from the pre-determined and agreed paths;
- The instruments and equipment to be used and the modalities of their use by inspectors are to be limited to those agreed upon;

- The duration of the inspection activities may be limited to an agreed maximum time; and
- Access may be delayed by up to 2 hours.

The notification of an LFUA inspection is to be made during a routine inspection at a facility or as an "unannounced" inspection provided for in the paragraph 84, INFCIRC/153. Notification in the course of other activities at the facility is similar to that provided for complementary access in INFCIRC/540.

5. Implementation of LFUA Model

In March 1983, soon after the conclusion of the HSP, an exchange of letters was made between the IAEA and the technology holders, and among technology holders themselves, confirming that the safeguards model contained in the final HSP report shall be applied uniformly to specific plants, either existing or planned, in both NWSs and NNWSs. Negotiation of facility attachments (FAs) or legal instruments for assuring uniform application of the HSP model, was left to the parties concerned.

Accordingly, in the case of Japan, an FA re-negotiation was initiated for the then operating Ningyo Enrichment Pilot Plant and the FA based on the LFUA model was implemented in September 1985. The Ningyo Enrichment Demonstration Plant soon followed suit. Today, the Ningyo enrichment facilities have stopped operation and are currently under or awaiting decommissioning. The LFUA model was incorporated in the FA for the commercial gas-centrifuge uranium enrichment plant at Rokkasho.

The LFUA model has been implemented equally in URENCO facilities, i.e. in NWS (Capenhurst, UK) and in NNWSs (Gronau, Germany and Almelo, the Netherlands) as agreed in the HSP parties.

Since then, some adjustments have been made as required, e.g. the inclusion of environmental samples taking during LFUAs, DIVs and routine inspections outside cascade halls, and the application of C/S measures as necessary, reflecting technological improvements.

Plans for constructing commercial gas-centrifuge plants in Australia and the USA hit a setback soon after the conclusion of the HSP due to reduced market demands. In recent years, however, the construction plans of two large commercial gas-centrifuge plants with the capacity of 3,000 and 3,500t SWU/y respectively are under way in the USA. At present, no decision to apply the LFUA model to these plants has been announced.

6. Significance of HSP in Historical Perspectives

More than two decades have passed since the conclusion of the HSP. There have been numerous developments and improvements in enrichment and safeguards technology and in the legal framework for the implementation of safeguards. New safeguards methods and technologies have become available for routine use, including environmental sampling, remote monitoring, a continuous enrichment monitor (CEMO) and SNRIs (short notice random inspections). A few countries beyond the HSP parties have become technology holders of gas-centrifuge enrichment. Further, many countries are implementing additional protocols to their comprehensive safeguards agreements with the Agency. It has become necessary to develop new safeguards approaches to reflect these developments. In December 2005 SAGSI (Standing Advisory Group on Safeguards Implementation) reviewed and approved the Agency's proposed approach for gas-centrifuge enrichment facilities. Details regarding this proposal are the subject of another Symposium paper.

In light of these developments, what is the significance of the HSP in a historical perspective? The following points can be made:

- The HSP was a precedent for collaboration between the technology holders and the Inspectorates with the goal of establishing an effective and efficient safeguards strategy for a sensitive facility type. It led to the formulation of another important safeguards project called

LASCAR (Large Scale Reprocessing) with regard to safeguarding large-scale commercial reprocessing plants. The safeguards approach for Rokkasho Reprocessing Plant is based on the findings of LASCAR.

- In conjunction with the HSP, each technology holder prepared and conducted programmes to demonstrate to the Inspectorates various techniques in practical situations at their enrichment plants. These demonstrations were considered as useful, informative and well organized. This encouraged facility operators to offer their facilities as test beds for demonstrating new safeguards methods and technologies through respective Member States' Support Programmes.

- The IAEA and EURATOM also worked together on the common ground as the Inspectorates, which nurtured the relationship of mutual cooperation and trust between the two organizations, leading to the conclusion of the "New Partnership Agreement" (NPA) later.

- Though INFCIRC/153 provides for unannounced inspections (UIs), they had not been implemented routinely. The HSP paved the way for routine implementation of UIs through universal application of the LFUA model to relevant gas-centrifuge facilities.

7. Concluding Remarks

The HSP proved to be a highly useful exercise for the safeguards community, i.e. to the Inspectorates and the technology holders including both operators and State authorities. In view of the increased cooperation with SSACs that is foreseen and indispensable for the implementation of integrated safeguards, the HSP experience of good cooperation between the Inspectorates and the technology holders provides a good precedent.

Personally speaking, the HSP proved highly beneficial to the author who served as the Secretariat of the HSP in charge of drafting its final report. Attending the Plenary and the Team meetings, the author could visit various cities and facilities in the HSP parties and establish a life-long friendship with various participants through sometime very heated discussion in these meetings. This has been a real asset.

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Model safeguards approach for gas centrifuge enrichment plants

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Abstract. An improved model safeguards approach for gas centrifuge enrichment plants (GCEPs) has been developed by the IAEA. This model builds on the basic safeguards approach for commercial gas centrifuge enrichment plants that was established by the Hexapartite Safeguards Project (HSP) in the early 1980s. There were a number of reasons to update the HSP approach: since the HSP concluded its work. the plants covered by it have advanced in technology and grown in separative work capacity; operational procedures have changed and more plants are scheduled to be introduced; a number of States that did not participate in the HSP have acquired the technology and built GCEPs now subject to safeguards, or are planning to build GCEPs; and safeguards technologies have evolved during this period. This paper describes the revised model approach under traditional safeguards.

1. Background and scope

The IAEA has been safeguarding commercial gas centrifuge enrichment plants (GCEPs) for over 25 years, based on the approach established by the Hexapartite Safeguards Project (HSP) in the early 1980s. Over this period, enrichment plants have advanced in technology and grown in size, more States have acquired enrichment technology and have built plants subject to safeguards, and safeguards technologies have evolved. Plant-specific safeguards approaches have been developed and implemented by the IAEA taking into account these factors.

In 2004, based on an internal review, the Department of Safeguards decided there was a need to update the model safeguards approach for GCEPs to take into account advances in enrichment technology and safeguards techniques and the changed circumstances. An updated model approach was drafted and reviewed internally. To validate and possibly add to the safeguards verification techniques included in the model approach, the IAEA convened, in April 2005, a five-day Technical Meeting on Techniques for IAEA Verification of Enrichment Activities. The main focus of the technical meeting was the application of verification techniques to the safeguarding of GCEPs. The IAEA brought together uranium enrichment technologists, safeguards specialists, experts in enrichment activity verification and detection techniques, GCEP operators, and safeguards Departmental staff. Conclusions and recommendations from the meeting included:

(a) Use of installed instrumentation (including the use of authenticated operators' systems) to quantify material in vessels connected to the process.

(b) Performing unannounced inspections or short notice random inspections (SNRIs) in conjunction with a mailbox-type system for improving efffectiveness and efficiency.

(c) Employment of a defence-in-depth approach for the application of safeguards in GCEPs.

(d) Use of environmental sampling as a powerful safeguards tool and a strong deterrent inspite of results that are not yet timely.

The Secretariat presented its draft model safeguards approach for GCEPs to the Standing Advisory Group on Safeguards Implementation (SAGSI) in December 2005. The Secretariat sought SAGSI's advice with respect to the correctness of the defined safeguards objectives, the continuing role of limited frequency unannounced access (LFUA) inspections in the safeguards approach, and the importance of ensuring defence-in-depth. SAGSI endorsed the Secretariat's model safeguards approach. SAGSI noted that there is no suitable single generic approach, given the widely differing facility characteristics. The Secretariat needs to select the optimal combination of measures to suit each facility and to provide defence-in-depth. In SAGSI's view, LFUAs to the cascade halls should continue to be a key measure. Other important measures include unannounced inspections or SNRIs combined with mailbox declarations, and environmental sampling.

Based upon the recommendations from the technical meeting, the advice of SAGSI, and a thorough internal review of the revised model safeguards approach, in June 2006 the Department of Safeguards approved the improved Model Safeguards Approach for Gas Centrifuge Enrichment Plants for implementation. The objective of safeguards at a GCEP and the key elements of the revised model safeguards approach are presented below.

2. Characteristics of a commercial gas centrifuge enrichment plant

A typical large commercial GCEP is characterized by:

(a) Separative work unit (SWU) capacity: 2900 t SWU/y.

(b) Feed: 5000 t U in 590 48 inch cylinders.

(c) Tails: 4500 t U in 540 48 inch cylinders.

(d) Product: 500 t U in 290 30 inch cylinders.

The plant consists of feed and take-off areas, blending stations and several cascade halls containing multiple cascades connected in parallel with a large number of centrifuges per cascade. Depending on the design of the cascades, there may be valves inside the cascade halls for (or which would allow for): take-off points for UF_6; isolation of individual or sets of centrifuges; and reconfiguration of the cascade to produce higher enrichments.

In the HSP approach, measures to detect and to deter undeclared high enriched uranium (HEU) production include access to cascade halls; for plants without valves or take-off points, access to the cascade halls focuses mainly on visual observation; for plants with such valves or take-off points, emphasis is placed on non-destructive assay (NDA) measurements. In 1996, the collection of environmental swipe samples inside cascade halls was added as a routine safeguards measure.

3. Assumptions about undeclared enrichment plants

Under traditional safeguards it is assumed that a proliferator may have an undeclared enrichment plant. However, much would depend on the capacity of such a plant. Although a R&D or pilot-scale plant with rather small capacity could be used (say 500 kg SWU/y), it is also possible that a proliferator, having mastered the technology in a small plant, would seek to replicate it in a much larger plant. However, the proliferator might want to limit the overall size of such a plant to reduce its detectability. Take for example a 50 t SWU/y plant. Typical, early commercial cascade halls produce about 20 kg SWU/y/m^2. Thus, a 50 000 kg SWU/y plant would require a cascade hall of about 2500 m^2 (50 m x 50 m). Since the cascade hall takes up about 60% of the total building floor area, the building would be about 4000 m^2 (80 m x 50 m). Whilst quite large, such a building would be comparable with other large industrial buildings. More modern centrifuges having a higher individual SWU capacity would take up less space for an equivalent total capacity. Additionally, several floors of cascades could be built, further diminishing the size of the plant. The large energy consumption of these plants means that electrical distribution networks would be visible, especially for a clandestine plant in remote areas.

4. Acquisition paths

There are three main paths to be considered for the acquisition of HEU by a State having a GCEP with a declared maximum enrichment level of ca. 5% ^{235}U:

(a) Diversion of nuclear material from the declared feed, product or tails by the false reporting of material unaccounted for (MUF), shipper-receiver difference (SRD), or operator declarations reflected in operator-inspector difference statistics (D). The diverted material is shipped to an undeclared enrichment plant for further enrichment to HEU.

(b) The introduction of undeclared feed into the plant for enrichment to a level less than or equal to the declared maximum. The product from this activity is not declared and is shipped to an undeclared enrichment plant for further enrichment to HEU. This path is known as ‘excess production’.

(c) Undeclared production of HEU at the declared GCEP by reconfiguration or operation of the cascades in recycle mode. The HEU product is not declared and is shipped to an undeclared location. If declared feed is used, the missing material is ‘diverted’ into MUF, SRD or D as in (a). Alternatively, undeclared feed could be used as in (b).

A fourth acquisition path, misuse of the facility to produce enrichments higher than the declared maximum but less than 20%, with enrichment to HEU at an undeclared enrichment plant, is possible but not considered likely. It would have the disadvantages of both being open to detection by environmental sampling and requiring further enrichment at an undeclared plant to produce HEU. In any case, this acquisition path would be covered by measures available under the three main paths.

5. Safeguards objectives

The technical objective of safeguards (paragraph 28 of INFCIRC/153 (Corrected)) is “the timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or of other nuclear explosive devices or for purposes unknown, and deterrence of such diversion by the risk of early detection. From this overall objective, three specific safeguards objectives can be derived to cover the acquisition paths referred to above. These objectives and the possible safeguards measures available to meet them are discussed briefly below.

5.1. Objective 1:

The timely detection of the diversion of significant quantities of natural, depleted or low enriched UF_6 from the declared flow through the plant, and the deterrence of such diversion by the risk of early detection.

In the context of this objective, the meanings of ‘timely detection’, ‘significant quantities’ and ‘early detection’ need further clarification.

Timely detection is usually equated with the formal timeliness goal, one year in the case of depleted, natural or low enriched uranium (DNLEU). However, given that the acquisition path involves enrichment at an undeclared plant, it must be recognized that the time taken to produce one significant quantity (SQ) of HEU depends on the capacity of that plant and the enrichment of the feed. For example:

(a) Assuming a 50 t SWU/y plant with natural feed and 0.3% tails, one SQ of HEU at 90% ^{235}U would be produced in about six weeks.

(b) Assuming 5% LEU feed with 0.3% tails, one SQ of HEU at 90% ^{235}U would be produced in about one week.

The time taken to produce one SQ of HEU of a particular enrichment from feed of a particular enrichment will vary inversely with the separative work capacity of the plant. For effective safeguards, some measures should be included in the approach that will give a capability for detection of diversion in periods much shorter than the DNLEU timeliness goal of one year. Such measures would also contribute to meeting the objective of deterrence of diversion by the risk of 'early detection'. A reasonable goal would be one month, which is the timeliness goal for detecting the diversion of HEU.

With regard to 'significant quantities', these are normally equated with the formal SQ amounts defined by the current IAEA safeguards criteria: 25 kg ^{235}U for HEU and 75 kg ^{235}U for DNLEU. However, since most of the ^{235}U can be separated from DNLEU in a GCEP if the tails value is set low, 75 kg of ^{235}U in DNLEU could produce slightly more than two SQs of HEU. According to the formal SQ figures, one SQ of 5% LEU is 1500 kg U total, whereas 25 kg of ^{235}U is contained in only 500 kg of 5% LEU. Therefore, for effective safeguards, accountancy measurements ideally should be able to detect the diversion of quantities smaller than the formal SQ values for DNLEU.

The HSP approach for commercial GCEPs uses annual physical inventory verification (PIV) and routine inspections outside the cascade areas to verify the operator's declared flows and inventories of nuclear material. The approach aims to achieve 100% coverage of the flow, i.e. that all receipts and planned shipments should be available for verification. The amount of material actually verifie**d** is defined by the current safeguards criteria (e.g. medium detection probability for gross, partial and bias defects for all material connected to or removed from the process). The routine inspection frequency and duration needed to achieve these requirements depend on the throughput of the plant and the length of time that the material can be kept available for verification. Typically, monthly inspections by two inspectors of up to five days duration are carried out at large plants. An alternative means of achieving 100% coverage would be to use unannounced inspections or SNRIs with a mailbox for operator declarations of feed, product and tails cylinders. A similar concept is already used in LEU fuel fabrication plants. The use of the SNRI approach is advantageous in that it would also meet Objective 2 (detection of excess production).

Under the HSP approach at GCEPs, routine inspection results including NDA and destructive analysis (DA) are used in evaluating the U and ^{235}U material balances (MUF and D). For large plants the uncertainty of D for ^{235}U is large, leading to low diversion detection probabilities due to the limitations of available NDA techniques and the fact that depleted uranium cylinders are not routinely homogenized. In contrast, the uncertainty of D for uranium is low due to the accuracy of weighing systems, leading to high diversion detection probabilities for uranium. Both U and ^{235}U material balances continue to be evaluated. However, an optimising technique is used for calculating DA sample sizes rather than the traditional formulae. This optimal sample size maximizes the detection probability for ^{235}U, although it does not reach the medium detection probability goal. The optimal sample size attains the point of diminishing return where taking additional DA samples does not substantially increase the detection probability.

Material balance evaluation as an element of the HSP approach is aimed not only at detecting diversion of declared flows, but also at detecting indications of misuse of the facility to produce HEU or to process undeclared feed material to produce undeclared LEU. If these objectives relating to misuse can be met by other means, then the sample size requirements for material balance evaluation could be significantly reduced.

The use of authenticated operators' accountability precision balances could significantly improve the accuracy achievable in the material balance evaluation. The approach recommended here is therefore to maximize the accuracy by use of the operator's accountancy measurement system, with sufficient authentication and independent verification to retain credibility. This credibility is achieved by ensuring that 100% of the flow is made available by the operator for verification by the IAEA and that at least 20% of the flow is actually verified on a random sampling basis. Flow, in the case of centrifuge enrichment plants, is the uranium feed into the cascades and the withdrawal from them of enriched and depleted uranium. This is closely correlated to receipts and shipments, but may not necessarily be the same.

5.2. Objective 2:

The timely detection of the misuse of the facility in order to produce undeclared product (at the normal product enrichment levels) from undeclared feed and the deterrence of such misuse by the risk of early detection.

The ‘excess production’ scenario is not explicitly covered by the HSP approach. At the time of discussion, the HSP considered that flow verification was adequately covered by conventional material accountancy for plants with separative capacities up to about 2000 t SWU/y (a figure in any case that is now surpassed in some of the largest plants); it did not explicitly address the ‘excess production’ scenario. To address this scenario, flow verification requires verification not only of the content of the declared cylinders received or filled since the previous inspection but also that the only cylinders attached to or detached from the cascade were those so declared. Possible safeguards measures include the following:

(a) A mailbox approach with unannounced inspections or SNRIs where the operators would need to frequently declare all planned movements and locations of cylinders into an IAEA approved mailbox and hold those cylinders for an agreed time for verification, including DA sampling. The mailbox/SNRI approach could, depending on verification requirements, allow a reduction in the total number of routine inspections, and the LFUAs to the cascade halls could be carried out at the same time.

(b) Authentication and remote monitoring of operator load cells where an authenticated signal from the feed and withdrawal stations’ load cells would provide a constant record of the feeding or withdrawal of UF_6. Load cells used for process control are less accurate than precision balances, which are used for nuclear material accountancy. The use of both systems by the IAEA would increase the material balance verification capabilities, both in terms of detection probability and verification coverage. Authentication and calibration of the load cell reading is necessary. Joint-use equipment is subject to Departmental approval on a case-by-case basis.

(c) Surveillance of the feed and withdrawal stations and sealing of the full and empty feed, product and tails cylinders; this may be onerous for the operator and the IAEA in large plants.

(d) Measures to guard against the introduction of undeclared feed and withdrawal stations (e.g. unannounced inspections or SNRIs, and containment and surveillance (C/S)).

(e) Flow measurements from installed in-line flow meters. Combined with enrichment measurements of feed, product and tails (from either installed enrichment monitors, NDA measurements on pipes or cylinders, or DA measurements), flow meters would provide a direct measure of the material fed to and withdrawn from the plant. The flow-meter method, however, requires access to process gas pressure information which, depending on where the meter is installed, may require access to sensitive information which some operators are not prepared to provide, and may not have the necessary precision.

(f) A SWU balance approach where the objective is to calculate (e.g. on a monthly basis) the SWUs needed to produce the declared product and tails from the declared feed and compare this figure with the declared SWU capacity of the plant. Hence any apparently unused SWUs that could have been used to produce undeclared material could be detected. However, it should be noted that the operator is not required to declare the actual SWUs used, only the total design capacity of the plant, and that this figure is not accurately verifiable. It may be that the SWU balance approach is suitable only for small facilities.

(g) Continuous inspector presence.

5.3. Objective 3:

The timely detection of the misuse of the facility to produce UF_6 at enrichments higher than the declared maximum, in particular HEU, and the deterrence of such misuse by the risk of early detection.

To detect enrichment above the declared range, and in particular any undeclared production of HEU, the HSP approach uses LFUAs to the cascade halls for visual observation (to detect any changes to the cascade configuration or the introduction of valves or take-off points) and NDA measurements. Environmental sampling inside the cascade halls is now also carried out during LFUAs.

For plants conforming to the classical HSP model (i.e. those that cannot produce HEU without physical alteration of the cascade pipe configurations), the LFUA approach for detecting undeclared HEU production remains adequate when the LFUA inspections are truly unannounced. Improvements in centrifuge and cascade design post-HSP led to the IAEA being notified by the relevant technology holders that it was no longer possible to rely on visual observation of the cascade to rule out the possibility of HEU production. For these newer plants, and for other plants where (i) there is a lack of transparency, (ii) there are valves that would allow rapid reconfiguration of the cascade, or (iii) unannounced inspections are not possible, the LFUA approach relying mainly on visual observation is not sufficient.

Where allowed, environmental sampling (with particle analysis) during LFUAs gives a reasonably effective detection capability for current as well as past production of HEU, and also for enrichments between 5% and 20%; however, it does not meet the one-month timeliness goal under the current sampling frequencies and analysis times. A period of three to four months from sampling to evaluation of results is typical for a routine sample. Also, cross-contamination from other sites or from earlier activities on the same site can produce misleading results. The continuous enrichment monitor (CEMO) applied to product header pipes provides a timely detection capability. The portable cascade header enrichment monitor (CHEM) provides a possibility for NDA measurements in the cascade halls during LFUAs. The use of CEMO or CHEM is plant specific, depending on the diameter, thickness and material of the piping. As no one method provides the perfect solution for all cases, the capability to detect undeclared HEU production should be based on a combination of measures — ‘strength in depth’ — depending on facility specific considerations. These measures should include:

(a) Design information verification (DIV) in order to establish whether there are valves and take-off points in the cascade hall. Photographs and design information may be kept on site under IAEA seal.

(b) LFUAs to the cascade halls to carry out:

- Visual observation (comparison against photographs and design information).
- Environmental sampling.
- Portable NDA measurements.
- DA sampling from cascades, where possible.
- Installation of NDA equipment on product headers (e.g. CEMO) with remote monitoring to provide an immediate indication of any HEU production.
- Application of C/S inside cascade halls (particularly where transparency is limited and the operator does not need frequent access to the cascades).

For flexible plants where CEMO is not installed, the frequency of LFUAs would be significantly increased, with the emphasis on portable NDA measurements (e.g. CHEM) and the taking of environmental samples. Current applications of CHEM have a rather low detection probability due to the small number of measurements that can be made in the time available. CEMO and CHEM are IAEA-approved instruments; in the future other instruments are expected to become available.

6. Conclusions

The safeguards approach for a GCEP should be designed to meet all three of the objectives listed above. The main features of the approach should be:

(a) Annual PIV.

(b) DIV.

(c) Nuclear material accountancy verification of declared flows. The operator makes available 100% of the flow for verification by the IAEA with at least 20% of this flow being actually verified on a random basis according to the categories, detection probabilities and defect levels below:

- Low enriched UF_6 cylinders are item counted and verified with medium detection probability for gross defects by NDA, for partial defects by load cell weighing and NDA, and for bias defects by precision load cell weighing and DA sampling.

- Natural UF_6 cylinders are item counted and verified with medium detection probability for gross defects by NDA, for partial defects by load cell weighing and NDA, and for bias defects by precision load cell weighing and DA sampling.

- Depleted UF_6 cylinders are item counted and verified with medium detection probability for gross defects by NDA, for partial defects by load cell weighing and NDA, and for bias defects by precision load cell weighing and DA sampling.

This may be achieved by announced or random inspections. Verification of declared UF_6 weights by authentication and use of operator's precision balances should be used where possible to achieve greater accuracy in the material balance evaluation.

(d) Measures to confirm that there is no undeclared production of LEU enriched to levels not higher than the declared maximum level. These measures should be able to detect the introduction of both undeclared cylinders into declared feed and withdrawal stations and undeclared feed and withdrawal stations. Random inspections should provide the main element of these measures, complemented by C/S, monitoring of authenticated load cells or other measures as appropriate.

(e) Measures to detect the production of uranium at enrichments levels higher than the declared maximum level, in particular HEU. These measures should include LFUAs inside the cascade areas to carry out visual observation, environmental sampling, DA sampling (where possible), and NDA measurements. For flexible cascades, continuous monitoring to assure the absence of HEU production should be performed where practicable. Where continuous monitoring of such cascades is not practicable, the frequency of LFUAs should be increased significantly.

7. Current status of implementation

Early in the evolution of the improved safeguards approach for GCEPs, the IAEA recognized the need for a field trial to test the main elements of the improvements. The IAEA discussed the need to conduct a field trial with relevant State authorities and it was agreed to carry out such trials. Planning of the field trial is thus underway, including initial discussions with the relevant State authorities and GCEP operators.

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Evaluating new MC&A concepts for gas centrifuge enrichment plants

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Abstract. This paper describes two possible advanced safeguards techniques for material control and accounting (MC&A) to augment the IAEA's model safeguards approach for gas centrifuge enrichment plants. The first is designed to address the problem of how to detect clandestine production of significant quantities of HEU. Neutron detectors are proposed for monitoring the composition of material as it accumulates in cold traps outside of the cascade hall. These cold traps accumulate kilogram quantities of UF_6 from feed, product and tails portions of the system. Changes in relative enrichment levels of UF_6 can be potentially be obtained by monitoring the total neutron counts over time, whereas absolute enrichment values can be obtained via coincidence counting. The second technique is designed to detect diversion of LEU. The proposed approach is to apply increased surveillance with randomized inspections, such as "Mailbox"/Short Notice Random Inspection (SNRI), combined with real time mass balancing and a Data Consistency Monitoring System (DCMS). These two techniques would expand the capabilities of the IAEA to detect undeclared HEU production and undeclared LEU production from undeclared feed. The first approach can be applied to current enrichment facilities under existing access agreements. The latter is envisioned under future enhanced access provisions.

1. Introduction

Present IAEA safeguards have three principal concerns at gas centrifuge uranium enrichment facilities declared for the production of low-enriched uranium (LEU) with <20% ^{235}U [1]. These are:

1. production and diversion of a significant quantity of uranium with enrichment greater than declared (in particular, highly enriched uranium (HEU) with ≥20% ^{235}U)
2. diversion of a significant quantity of declared uranium (particularly in the form of LEU product).
3. production of LEU in excess of declared amounts (e.g., using undeclared feed).

The goal of IAEA safeguards is the ***timely*** detection of diversion of a ***significant quantity*** (SQ) of nuclear material. The IAEA, also referred to in this paper as the Agency, defines an SQ of ^{235}U contained in LEU and in HEU as 75 kg ^{235}U and 25 kg ^{235}U, respectively. The IAEA defines the timeliness goal for detecting a diversion of ^{235}U contained in LEU as one year and in HEU as one month. For LEU, the IAEA goal is to detect, within one year and with a 50% detection probability, the diversion of 75 kg of ^{235}U contained in LEU. For HEU, the goal is to detect, within one month and with "high confidence", the undeclared production of 25 kg of ^{235}U contained in HEU.

The current international safeguards approach for gas centrifuge enrichment plants (GCEPs) derives from the Hexapartite Safeguards Project (HSP) of the early 1980s. This approach by the IAEA includes inspection activities outside cascade halls (primarily to detect diversion by verification of declared nuclear-material flows and inventories) and inside cascade halls (primarily to detect production of HEU). By proposing a number of diverse monitoring techniques, this paper seeks to improve inspection activities outside the cascade halls in order to address the above three principal concerns. By its very nature, monitoring is somewhat plant specific, but the underlying principals should apply to any gas centrifuge uranium enrichment facility.

2. The Current Safeguards Approach

The activities inside cascade halls are based on a Limited-Frequency Unannounced Access (LFUA) approach. During a LFUA inspection activity, the operators agree to provide inspectorate access to the cascade halls within two hours, either during the course of an announced routine inspection or on a completely unannounced, random basis. The inspectors verify if the cascade hall centrifuge arrangements and piping are consistent with photographs that the IAEA and operator have taken and verified as part of Design Information Verification (DIV) activities. The LFUA, in essence, verifies that no changes have been made to the declared cascade hall. This provides some measure of verification that the cascades have not been altered to provide undeclared HEU enrichment capacity.

The IAEA also has in some facilities installed NDA equipment to detect undeclared HEU production. This Continuous Enrichment Monitor (CEMO) is installed on the product header pipes as a “yes/no” monitor to verify that the product enrichment is not above operator declared assays. The CEMO system will send a message to the IAEA every day stating that the product assay is within acceptable bounds, “yes”, or the product assay is above acceptable bounds, “no”. The system also sends a state-of-health signal to insure the IAEA that the CEMO is functioning normally. An inspector will immediately travel to the facility and investigate the situation if either a “no” or “poor” state of health message is received [2].

The Agency and EURATOM base activities to be performed outside cascade halls to verify the declared nuclear material balance on traditional techniques. These techniques include examination of records and reports; gross, partial, and bias-defect verification measurements of relevant nuclear materials; and the application of containment and surveillance techniques to maintain continuity of knowledge. The HSP also agreed that provisions should be made to give the Agency(s) the opportunity to verify the feed, product, and tails before they are fed to, or shipped from, the plant. The HSP declared that the mode of inspection would be intermittent. For facilities up to about 1000 metric tonnes of separative work units per year (MTSWU/yr), the HSP expected the average frequencies of routine inspection visits for activities outside and inside the cascade areas to be in the range of 12-15 times per year and 4-12 times per year, respectively.

To verify the uranium in the cylinders in which the UF_6 is stored during shipment, is fed into the enrichment process and is withdrawn from the enrichment process, the IAEA depends on the aforementioned traditional identification, non-destructive assay (NDA), and destructive assay (DA) techniques. Inspectors receive a declaration of materials in the plant, the static inventory and the flow inventory, e.g., shipments and receipts of UF_6. In accordance with Agency criteria for random sampling to enable a 50% probability of detection, they will create a sampling plan and verify the cylinders. The inspector will item-count all the static and flow cylinders and identify the cylinders by serial numbers that are marked or welded on the cylinders. According to the sample plan, the inspectors verify the product cylinders with a HPGe detector coupled to a multi-channel analyzer and an ultrasonic thickness gauge for gross and partial defects. The inspectors weigh the cylinders with a load cell or authenticated operator’s scale, depending on the facility and record the weight and enrichment of the product. The feed and tails cylinders are verified with a NaI detector coupled to a multi-channel analyzer for gross and partial defects. They, too, are weighed by the IAEA’s load cell or an authenticated operator’s scale, depending on the facility.

To detect a bias defect, the inspectors must draw DA samples from feed, product, and tails cylinders according to a sampling plan for that year, which specifies the number of random DA samples that should be taken from each stratum. The inspectors must perform DA sampling at virtually every interim inspection so as to have the proper number of DA samples, and to draw a valid DA sample representative of the population pool of the year's flow of material. The inspector will, in some cases, have to be well-informed and diligent about the sampling procedure.

The IAEA evaluates the nuclear-material balance during inspections. For a bulk-handling facility such as an enrichment plant, there will be always be some material unaccounted for (MUF). The MUF is evaluated, as well as D and MUF-D. D is the calculated operator/inspector difference statistic for strata with bias-defect and partial-defect verification. MUF-D, the inspector's estimate of MUF, is the difference between the estimated MUF and estimated D. The IAEA examines these values for both statistical and safeguards significance.

The conclusions of the HSP have been the basis for IAEA safeguards at gas centrifuge enrichment plants since 1983. However, the HSP did not address the question of undeclared feed. The current IAEA safeguards approach at gas centrifuge enrichment plants does not involve any specific measures for detection of undeclared feed or the undeclared product and tails that might be produced from it. However, as a result of the IAEA "Programme 93+2", environmental sampling inside and outside the cascade halls emerged as an additional tool for the detection of HEU production. Furthrmore, in the last year the IAEA investigated making changes to the HSP safeguards and is in the process of issuing a new model safeguards approach that will enable the IAEA to improve its ability to detect undeclared feed.

3. Advanced Safeguards Under Current Access Rights

To address some of the outstanding issues described above, we have devised a new approach that is promising for detection of HEU production at the proposed National Enrichment Facility (NEF) in the USA. Although developed on the basis of the NEF design, which is the latest URENCO gas centrifuge plant design, it is likely that this approach will be applicable to other gas centrifuge enrichment plants [3].

The cold and chemical traps in the UF_6 handling area of the NEF are designed to collect and purify material from each of the centrifuge cascades within a given cascade hall. Because all material being enriched is purified of light gases, some fraction of the total mass of cascade material will be represented in the chemical or cold traps. Separate chemical and/or cold traps are used to purify the feed, product, and tails material, making them ideal for detecting a variety of possible clandestine activities in different parts of the system. Furthermore, because the UF_6 handling areas are outside the cascade halls, the traps provide a unique opportunity to monitor changes in enrichment over long time periods.

In this paper, we will focus on the cold traps owing to the higher mass accumulation relative to the chemical traps. The purpose of the cold trap is to separate UF_6 from light gases (e.g., HF) that accumulate during the enrichment process so that the purified UF_6 can then be recycled back into the cascade. The design operation of the cold trap is very simple. The cold trap is a cylindrical tank surrounded by a layer of thermally moderating oil. When the oil is cooled, the UF_6 within the trap is desublimed onto the internal walls of the trap. The HF remains in the gaseous phase and is vented from the cold trap. However, some UF_6 also remains in the gaseous phase. This gaseous UF_6 vents from the cold trap along with the HF. Following venting, the thermally moderating oil in the cold trap is heated and the UF_6 sublimes. This UF_6 passes back into the cascade system for further processing. Meanwhile, the majority of the UF_6 and HF mixture vented from the cold trap must be prevented from reaching the vacuum pumps that lead to the Gaseous Effluent Vent System where final gas purification takes place. A chemical trap placed downstream of the cold trap in between the cold trap and the vacuum pump protects the vacuum pumps. These chemical traps contain both activated charcoal designed to trap UF_6 and Al_2O_3 designed to trap HF gas.

The accumulation of UF_6 in the cold trap is needed to estimate the amount of radiation that will be detectable by monitoring the trap while it is being filled. The accumulation rate should remain relatively constant over time because it is controlled largely by the carefully engineered design of the centrifuges within a cascade. The mass of material collected in the traps is continuously monitored in order to ensure that the traps are working effectively, and are emptied when full or saturated. Over time, the cold traps can accumulate large amounts of UF_6 (up to several kilograms). Available information suggests that the product cold traps fill to 20 kg of UF_6. Because the mass of UF_6 that accumulates over a given time is sufficiently large, there is a reasonable probability of devising a radiation detector that will be sensitive enough to measure the degree of enrichment of the accumulating material.

The possible range of uranium enrichments entering the cold traps under normal operating conditions can be generally understood by knowledge of the types of traps used to service each cascade hall. Note that each vacuum pump set contains an activated charcoal trap and an Al_2O_3 trap. Traps that service the cascade halls include:

1. feed purification cold trap
2. feed purification vacuum pump
3. product vent cold trap
4. product vacuum pump trap
5. tails vacuum pump

As a consequence of the traps servicing particular processes, the level of enrichment in a given type of trap should correspond to the expected range of enrichments in the feed, product, or tails.

3.1. Monitoring Process

3.1.1. Neutron Detection Theory

The proposed enrichment monitoring approach uses neutron detection, rather than gamma detection (as used in the CEMO), owing to known challenges with gamma measurements arising from uranium holdup and the large thicknesses of material deposits in the chemical and cold traps.

During the accumulation of UF_6 in the trap, two types of neutrons are emitted: spontaneous fission neutrons, and neutrons produced indirectly by uranium alpha decay, when alpha particles collide with adjacent fluorine atoms and release a neutron. For the three uranium isotopes known to be present when enriching natural uranium, ^{234}U, ^{235}U, and ^{238}U, most spontaneous fission neutrons are produced by ^{238}U, whereas both ^{234}U and ^{235}U have relatively low spontaneous fission yields. In comparison, ^{234}U has a much higher (α, n) neutron yield than both ^{235}U and ^{238}U. During the enrichment process in a cascade of centrifuges consisting of successive stages of separation, ^{238}U is separated via mass difference from ^{234}U and ^{235}U. The ^{234}U and ^{235}U are increasingly enriched, but the ^{234}U /^{235}U ratio is nearly constant over a wide range of enrichments. The result of the relative enrichment of ^{234}U during this process is that the number of (α, n) neutrons increases with increasing enrichment, providing theoretical grounds for further examination of neutron counting methods for monitoring relative and absolute enrichment levels.

Given data for the value of the ^{234}U /^{235}U ratio, it is possible to calculate the expected total neutron (spontaneous fission plus (α, n) neutrons) production from UF_6 containing a range of ^{235}U enrichments. We assume that thick target yields are appropriate owing to the large masses involved. To illustrate the gross changes in expected neutron count for different levels of enrichment, Fig. 1 plots the cumulative number of neutrons produced in a cold trap that accumulates a total uranium mass of 14 kg in about 80 minutes. Actual accumulation times may vary, and may be shorter for product than feed cold traps. In this example, the initial uranium enrichment level is equal to 5% in all cases, and is increased after seven days (corresponding to about 3 kg of uranium) to 20%, 50%, 70%, and 90% enrichment until an additional kilogram of uranium is added (over about 2 days). The enrichment level was then returned to 5% for the remaining nine days until the total accumulated mass in the trap was equal to 8 kg of uranium.

The results of these calculations (Fig. 1) show that for 5% constant enrichment, there is a linear increase in the cumulative neutron count with additional uranium. In comparison, if the relative degree of uranium enrichment increases, the slope also increases. The steeper slopes positively correlate with higher relative degrees of enrichment. Subsequently, if the enrichment returns to 5%, the slope is again consistent with the 5% slope. However, the cumulative neutron count is, of course, higher than at the 5% constant enrichment level because of the added mass of enriched material.

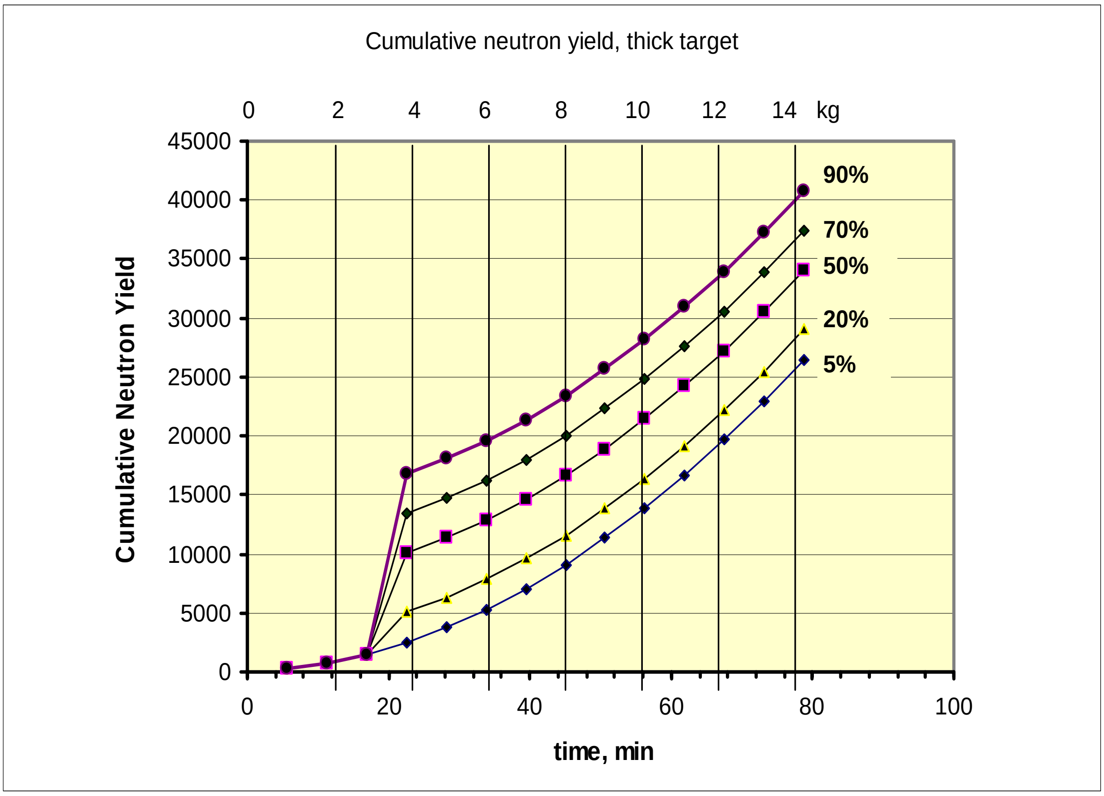

FIG. 1. Cumulative neutrons produced over time for uranium enrichments corresponding to 5%, 20%, 50%, 70%, and 95% enrichments. The initial enrichment is equal to 5% in all cases, but is increased to varying enrichments at a mass of ~3 kg, and then returned to 5% at ~4 kg. The slope of the trend increases with increasing enrichment, indicating that continuous neutron measurements should enable detection of increases in relative enrichment levels over time.

The relative increases in slope, owing to greater enrichment levels with respect to the 5% enrichment trend, suggest that long-term monitoring of chemical or cold traps may be a viable means by which to detect significant changes in enrichment. Measured increases in neutron counts/time over long time periods could be compared with pre-determined calibration slopes to help indicate whether the Agency should investigate the situation to determine if any undeclared activities are occurring. It is also possible that, in additional to the relative enrichment information that would be provided by this approach, coincidence counting, as described below, could also be used to determine specific enrichment values. However, a "yes-no" threshold enrichment value approach, as used presently with the CEMO, could prevent the specific enrichment values, which the operator may want to protect as proprietary data from being released to the IAEA.

3.1.2. Coincidence counting

Neutrons from spontaneous fission and induced fission are emitted essentially simultaneously. Spontaneous fission is usually accompanied by the simultaneous emission of more than one neutron. Thus, an instrument that is sensitive only to coincident neutrons will be sensitive to the presence of only particular uranium isotopes. These coincident fission neutrons are not coincident with background neutrons, or with those from (α, n) reactions because the latter have random arrival times. Thus, fissions induced by an (α, n) neutron source can be distinguished from other sources by using a coincidence circuit.

When ^{238}U is present in kilogram quantities, it will have a measurable spontaneous fission yield. Fission can also be induced in fissile isotopes such as ^{235}U, such that a sample containing large quantities of ^{235}U can be assayed by coincidence counting of induced fissions. The induced coincidence response is a measure of the quantity of fissile isotopes present. The parameter "α" is the ratio of the (α, n) neutrons to the fission neutrons and is therefore a measure of the ratio of ^{234}U to ^{238}U. Since the ^{234}U concentration increase follows the enrichment of ^{235}U, the parameter α is also a measure of the enrichment. This dependency is nearly exponential, so that the measurement of the α is a sensitive diagnostic of enrichment.

3.1.3. Theoretical neutron detector performance

The theoretical efficiency and sensitivity of neutron detectors for total neutrons and coincident neutrons can be calculated using the Ensslin Figure of Merit Code (EFOM) [5]. First, the total number of neutrons produced by enrichment of natural uranium as UF_6, including contributions from ^{234}U, ^{235}U, and ^{238}U, are calculated. The EFOM can then calculate the expected neutron count rate based on input information pertaining to the predicted (α, n) neutrons, spontaneous fission neutrons, and detector parameters (Table 1).

TABLE 1. Input values used to calculate detected neutron count rate and uncertainties

Variable	Value	Unit
mass value	1000	grams
efficiency	30	percent
"α" = (α,n)/Spontaneous fission neutrons	24.8	
count time	1000	seconds
multiplication	1	
die away time	50	microseconds
interrogating neutron rate	0	per second
gate width	64	microseconds
predelay	3	microseconds
background trigger rate	0	
counting deadtime	0	nanoseconds
active neutron calibration	0	doubles/gram
additional doubles gate loss fraction	1	
additional triples gate loss fraction	1	
active neutron coupling coefficient	0.5	

Calculations based on Ensslin Figure of Merit Code, version 1.2, 2004, Copyright M. Pickrell LANL

3.1.4. Calculated statistical results for detection of total neutrons and coincident neutrons

The neutron counts and uncertainties were calculated for a theoretical neutron detector having the characteristics provided in Table 1. For the singles neutrons (spontaneous fission + (α, n) reaction), based on an accumulated mass of 2000 grams of 16% enriched material added, and a total count time of 1000k seconds, the singles count rate is about 300 counts per second with an error on the order of 0.1%. The coincident (or doubles) neutron uncertainties will be larger, and will increase at higher values of enrichment. However, by the time the cold trap is approaching capacity, the mass will be much greater than 2kg, such that it should be possible to obtain reasonable uncertainties by coincidence counting.

3.2. Future work

The proposed neutron total- plus coincidence counting approach is promising for application within an centrifuge enrichment facility to monitor the composition of feed, product, and tails in cold and chemical traps. The next step is to consider how the mass and composition of material in the traps corresponds to the changes in mass and composition of material in the cascade. The sensitivity of the detection method to changes in the level of enrichment is partly dependent on the mass proportion of enriched material entering the trap, and the possible range in $^{234}U/^{235}U$ of the material. Additional calculations and Monte Carlo modeling with the MCNP code are in progress to validate the proposed measurement technique.

4. Advanced Safeguards with Increased Access Rights to Data

4.1. Increased Surveillance with Randomized Inspections

"Mailbox" declarations have been used in the last two decades to verify receipts, production, and shipments at some bulk-handling facilities (e.g., fuel-fabrication plants) [6]. The operator declares the status of his plant to the IAEA on a daily basis using a secure "Mailbox" system such as a secure tamper-resistant computer. The operator agrees to hold receipts and shipments for a specified period of time, along with a specified number of annual inspections, to enable inspector access to a sufficiently large statistical population of UF_6 cylinders and fuel assemblies so as to achieve the desired detection probability. The inspectors can access the "Mailbox" during randomly timed inspections (unannounced or SNRI as agreed by the State, Agency, and operator) and then verify the operator's declarations for that day. Previously, this type of inspection regime was considered mainly for verifying the material balance at fuel-fabrication, enrichment, and conversion plants [7].

In order to detect activities associated with undeclared LEU production at GCEPs, Brookhaven National Laboratory has expanded the "Mailbox"/SNRI concept to include declaration and verification of UF_6 cylinder operational data coupled with enhanced video surveillance in the feed and withdrawal areas[1]. Since the "Mailbox" declarations would also include data relevant to material-balance verification, these randomized inspections would replace the scheduled monthly interim inspections for material-balance purposes; in addition, the inspectors could ***simultaneously*** perform the required number of LFUA inspections needed for HEU detection. The advantages of the proposed approach become clearer when one considers the effort involved in performing the present scheduled Physical Inventory Verification (PIV), 11 scheduled interim inventory verification (IIV) inspections, and the 6-14 random LFUA inspections. A proposed regime of a PIV and 13 SNRIs [1] that includes the activities in the PIV, 11 IIVs, and the 6-14 LFUAs would encompass less total travel, inspection activity, and about the same level of intrusiveness to the plant as would the PIV, 11 IIVs, and the 6-14 LFUAs of the present IAEA approach. Furthermore, the "Mailbox"/SNRI concept would provide improved detection capabilities for a wider range of diversion schemes than the present IAEA approach.

4.2. Real-time Mass Balancing

The IAEA has experience with the use of a load-cell-based weighing system for HEU feed cylinders obtained during the IAEA verification experiment on HEU downblending at the Portsmouth Gaseous Diffusion Plant [8]. The load cells provided data to IAEA computers, and the data could be used to trigger video-surveillance cameras under certain conditions. A new type of centrifuge safeguards approach is proposed that would consist of putting a load-cell-based weighing systems on all feed, product, and tails feed and/or withdrawal stations, along with cameras that could monitor the attachment and detachment of a cylinder from a station. By development of accurate load cell weighing stations, one could have a rather precise knowledge of the material balance of the uranium flows in the plant. The amount of uranium contained in feed, product, and tails cylinders at any time is on the order of 10^2 to 10^3 times the process uranium inventory. Hence, real time accountancy based summing the product and tails load cell weights, and subtracting the feed load cell weights, is practicable.

This value should be close to zero with a band of uncertainty to be determined from the random and bias errors of the load cells. If the summed value strays from zero, it would indicate an anomalous reading. This anomalous reading would indicate feed and withdrawal at unconventional ports to/from the process. Combining video surveillance of the feed and withdrawal stations and a thorough design verification of the plant with the data from the load cells gives increased confidence that one can detect undeclared LEU production from the undeclared feed scenario. A similar system should be installed at the product blending stations to insure that no LEU product is siphoned off from the product coming from the process cascades prior to being shipped off-site.

A load-cell-based weighing system with real-time accountancy (RTA) and video surveillance may eliminate the need for strict ^{235}U balance verification measures. If one has confidence that this system can detect any undeclared feed and withdrawal at centrifuge plant, then the need to check the ^{235}U assay becomes superfluous. Random inspections that would verify feed, product, and tails assay declarations and material balance evaluations would confirm that the assays declarations were in order. The advantage of this approach is that the inspector need not use NDA techniques to verify the ^{235}U assays, which have large uncertainties (e.g., International Target Values (ITV) 2000 for NDA for UF_6; natural uranium = 10% random, = 8% systemic (NaI detector),; LEU = 4% random, = 2% systemic (HPGe detector),; depleted uranium = 20% random, = 15% systemic(NaI detector),) compared with those involved in weighing the UF_6 (e.g., International Target Values 2000 for weighing with load-cell based weighing system (LCBS) or electronic balance (EBAL) = 0.05% random, = 0.05% systemic) [9]. Using the ITV 2000 values one can calculate the uncertainities in an operator-inspector difference with a 99.73% confidence interval (3-σ spread) of UF_6 feed, product, and tails with measurements of weight alone (i.e., uranium balance only), as well as with measurements of weight and isotopic assay (i.e., uranium and ^{235}U balance). Weights alone give, for all three strata of UF_6 , a 3-σ uncertainty of 0.43%. However, weight and isotopic assay give, for feed, product, and tails, a 3-σ uncertainty of 38%, 13%, and 75%, respectively. If higher confidence is needed than that provided by RTA and video surveillance methods, then the uranium and ^{235}U balance even with its high uncertainities must be verified because undetected feed and withdrawal would otherwise permit uranium diversion.

4.3. Data Consistency Monitoring

Data consistency monitoring is an extension of real-time mass balancing, and is based on the premise that a time-history collected from one measurement source should be consistent both with time histories collected at other sources and with the operator's declaration of plant operation. Akin to the term 'solution monitoring' that is used in reprocessing plants, we use here the term 'data consistency monitoring' to describe the process of collecting and correlating these histories. Although the role of such monitoring in GCEPs has been described previously [10], few details were given at that time, and there is little evidence to suggest that any of the ideas were ever developed further. The focus here is on the need to confirm that the materials handling, feed and receipts systems are operated as declared, so that additional systems would have to be installed, at least temporarily, to enable an undeclared activity. Such an activity should be detectable through surveillance.

The collected time histories can be evaluated in at least two different ways: reference-signature-based and model-based. In both approaches, key features in the time histories could be marked as either events or sub-events. In reference-signature-based approaches, event sequences are then correlated with one or more reference signatures to determine if a particular activity (identified *a priori*) has occurred. In model-based approaches [11], sub-event sequences are correlated with sets of possible physical activity hypotheses.

Identified hypotheses are then used as boundary conditions to a quantitative mathematical model, which is solved to predict measurement histories that can be compared with those collected.

4.3.1. An Example Based On A UF_6 Feed System

The detailed design of any Data Consistency Monitoring System (DCMS) would depend considerably on the plant layout and its control and instrumentation provision. Though operation of the cascades themselves is virtually totally opaque to outsiders, any facility must have provision to enable the operators to:

1. isolate individual cascades
2. meter the flow into an individual cascade at an appropriate pressure
3. connect individual feed cylinders
4. collect product into cylinders
5. collect tails into cylinders

It is important to appreciate that the feed system will be a carefully designed, commissioned, and operated component, because of the need to ensure that UF_6 enters a cascade at the correct flow rate, temperature, and pressure. Although its design will vary from facility to facility, the feed system instrumentation and valves ensure a controlled system is maintained. The feed system controls manage the emptying and switching of cylinders which store the solid feed. To facilitate feed cylinder management, weighing systems are often provided. In the design for the URENCO National Enrichment Facility in the USA, a separate UF_6 feed system is dedicated to each individual cascade hall [3]. Gaseous UF_6 feed flows from Solid Feed Stations that are connected in parallel to the main feed header.

We here evaluate the potential to process 5% LEU as feed instead of natural uranium in this type of facility. LEU is typically stored in 30B cylinders, and would have to be connected into the feed stations. However, these cylinders are significantly smaller than the 48Y feed cylinders that would normally be used. Consequently, if the operator declared normal operation with 48Y cylinders, the observed and expected operational patterns for various weight, pressure, and valve opening histories would differ. Attempts to 'mask' this activity could be detected through inconsistencies in the measured mass over time. The modeled differences in observed signatures can be seen in Fig. 2, which compares weight histories that might be observed during normal operation (on the left) with those that might be observed if the cascade hall was used to enrich the contents of three 30B cylinders. Here we have 6 feed stations operating, with 3 stations feeding the enriched material**.** Considerable activity would also be observed in one or more of the feed valves during each change-over (Fig. 3).

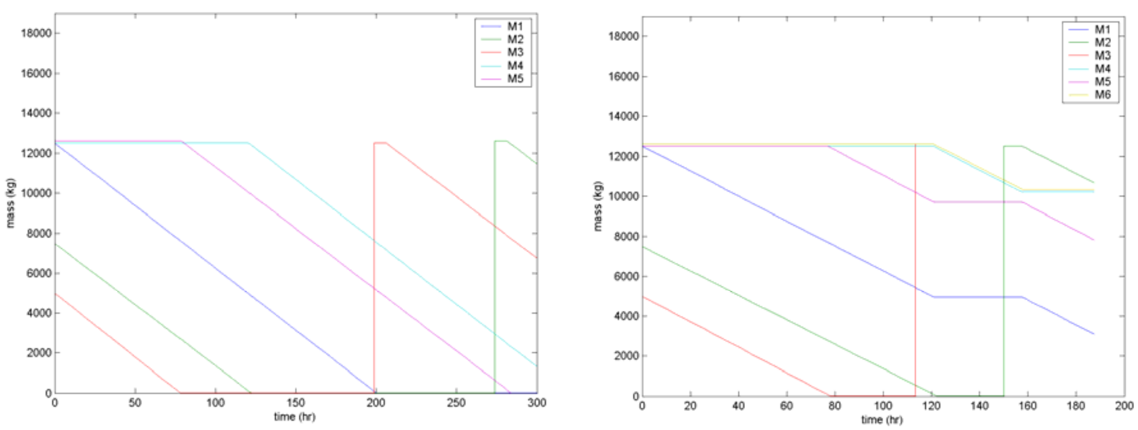

FIG. 2. Normal (left) and abnormal operations(right)

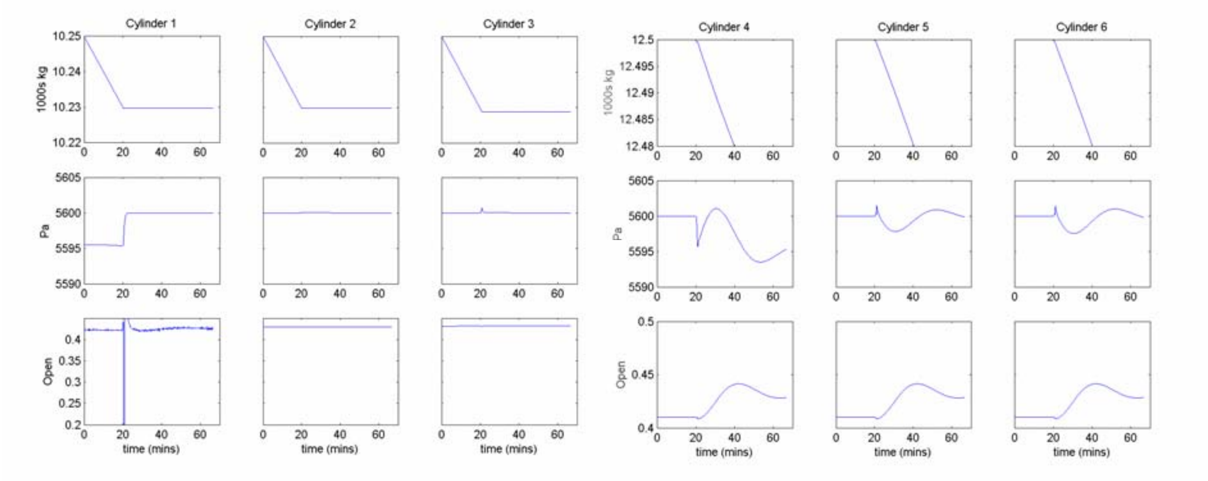

FIG. 3. Cylinder histories at station change-overs

The proposed evaluation system should ultimately be able to correlate activities both upstream and downstream of the cascade hall, and accommodate the cascade hall inventory, cascade equilibrium times, and abnormal operations like unplanned removals.

4.4. Conclusions

Current NDA measurements on the cascade header pipes, which reside outside the cascade hall, do not address the potential for diversion of material into vessels within the cascade hall itself. In addition, LFUA, which permits random inspections on short notice, does not permit continuous monitoring needed to eliminate potential diversion between inspections. By examination of the current safeguards approaches, we have identified new approaches to MC&A for gas centrifuge enrichment plants that represent enhancements over existing techniques. These can be divided into two types: the first, non-destructive assay of chemical and cold traps, and the second, verification using feed/product/tails weighing systems, normal header/station pressure measurement sensors, and detection of valve openings. We propose that these new MC&A techniques, that provide continuous monitoring of enrichment in all parts of the plant (feed, product, and tails), will allow the IAEA to better reach its goals under the new model safeguards approach.

The first technique applies neutron detectors to continuously monitor the contents of chemical and cold traps, thereby enabling inspectors to detect relative changes in enrichment over time as the traps are filled. This approach is complementary to current methods, and offers some new advantages: (1) unlike gamma-based systems, total neutron-based counting systems are not affected by hold-up within the header pipes, which for gamma measurements can give rise to errors in determining the level of enrichment; (2) the cold traps reside in a processing area located outside the cascade hall, and therefore can be readily accessed by the IAEA; (3) material being processed by the facility must be purified of light gases prior to further enrichment, meaning that some fraction of all material in the cascade should be represented in the cold traps; (4) separate traps are used for feed, product and tails, thereby allowing inspectors to monitor material compositions throughout the system, and (5) with the addition of coincidence counting, the total neutron-based monitoring approach can be further enhanced to provide specific information on the level of enrichment, but can be applied as a “yes-no” monitor to protect proprietary plant information. Preliminary calculations suggest that this approach will be fruitful to detect undeclared HEU production at the latest generation URENCO facilities, and probably at other types of gas centrifuge enrichment facilities.

The second technique combines increased surveillance using randomized inspections, such as SNRI, with real time mass balancing and a DCMS. The proposed approach would come with increased access to the operator’s data. An expanded "Mailbox"/SNRI concept, including declaration and verification of UF_6 cylinder operational data to detect activities associated with undeclared LEU production, can be coupled with enhanced video surveillance in the feed and withdrawal areas, at GCEPs. Authenticated load cells on the feed and/or withdrawal stations for feed, product, and tails would provide further verification of the operator's declaration in real time, because he operator's declarations in the Mailbox should match what is observed on the load cells. The DCMS will take the time-history collected from one measurement source and verify that it is consistent with the time histories collected at other sources, and with the operator's declaration of plant operation. Finally, the simplest form of the DCMS proposed here is mass balance. By gaining access to cylinder weighing system data, we believe that it should be possible to evaluate mass balances in real-time. Hence, the DCMS will provide a level of timely assurance that a facility is operating as declared.

These two techniques would expand the capabilities of the IAEA to detect undeclared HEU production, as well as undeclared LEU production from undeclared feed. Both of these techniques should assist the IAEA to reach its detection goals for enrichment facilities which are a special concern for the new model safeguards approach for enrichment facilities.

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Considerations on enhanced safeguards approach for centrifuge enrichment facilities

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Abstract. Safeguarding centrifuge enrichment facilities remains a key point of the safeguards system due to technical features of the process – large separation factor, low energy consumption, small equilibrium time and flexibility for configuration's change. The safeguards approach presently applied to commercial centrifuge enrichment facility is based on the Hexapartite Project that was developed in a totally different technical and political framework and seems to be not appropriate to cover some important misuse scenarios. The present non-proliferation environment, the existence of facilities with separative capacities varying from few hundreds to several millions of SWUs, the increased flexibility of cascade design and the new available methodologies and techniques for verification make necessary to review the safeguards approach for centrifuge enrichment facilities. This is a very relevant task, considering the improvement on the separative efficiency and the growing number of countries possessing centrifuge technology. Based on the last developments in this area, this paper analyses some alternatives for developing an enhanced safeguards approach for centrifuge enrichment facilities.

1. Introduction

The application of safeguards to enrichment facilities has been always a critical point of the international verification regime due to the capability of such facilities to produce unirradiated direct use material. The centrifuge enrichment technology has characteristics - large separation factor, low energy consumption, small equilibrium time and flexibility for configuration's change - that make it even more critical from safeguards standpoint. The current safeguards approach applied to commercial gas centrifuge enrichment facility is based on the Hexapartite Project [1] that was developed in the early 1980s in a totally different technical and political framework and seems to be not appropriate to cover some important misuse scenarios. In addition, the current safeguards criteria are not applicable to small centrifuge enrichment facilities. Over the last 25 years the situation has changed enormously. A growing number of countries have constructed centrifuge facilities; the most of them are under IAEA safeguards. Improvements of the centrifuge technology resulted in increased separative capacity and efficiency that allow the construction of more compact and flexible enrichment facilities. Besides that, concerns with the protection of technological, commercial and industrial confidential information, and, on the other side with proliferation of sensitive information difficult the preparation of a safeguards approach. The new safeguards environment under the Additional Protocol and the need to improve the capability to detect undeclared activities represent a challenge to be faced.

The present non-proliferation environment, the existence of facilities with separative capacities varying from few hundreds to several millions of SWUs and the new available methodologies and techniques for verification made necessary to review the safeguards approach for centrifuge enrichment facilities. Recognizing this fact the IAEA convened in April 2005 a Technical Meeting on Techniques for Verification of Enrichment Activities, where several techniques and methodologies were presented and discussed [2]. Several papers have deal with new safeguards methodologies and techniques. In this context, this paper reviews some alternatives for developing an enhanced safeguards approach for centrifuge enrichment facilities.

Some important boundary conditions have to be considered in implementing a safeguards approach for centrifuge enrichment facility. There are some commercial, industrial and technological information that the facility operator wants to protect. Technological information can also be very sensitive from the non-proliferation standpoint. The effective implementation of safeguards however requires information that nevertheless should be the minimum necessary to able the verification according to the safeguards agreements. All those point make clear that a compromise between effectiveness/efficiency and transparency is necessary to implement a sound safeguards approach.

The current trend is to implement a facility specific safeguards approach that would include some specific measures to a basic generic approach, using the available set of tools – techniques and methodologies. It should be emphasized that the specific approach has to satisfy all safeguards objectives. To meet all safeguards objectives, the measures have to provide continuity of knowledge in terms of operation and inventories, adequate deterrence and timely detection. Because that, an appropriate level of redundancy is necessary and required.

2. Safeguards Objectives for Centrifuge Enrichment Facilities

As provided for in the safeguards agreements, the safeguards objective is the *timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or of other nuclear explosive devices of for purpose unknown, and deterrence of such diversion by the risk of early detection*. For enrichment facilities this general objective is translated into three technical objectives:

a) The detection of the diversion of 75 kg of U-235 contained in declared natural, depleted and low enriched uranium (LEU);
b) The detection of facility misuse to produce undeclared uranium with enrichment level lower than or equal the declared maximum (excess production), which is informed by the operator in the DIQ;
c) The detection of facility misuse to produce uranium with enrichment higher than the declared maximum, in particular high-enriched uranium (HEU).

Regarding to those technical objectives, the following considerations have to be done:
(i) The diversion or misuse has to be detected on a timely basis what, under traditional safeguards, means one month for HEU and one year for LEU;
(ii) For facility misuse the concept of significant quantity cannot be applied strict sense, because the undeclared production means already an anomaly;
(iii) The risk of early detection, i.e. the deterrence should be always considered although it cannot be quantified. Safeguards measures that introduce deterrence are very relevant in centrifuge enrichment facility because the potential capability to produce unirradiated direct-use material; (iv) for enrichment facilities, even under traditional safeguards, the detection of undeclared nuclear material and activities (facility misuse) was always considered.

3. Basic Measures of a Safeguards Approach

Basically three kinds of measures shall be applied at centrifuge enrichment facilities:

(a) Measures to verify independently the inventories and flows of nuclear materials that are declared by the facility operator; these measures include the verification of nuclear material inventory and flows during physical inventory verification (PIV) and interim inspections;
(b) Measures to verify that all product has enrichment lower than the declared maximum; these measures include the verification of the uranium enrichment, the verification that all uranium routes are only that declared in the DIQ, and that the cascades are connected as declared in the DIQ;
(c) Measures to confirm that only verified feed have been processed; these measures include, in addition to that described in item (b), the verification that only declared UF6 cylinder are connected and disconnected at the feed and withdrawal station (F/W station).

4. The Current Safeguards Approach

The current safeguards criteria are based on the safeguards approach implemented under the Project Hexapartite, which was developed by the technology holders in the early 1980s. The Project is very well documented and the safeguards approach has reflected the political and technical framework at that time. The concept of limited frequency unannounced inspections (LFUA) has been adopted. Other concepts like the introduction of 100% perimeter control have been abandoned due to the high costs, lack of technology and intrusiveness.

The operator undertakes the feed cylinders will not be connected to the process and the product and tails cylinders will not be shipped out or blended before they are made available for verification by the inspectors. For this purpose, there is full access to the F/W station, including the use of NDA and sampling of cylinders, and NDA of cascade headers outside the cascade area. The actual inspection frequency depends on the facility throughput and facility practice with regard to the handling of UF6 cylinders. An annual physical inventory taking (PIT) is performed by the facility operator and verified by the inspectors (PIV). All these activities are performed outside the cascade hall and cover the scenario of diversion of declared uranium. The main problem regarding these measures is that the material balance uncertainties are relatively large for high throughput facilities. In addition to reach an adequate level of detection probability a large number of samples for destructive analysis (DA) has to be collected, what has an important impact in the safeguards efficiency and effectiveness.

Inside the cascade hall, measures are applied to detect HEU production based on LFUA concept. Unannounced access to the cascade hall (delay up to 2 hours) is provided and visual observation is performed to verify that the cascade piping and configuration are as declared in the DIQ and the absence of additional F/W points inside the area. Like valves or sampling points. In addition to the visual observation, NDA measurements to verify the enrichment of the product current are also foreseen. Monitors for continuous enrichment measurement were developed and are applied in some commercial facilities, although the low detection probability. Their application however is facility specific because the need of a minimum tube diameter and maximum thickness, due to the low gas pressure and possible material deposited inside the tube. The measures inside the cascade hall cover in some extent the scenario of HEU production. The introduction of environmental sampling in the middle of nineties has increased substantially the effectiveness to cover this scenario and this measure has been incorporated to the current approach, although this measure, in general, does not meet the timeliness goal.

The main problems of the current approach can be summarized, as follows

- (i) The main disadvantage is the lack of measures to cover the scenario of excess production because it does not consider the verification that only declared uranium is processed, that means only declared cylinders are feeding the process.
- (ii) Visual observation inside the cascade hall alone may not be enough to assure the absence of product with enrichment higher than declared or HEU production because the large flexibility of the modern facilities;
- (iii) It is difficult to assure the absence of HEU product during DIV of cascade hall in large facilities;
- (iv) The traditional material balance closing once a year is associated with large uncertainties that, for large throughput can go up to 5%. on SWU basis. In a typical 1,000,000 SWU facility this means that about 50,000 SWU could be diverted.
- (v) The current flow verification procedure, although the large inspections effort associated, does not cover randomly all the internal and external transfers;
- (vi) In general, the current approach cannot be directly applied to small facilities, like pilot plants and research and development facilities, which do not have an operational routine and where access to the cascade hall is very sensitive. For those plants that have very low inventories and throughput, the misuse scenario is the most dominant. The current safeguards criteria, however, foresees for such plants only a PIV and a DIV per year and should be updated.

In order to increase the effectiveness and efficiency of the current safeguards approach, several techniques and methodologies have been analyzed and developed over the last years; others are in study. Many of them have been presented in a technical meeting convened by the IAEA in April 2005. Based on these developments, a summary of some potential alternatives for implementing an enhanced safeguards approach is presented in the next section.

4. Methods to enhance the safeguards approach: A review

Several methods have been developed over the last years to improve the safeguards approach in order to meet more effectively all technical objectives. It seems appropriate to classify them according to the technical objectives they are intended to meet. In many cases a method may be applied to cover, at least partially, more than one technical objective and this is referred to. A summary of this analysis is presented in Table I.

(i) Methods to improve the detection of the diversion of declared uranium

Conventional nuclear material accountancy is applied, including the periodical physical inventory verification and flow verification during interim inspections. To improve its effectiveness the following methods can be applied

- A **mailbox approach** associated with unannounced inspections or short-notice random inspection to increase the coverage of flow verification; this approach requires a detailed, precise and timely operational program and an authenticated data recording. It is not expected that its application present significant difficulties. Encrypted e-mail can be used to transmit the information. Unannounced inspection should be normally used because the introduction of deterrence in order to meet also other technical objectives. With very few exceptions, the Agency is able to perform unannounced inspections (2 hours maximum delay) to the present safeguarded centrifuge enrichment facilities. Anyway surveillance and containment measures can be used to increase the effectiveness of the unannounced inspection in a case-by-case basis. Short-notice random inspection can also be performed, but in this case this method would cover only this technical objective. A reduction of the number of interim inspections is not the objective and is not expected.
- **Intensive use of operators measurement system** is appropriate to reduce the mass balance uncertainties through the more precise operators load cells and precision balances. Continuous authenticated record of cylinder attachment and detachment and associated weight data allow a more frequent material balance and SWU closing. Further developments are required on the system authentication, but a solution seems quite feasible on short time. A temporary alternative could be the verification of the operators measurement system during unannounced inspection using standard weights.
- **Flow and enrichment monitors** applied to the feed, product and tails currents allow the more frequent closing of mass balance and therefore reducing the associated uncertainties. The current systems are under evaluation and should be tested at facility conditions. Further developments are necessary in order to increase the precision, to reduce the costs, and to become more users friendly, but the method would be very effective to meet all technical objectives. Information on the process gas pressure is required and this may difficult the method application in some facilities, however due to the large effectiveness gain the use of this method should be normally considered.
- **An operational SWU balance approach** may be applied to confirm periodically the SWU capacity usage declared in the operational programme. It consists on the closing of the SWU balance using a combination of integrated uranium flow and enrichment DA measurements on the input/output streams, including sampling of the feed, product and tails lines in the F/W station. This approach requires the provision of timely advanced operational information, including the projected SWU usage. The basic information however are quite similar to that provided on the mailbox approach. In addition to that the intensive use of operators measurement with the continuous and more precise data increases significantly the accuracy of the SWU balance closing.

This allows the method application also for large facilities. Adjustments due to blending operations can be easily performed if the correspondent data are provided in the operational programme or mailbox approach. This method can also be applied to meet the other technical objectives.

Table I. Summary of the available methods

		Technical objective		
Method	**Purpose**	**To detect diversion of declared uranium**	**To detect excess production**	**To detect uranium with enrichment higher than the maximum declared**
Mailbox approach plus unannounced inspections to F/W station	To verify randomly 100% of the internal and external flow and confirm that only declared cylinders are connected to process	Increasing the coverage of flow verification	Detection of undeclared feed at F/W station through unannounced inspections	Detection of undeclared feed at F/W station through unannounced inspections
Intensive use of Operators' measurement system	To improve the mass and SWU balance	Reduction of mass balance uncertainties	No	No
Use of flow and enrichment monitors	To verify the product enrichment is lower than the maximum declared To confirm the mass and SWU balance	Frequent closing of mass and SWU balances	Detection of SWU diversion	Measurement of product enrichment; Detection of SWU diversion
Operational SWU balance approach	To verify periodically the used SWU capacity during a time period	Frequent closing of mass and SWU balances	Detection of SWU diversion	Detection of SWU diversion
Surveillance at F/W station integrated with electronic seals	To verify that only declared feed cylinders are connected to process	No	Detection of undeclared cylinders	Detection of undeclared cylinders
Use of enrichment monitor	To verify the product enrichment is lower than the maximum declared	No	No	Measurement of product enrichment
Enhanced DIV (Inside the cascade hall)	To verify periodically the cascade configuration is as declared To verify the absence of non declared cylinders	No	Detection of changes of the cascade configuration; Detection of undeclared cylinders or F/W station	Detection of changes of the cascade configuration; Detection of undeclared cylinders or F/W station
Environmental swipe sampling	To verify the product enrichment is lower than the maximum declares	No	No	Detection of enriched uranium particles

(ii) Methods to improve the detection of excess production

- **Mailbox approach** associated with unannounced inspections to the F/W station can be applied to confirm that only declared cylinders are connected to the process. (See also previous section)
- **Surveillance at F/W station integrated with electronic seals** can be applied to confirm that only declared cylinder are attached to and detached from process. This method may be not appropriate for large throughput facilities due to the large number of cylinder movements. In addition to that, the necessary operators action to attach and detach seals may introduce some implementation difficulties. The method can be used however on a case-by-case basis and as temporary alternative solution.
- **Flow and enrichment monitors** allow the more frequent closing of SWU balance and therefore the detection of SWU diversion. (See also previous section)
- **An operational SWU balance approach** may be applied to confirm periodically the SWU capacity usage declared in the operational programme and therefore to detect SWU diversion. (See also previous section)
- **Enhanced DIV** inside the cascade hall is used to verify periodically that the cascade configuration is as declared, to verify the existence of valves and take off points, to verify the absence of non-declared cylinders and to verify the continuity of the header pipework to the F/W station. The DIV is performed during unannounced inspections to the cascade hall. Normally the activities performed during the DIV are: visual observation, DA sampling from the currents, NDA measurement of the process pipes and environmental swipe sampling. Photographs and surveillance can be used as complementary measures at facilities where there are restrictions to direct visual observation. The use of 3-D laser range finder may also be used on a case-by-case basis as complement for the visual observation. When enrichment monitors are not available, neutron systems seem to be a potential useful method to be applied in the next future.

(iii) Methods to improve the detection of uranium with enrichment higher than the maximum declared

The present methods applied directly to meet this technical objective are the environmental swipe sampling and enrichment monitors. The methods to detect the SWU diversion and to confirm that only declared cylinder are connected to the process are used indirectly to meet the technical objective (c). However there is a major need for timely detection to confirm the absence of uranium with enrichment higher than the maximum declared, especially high enriched uranium. Two approaches have being suggested: near real-time enrichment monitoring by improved systems and timely environmental sampling.

Reliable enrichment monitors should provide accurate conclusion in a non-intrusive way. Currently no enrichment monitor system meets all Agency needs for centrifuge cascades, the main difficulties being the very low U-235 content at low gas pressure (1 to 3 Torr) that requires a minimum diameter (>10 cm) and wall deposits that may overwhelm UF_6 gas. Currently used portable NDA equipment (e.g. CHEM) is facility specific. Alternative portable gamma and neutron measurement devices, which better meet requirements for on-site verification, may be developed in medium time. Unattended enrichment monitors are also suggested. Continuous enrichment monitors may be permanently installed at cascade. The option include γ or neutron monitoring systems and in-line mass spectrometry. The further development of non-intrusive passive γ measurement seems to be preferred in a medium time. Monitoring systems considering neutron detection techniques (e.g. plastic scintillators, optical fibers and 3He tubes) may take longer to be developed but present a good potential. In-line mass spectrometry using authenticated operators' system should be further investigated and a conclusion can be drawn in short time.

Environmental swipe sampling is the most powerful tool to meet this technical objective. Although the conclusions generally do not accomplish the timeliness requirement, the not-quantifiable deterrence effect cannot be underestimated. Anyway swipe samples collection, transportation and analysis procedures may be optimized to reduce the numbers of low value samples, to improve timeliness and to reduce costs.

5. A Basic Model Enhanced Safeguards Approach

The basic model enhanced safeguards approach consists of applying several methods in order to meet the three technical objectives and, in principle, should be implemented to all centrifuge enrichment facilities. The safeguards measures have to provide continuity of knowledge in terms of operation and inventories, adequate deterrence and timely detection. Because that, an appropriate level of redundancy is necessary and required. Additional measures can be introduced on a case-by-case basis building specific approaches, so that the safeguards objectives are reached. Considering the available methods, one can summarize the basic model as follows:

- DIV during unannounced inspections; at least the same or equivalent LFUA activities should be performed;
- One PIV once a year;
- Interim inspections, normally unannounced inspections in connection with mailbox approach.
- Use of flow and enrichment monitors, or operational SWU balance approach, or continuous use of operators measurement system or application of integrated containment and surveillance system at F/W station;
- Environmental swipe sampling inside the cascade hall, F/W station, blending area, main vacuum system

6. Potential Impact on Integrated Safeguards Approach

Significant departures from the basic approach are not expected when implementing integrated safeguards approach for centrifuge enrichment facilities due to their capability to produce unirradiated direct use material. Assuming that a conclusion on the absence of undeclared nuclear material in the State can be drawn - a basic requirement to the implementation of integrated safeguards - the relevant misuse scenario at centrifuge facilities will be the production of uranium with enrichment higher than the maximum declared, with less importance provided to the scenario of excess production and, in consequence, to methods used to meet this scenario.

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Applying enhanced safeguards approaches at centrifuge enrichment facilities

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Abstract. ABACC has been involved in the application of safeguards to small sensitive enrichment facilities since the start of its activities in 1994. The negotiation and implementation of safeguards approaches for small centrifuge enrichment facilities were important achievements that effectively showed the advantages of having a close cooperation between the international and the ABACC regional safeguards system.
Even though the experience gained initially was very important and useful, the approaches implemented were focused with priority on safeguards effectiveness and on the special provisions to protect sensitive information. As a consequence the approaches required large inspection resources, which limited their application to facilities with small installed enrichment capacities.
This paper deals with the new challenges to maximize safeguards effectiveness, the non-disclosure of sensitive information and the optimization of the inspection resources whenever the enrichment capacity is increased.
A safeguards approach that includes innovative elements was developed. Those elements include the random closing of the SWU and mass balance complemented by operational declarations through mailbox; the unannounced access to the cascade hall strengthened by the use of surveillance on some strategic points; managed visual access and comparison with reference files during DIVs and unannounced access; use of complementary conventional surveillance on connected cylinders at the feed and withdraw stations and on potential feed points; and, swipe sampling.
The potential application of new technologies is also foreseen in this approach. In the near future good progress is expected in the remote transmission of safeguards data (e.g. the mailbox declarations), the state of health of the surveillance system and the use of VACOSS seals in conjunction with the conventional surveillance on the feed and withdraw station.
The main safeguards measures involved in this approach are described. The main diversion-misuse scenarios and the correspondent safeguards measures are analyzed.

1- Introduction

The objective of this paper is to present the main elements used as control tools included in the safeguard approach applied to a commercial enrichment facility.

Even though the Hexapartite Project was taken as reference safeguards approach, new safeguards measures have been introduced in the light of the new technologies available and the need to maximize the safeguards effectiveness optimizing the inspection resources. Those safeguards measures are described in this paper and the coverage of the relevant diversion scenarios is discussed.

This safeguards approach has been developed by ABACC and the IAEA in the framework of the Quadripartite Agreement.

2- Description of the reference facility

In this paper a centrifuge commercial enrichment facility with a design capacity less than 200,000 kg SWU/y, has been taken as reference facility. The flow of nuclear material is low compared with other commercial enrichment facilities and consequently, the rate of connection /disconnection of cylinders from the process is not so high.

As usual in Centrifuge facilities with a large number of centrifuges ordered in cascades, they are grouped in modules housed in independent buildings. All the cascades are connected in parallel and the capacity of each cascade is declared in the design information. The facility will be operated to produce enriched uranium less than 5 % U235 handled in 30 B cylinders. The natural uranium used as feed material and the tails of depleted uranium will be handled in 48 Y cylinders.

In the reference facility all the junctions in the main headers between the cascade hall and the Feed and Withdrawal (F&W) station are welded. However, from a safeguards point of view consideration is given to non declared feed and withdrawal and for that reason the declared sampling points relevant for the safeguard approach are under containment and surveillance (C&S).

All the cascades have a common vacuum system and as usual in this type of facility, there is a common F&W station, where access to process weighing system is allowed for inspection purposes. The 48Y feed cylinders are connected to standard commercial autoclaves and the product and tails material are removed directly from the process in 30B and 48Y cylinders respectively.

3-Acquisition paths considered.

Scenarios like the existence of a clandestine facility or the misuse of other declared facilities are not excluded. The diverted or undeclared produced uranium could be shipped to one of these facilities for further enrichment.

Consequently, the following acquisition paths were considered:

a) Diversion of declared uranium;
b) The undeclared production of low enriched uranium (LEU) to a level less or equal to the declared maximum;
c) The undeclared production of high enriched uranium (HEU) or low enriched uranium to levels higher than the declared maximum.

The acquisition path applicable to **declared nuclear material** can be implemented

- Reporting false flow data or
- Reporting a false MUF/SRD.

In both cases, the detection of the concealment demands the introduction of gross, partial or bias defect verification of uranium and/or isotope content of selected items present during the PIV or subject to domestic transfers, complemented with the detection of the falsification of data.

The acquisition paths identified in hyphens b), and c) requires the misuse of the facility using the flexibility introduced in the design or introducing clandestine piping to modify the original configuration.

While the capacity installed is small, the misuse scenarios are an important concern from the safeguards point of view, particularly those associated with feeding the plant with undeclared feed material. The timely detection of unrecorded production of HEU remains the dominant concern.

4- Safeguards criteria requirement

In the case of facilities with an inventory/throughput higher than 1 SQ, the following activities are foreseen in the safeguards criteria:

- Periodic auditing of accounting and operating records, annual material balance evaluation and confirmation of nuclear material transfers.
- Annual PIV.
- Verification of domestic and international transfers.
- Verification of Internal flow (feed, product and tails cylinders) and inventory changes (category changes, measure discards, retained wastes, blending, etc.).
- Periodic verification of the operator's measurement system.
- Simultaneous verification of similar stratums, at different facilities, in order to prevent the presence of borrowed nuclear material during the PIV when the inventories in the other facilities, for similar stratums are greater than 1 SQ.
- Design information verification.
- Swipe sampling

Additional measures are to be implemented in order to confirm the absence of unrecorded production of direct use material, or any other misuse of the facility and to confirm the enrichment level is not higher than declared.

Under the framework of INFCIRC/153 or similar agreements like INFCIRC/435, the inspection activities addressed aim to verify that the declared nuclear material is not diverted, to confirm that the plant operates as declared and to detect any signatures of facility misuse can be implemented. The inspection activities will therefore assure that the operator's declarations about the facility are correct and complete.

5- Safeguard Approach

The following safeguards measures are included in the safeguard approach:

Design information verification.

To confirm the validity of the information provided in the DIQ and to verify that no changes have been introduced in the configuration of the cascades, main headers, UF6 F&W station, general vacuum station, strategic points and building (general containment). In addition, the absence of clandestine piping or unidentified support equipment introduced in the facility is confirmed.

This activity is carried during the PIV and in any opportunity the inspectors have access to the cascade hall (Unannounced inspections).

Swipe Sampling

This activity is carried out on random basis during the year. Swipe samples are collected following agreed procedures on those strategic areas referenced in the baseline of the facility Strategic areas located inside the cascade hall are sampled during unannounced inspections and areas located outside the cascade hall are sampled during interim inspections. In the PIV, samples can be collected at any area of the baseline.

Nuclear Material Accountancy

In order to evaluate the correctness and consistency of the accounting and operational information.

Managed visual access to the cascade hall

During the unannounced inspections and during the DIVs, all the relevant information for DIV purposes is accessible. Provisions to protect the disclosure of sensitive information are being arranged while the access of the inspectors to all the relevant information for safeguards purposes is assured. This activity is carried out following agreed procedures.

Extra Verification of U/U235 mass balance and SWU capacity usage

This activity is carried out during the PIV and on random basis during the year . This activity requires take DA samples from the feed, product and tail lines simultaneously during any inspection randomly selected, to obtain the data from the load cells at the UF6 F&W station and the provision of supporting information in advance.

The advance information given by the operator is the following:

- Amounts of F, P, and T intended to be processed for each of the next three months.
- The weights of the feed, product and tails cylinders connected to the process expressed as unified uranium (element and isotope).

- Projected SWU for the forthcoming three months period projected per month.

This information is provided to ABACC and IAEA simultaneously on monthly bases.

Operational programmes

In addition to the information requested to support the mass and SWU balance, the following operational information is requested in advance:

- Projected UF6 receipts from outside facilities.
- Projected Shipments from the facility
- Projected increment of the installed capacity
- Operational and maintenance activities in the cascade area, UF6 F&W station, and vacuum station that might have impact on the safeguards approach.
- PIV date.

Complementary measures

- C&S at the UF6 F&W station in order to maintain the knowledge on all connected feed and withdrawal cylinders.
- C&S measures on strategic points inside the cascade hall and vacuum system.
- Special coverage by C&S measures of any potential feed point in the feed line.
- Continuity of knowledge on the disconnected cylinders using VACOSS seals linked with the surveillance system.

Inspection effort

The inspection effort foreseen in this reference facility is one PIV plus the interim inspections to meet 100% coverage of the nuclear material flow.
Unannounced access is aimed at detection/deterrence of any misuse of the facility. The activities carried out during the unannounced inspections seek for confirmation that the configuration of the cascades has not been changed, that undeclared UF6 feed/withdrawal points have not been introduced and that the facility operates as declared. The installed capacity of the reference facility will increase gradually, for this reason the inspection effort established in the safeguards approach follows the provisions of the Hexapartite Project and takes into consideration the following:

- A maximum inspection effort compatible with the maximum capacity expected in the facility and,
- A minimum inspection effort compatible with the capacity of the first module.

Unannounced inspections can be triggered by IAEA or ABACC at any time in this facility.

6- Improvements introduced in this Safeguards Approach

The safeguards approach adopted for this facility does not apply any perimeter concept (neither permanent nor transitory), consequently containment and surveillance measures covers the continuity of knowledge on strategic points like cylinders connected to process and strategic points on feed, product and tail lines. The surveillance system has adequate redundancies to ensure maximal reliability of the system; however, a common failure still could arise. As an additional improvement, the system selected is prepared for remote transmission of state of health (SOH) of the server and cameras. The SOH initial testing is being programmed for the first half of 2007 and this provision would be implemented as soon as agreed procedures can be arranged with the national authority.

According this safeguards approach, the closing of the U/U235 mass balance and the SWU balance on random basis is complemented with monthly mailbox declarations.

The combination of the information requested on processed and produced nuclear materials during the last month and the advance information requested for the next three months period, allow ABACC and the IAEA to closely follow up on the production schedule, and to implement adequate planning of the verification activities in such a way that 100% of the internal flow can be verified and the timely detection of any change in the throughput or usage capacity.

Commercial available software is used for encryption and secure transmission through internet. This declaration must fulfil the following requirements agreed between the agencies.

- The declaration must be unalterable.
- Only one declaration is allowed for each period.
- It must be secure for IAEA and ABACC data bases.
- The declaration cannot be falsely denied by the National Authority/Operator.
- Only an authorized party can provide the declaration.

To achieve an adequate accuracy in the weighing system complete access and validation of the operator weighing system is required. The possibility to have an independent file where daily operational data can be recorded in parallel to the Operator data, are being considered as a methodology to validate such data.

Even though the remote transmission of the weights data is not yet foreseen in the approach, the monthly declaration complemented with the on-site independent files of daily data and surveillance provides an adequate material balance verification capability at any time during the material balance period. The verification of the calibration of load cells will be performed in any opportunity the inspectors have access to the load cells or cylinders have been disconnected from the process.

Finally, the use of Vacoss seals linked with the surveillance system at the F/W station is foreseen in this approach. This feature will be implemented once the frequency of connection-disconnection of the cylinders to the process becomes higher than the frequency of announced inspection. This tool would allow the confirmation of the operator’s declaration regarding the movements of cylinders connected to the process and allowing the operator to move cylinders out from the surveillance field of view without requiring the presence of the inspectors for such activity.

7- Acquisition paths coverage

A summary information is provided in Annex 1.

8- Conclusions.

This safeguard approach uses an ad-hoc procedure to replace the enrichment/flow monitors not yet available for commercial use for this facility. This ad-hoc procedure requires, from the facility, an operation very close to the operational program as declared in advance on monthly bases and, the continuous follow up from the IAEA and ABACC.

The randomization introduced in the safeguard approach for the mass and SWU balance closing and the balance of U235 through simultaneous DA samples from the F, P and T lines, is effective not only to detect diversion of declared flows, but also to detect misuse of the facility that implies the production of enriched uranium above the maximum declared or the use of undeclared feed to produced undeclared LEU.

Through the analysis presented in this paper we can observe that an adequate coverage for the most credible diversion/misuse scenarios applicable are met in this safeguards approach. However, some of the improvements highlighted in this paper are still under testing or implementation phase.

New technologies are continuously under development to increase the efficiency of the safeguards measures applied in general and, in particular, to centrifuge enrichment facilities. Taking into account this fact, the approved safeguard approach includes the provisions to be revised in order to incorporate new technologies and/or experiences gained in applying safeguards which could enhance the effectiveness and the efficiency of this approach. In particular, it allows the application of any further development able to improve the timely detection of any misuse of the facility or to strengthen any weak point before reaching the capacity for producing HEU in short time.

Annex 1

Coverage of the Acquisition paths

Acquisition Path	Concealed Method	Safeguards Measures
Diversion of declared uranium	- Removal/replacement with dummy, depleted ,natural or less enriched uranium - Diversion into the MUF/SRD	Closing the U and U235 mass balance through the verification of inventory and internal and external flow.
Undeclared production of LEU (≤ 5%U235) through declared UF6 F&W station	- SWU diversion. - Undeclared UF6 cylinders connected to the process at the F/W station.	- Closing the SWU balance at random (3 times a year). - Mailbox information on monthly bases. - Verification of Feed cylinders before connection to the process. - C&S at the UF6 F&W station and strategic sampling points. - Verification of connected cylinders to the process
Undeclared production of LEU (≤ 5%U235) through undeclared UF6 F&W station	- SWU diversion. - Clandestine F&W station. - Undeclared UF6 cylinders. - Undeclared empty cylinders - Clandestine piping -	- Closing the SWU balance at random (3 times a year). - Mailbox information on monthly bases. - Unannounced access to verify the absence of clandestine piping and UF6/empty cylinders. - DIV (reference pictures) inside and outside cascade hall.
Production of HEU or LEU higher than 5% U235	- SWU diversion. - Cascades reconfiguration	- Swipe sampling. - Closing the SWU balance at random (3 times a year). - DIV of cascade configuration during unannounced inspections.

Confirmation of the decommissioned status of a centrifuge uranium enrichment plant

The report is produced under the JASPAS task JPN C 01501

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Abstract. The IAEA has placed increasing priority on the conduct of comprehensive design information verification (DIV), including verification of the declared closed-down or decommissioned status of a nuclear facility. There is a defined process for the life cycle of a nuclear facility from “operating” to “closed-down” and “closed-down” to “decommissioned”. The process calls for States to submit revised design information, accordingly, based on changes of the status of the facility. There is little difficulty in verifying a change of status from "operating" to "closed-down", since that is mainly a matter of verifying that the nuclear material has been removed from the facility. It has proven much more difficult, in practice, to verify a change in status from "closed-down" to "decommissioned". To assist the IAEA in determining the decommissioned status of a facility, lists of "essential equipment" have been prepared for major facility types. However, the definition in INFCIRC/540 does not state how much of the essential equipment must be removed or rendered inoperable in order to consider that the facility, as a whole, is decommissioned for purpose of safeguards. Japan Support Programme for Agency Safeguards(JASPAS) wants to assist the IAEA in developing guidelines for several facility types on how many and which pieces of essential equipment must be removed or rendered inoperable in order to accept a State's declaration that a plant has been decommissioned. The confirmation of the decommissioned status of the gas centrifuge enrichment plant is produced as a second report of this task with reference to Japanese experience in decommissioning nuclear facilities.

1. Introduction

In recent years, the IAEA has placed increasing priority on the conduct of comprehensive design information verification (DIV), including verification of the declared closed-down or decommissioned status of a nuclear facility. There is a well-defined process for the life cycle of a nuclear facility from "operating" to "closed-down" to "decommissioned". The process calls for the State to submit revised design information as the status of the facility changes from "operating" to "closed-down" or from "closed-down" to "decommissioned". The IAEA then verifies the revised design information and confirms the declared status.

There is little difficulty in verifying a change of status from "operating" to "closed-down" since that is mainly a matter of verifying that the nuclear material has been removed from the facility (except for residual contamination). It has proven much more difficult, in practice, to verify a change in status from "closed-down" to "decommissioned".

"Decommissioned" is defined in INFCIRC/540 as meaning that residual structures and equipment essential for the facility's use have been removed or rendered inoperable. To assist the IAEA's inspectors in determining the decommissioned status of a facility, lists of "essential equipment" have been prepared for major facility types. Such lists are being regularly used by the IAEA inspectors during current DIV activities.

However, the definition in INFCIRC/540 does not state how much of the essential equipment must be removed or rendered inoperable in order to consider that the facility, as a whole, is decommissioned for safeguards purposes . Sometimes, facilities are totally demolished, in which case determination of their decommissioned status is simple. However, in many cases, some but not all of the essential equipment is removed or rendered inoperable.

It may be that the standard for accepting a State declaration that, for purpose relevant to safeguards, a facility has been decommissioned, should be different depending on whether or not the site will continue to exist. That is, a lower standard may be acceptable under circumstances that the site will remain because of the presence of other facilities / LOFs. This dimension is not addressed in this paper.

Japan's Support Programme is providing its assistance to the IAEA in developing guidelines for several facility types on how many and which pieces of essential equipment must be removed or rendered inoperable, in order to accept a State's declaration that a facility has been decommissioned. As a second step of this project, the determination of decommissioned status of the gas centrifuge enrichment plant (GCEP) is addressed using the experience gained while decommissioning the JAEA Ningyo-Toge and the Tokai JAEA R&D center.

2. Plausible proliferation scenario using GCEPs declared as being decommissioned

The most plausible proliferation scenario is the undeclared operation of a closed-down GCEP, as being decommissioned. At the time when the enrichment process is closed down and nuclear material including feed, product and tail of UF_6 are removed, routine inspections will stop. The capability to restart the isotope separation process may still exist. The diverter simply brings undeclared UF_6 feed to the facility and processes undeclared $HEUF_6$. In general, it is understood that there is an operational risk in restarting a closed down centrifuge equipment from stand point of equipment reliability, but it is not impossible. Accordingly, during the closed down stage to completion of decommissioning, the IAEA should conduct appropriate verification activities through DIV and complementary accesses to confirm the absence of undeclared activities in a facility.

Generally, the decommissioning process will involve the following steps;

a. termination of UF_6 gas feeding to the process,
b. discontinuance of centrifuge operation by cutting off the electric power supply of the equipment,
c. flushing out the residual uranium within the equipment and piping systems with IF_7 (Iodine Heptaflouride) or ClF_3 (Chlorine Trifluoride) gases (at this stage, the feed supply system and product recovery system are operated),
d. separation of the connections between the feed supply system, piping systems (cascades) and the product recovery system,
e. dismantlement of cascades including removal of centrifuges and piping systems,
f. dismantlement and removal of feed supply system and product recovery system,
g. dismantlement of electric power supply system including high frequency changer,
h. dismantlement of the on-line enrichment measurement system including mass spectrometers and gas circulation pipeline to the measurement system from several positions of the piping system, and
i. dismantlement and removal of infrastructure e,g,. cooling water supply system to the cascades.

The plant operator may change the activities mentioned in step c. and step d. He can remove the connections between the cascades, feed supply and recovery system of UF_6 (step d.) before flushing out of residual uranium (step c.). A new gas supply system will be then installed for flushing out the residual UF_6 out of the process.

The order of activities identified from step c. to step i. can differ for cost reasons according to an operator's assessment. This study does not address the dismantlement of centrifuge equipment itself (e.g., parts of rotor removed from the plant). The equipment should be further dismantled or destroyed under an appropriate verification regime both to assure no undeclared uses and to maintain confidentiality of knowledge relevant to the technical proliferation of GCEP [1].

The IAEA has a possibility to implement appropriate measures such as the application of immobilization seals and surveillance systems to the removed equipment until the start of dismantlement. At each step of decommissioning, the IAEA should verify the status of the decommissioning phase , in order to assure that the process is irreversible. Generally, necessary confirmation activities are as follows:

a. confirmation of closed-down phase,
b. confirmation of dismantlement of special equipment and infrastructures relevant to facility operations,
c. Verification of the status of removed special equipment, and
d. confirmation of further dismantlement or destruction of special equipment.

3. Plausible proliferation scenario using removed essential equipment

In the case of a GCEP, the building is a large edifice that encloses the enrichment system. Structural features of the building, except their great width and flat design[1], are not specific to an enrichment process. After the essential equipment, including the piping system for the isotope separation process, is moved out of the building, this latter is then simply a large structure that can be used for other purposes. However, since the essential equipment of a GCEP offers the possibility of a second hand use, it could be reused at undeclared locations.

This is the reason why tracking the removed essential equipment is necessary. Especially designed equipment, including parts of the rotor and the rotor itself, should be under appropriate surveillance until it is further dismantled or destroyed. However, it may be difficult and not very meaningful to attempt to track dual-use equipment and components, such as the electric supply system with high frequency changer and mass spectrometers as they can be used for other unrelated purposes. The IAEA should refer to the weighting of the indicators (*W*eak, *M*edium and *S*trong) associated with each piece of the essential equipment identified in the physical model of a GCEP, and accordingly assign a priority to the tracking efforts.

It is not possible to rebuild an operational GCEP with only auxiliary systems, dual use equipment and components, or individual equipment items, but the gathering of such equipment at one location must be seen as strongly indicating the intent of re-constructing such a plant. Therefore, the IAEA should pay more attention if such equipment is collected in one location.

4. The terms "rendered inoperable" during plant decommissioning

The term "rendered inoperable" for a GCEP relates to the step during the decommissioning stage, when the plant loses its capability for isotope separation. With regard to especially designed equipment and systems for the isotope separation, such as the rotors, piping systems and feed supply system and product recovery system, the term relates to the step when they have been removed or

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[1] The IAEA has no legal right to track the equipment removed outside a facility, even though it is still under safeguards. The appropriate verification regime, which is mentioned in this paragraph, would be possible only under a specific agreement or as a transparent measure accepted by the State/facility operator.

dismantled. Further advancement of the decommissioning stage, e.g., the destruction of the aforesaid equipment and systems reduces the possibility of re-activating the plant.

5. Recommended Verification activities at each stage of decommissioning process

Verification activities to be undertaken by the IAEA in order to assure the irreversibility of each step of the decommissioning process are as follows:

a. Termination of UF_6 gas feeding to the enrichment process

The IAEA applies seals and surveillance systems to the feed system of the plant and any remaining UF_6 cylinders. Verification of the seals and surveillance system records are carried out periodically. The IAEA continues to implement routine inspections to verify the inventories of nuclear material at the plant and to achieve any timeliness goal requirements. The routine inspection activities are to continue until the amount of nuclear material is less than 5ekg[2]. At that point, DIV and complementary access may continue as needed, but routine inspections are limited to a yearly PIV.

b. Discontinuance of centrifuge operation by cutting off the electric power supply of the equipment

The IAEA inspector can easily verify the stoppage of centrifuge operation by confirming the lack of shrill sound from cascades. When a cascade is operating, a clearly audible high frequency sound comes from each rotor. The IAEA can apply a seal to the breaker of the electric supply system to the cascades. The seal is verified periodically.

c. Flushing out the residual uranium within the equipment and piping systems with IF_7 or ClF_3 gases (at this stage, the feed supply system and product recovery system are operated)

At this step, additional verification activity is not required except verification of UF_6 recovered by flushing out of equipment and piping systems. Containers of recovered nuclear material should be sealed by the IAEA. Normally the amount of nuclear material will be small.

d. Separation of the connections between the feed supply system, piping systems (cascades) and the product recovery system

The IAEA applies a seal system to the separation points across the cascades, the feed supply pipeline and the product recovery pipeline. Heads of the pipeline connected to the cascade hole are covered by the metal plate and the seal system is applied between the metal plate and the head of pipeline. Tops of cascade piping systems are also sealed. Stoppers are applied to the top of the cascade and the seal system is attached between the stoppers and the tops. The seal system is verified periodically.

e. Dismantlement of the cascades including removal of the centrifuge rotors and piping systems

As part of design information verification to confirm the design information change submitted by the operator to declare the decommissioning stage, the IAEA verifies the status of dismantlement on a manner similar to that implemented during LFUA to the cascade hole.

f. Dismantlement and removal of the feed supply system and product recovery system

g. Dismantlement of the electric power supply system including the high frequency changer

h. Dismantlement of the on-line enrichment measurement system including the mass spectrometers and gas circulation pipeline to the measurement system from several positions of the piping system

j. Dismantlement and removal of the infrastructure e,g,. hot water supply system to the cascade

Based on the information relevant to the schedule and steps of decommissioning of the plant submitted by the operator, the IAEA confirms each status of dismantlement and removal of these auxiliary equipment and systems. For the confirmation, it is necessary to agree upon an appropriate arrangement for submissions and updates of dismantlement and removal schedules between the plant operator and the IAEA.

The IAEA can continue to carry out the confirmation activities as a part of activities in routine inspections to the plant. If the nuclear material held in the plant is less than 5ekg, the IAEA is limited to one routine inspection in a year [3]. After removal of all nuclear material, the plant is still considered as a facility until it loses the capability for enrichment [4]. The IAEA retains the right and has the obligation to carry out DIE and DIV based on the declaration of decommissioning status by the operator [5] until the plant loses the function as a facility. During that period, the IAEA has also the right to conduct complementary accesses to verify the absence of undeclared activities and nuclear material. After the facility has been decommissioned from safeguards point of view, the IAEA still has the right to carry out complementary accesses to confirm the decommissioned status of the facility.

Under the comprehensive safeguards agreement (INFCIRC/153 type safeguards agreement), one week advance notification to the state for the implementation of DIV is necessary [6]. The IAEA may need to seek the possibility to implement DIV with a short time advance notice. Generally, a DIV with a short time advance notice is a very effective measure to detect and to deter undeclared activities.

6. The decommissioning experiences in at JAEA Ningyo-Toge and the JAEA TOKAI R&D center

6.1 The JAEA Ningo-Toge

There are two enrichment plants in Ningyo-Toge. One is a pilot plant and the other is a demonstration plant. In the pilot plant, three types of centrifuge, i.e., OP1-A, OP1-B and OP-2 were installed. These centrifuges are shown in the figures -1 to -3. In the demonstration plant, two types of centrifuges, i.e., DOP-1 and DOP-2, were installed. These centrifuges are shown in figure-4 and fugure-5.

The pilot plant using OP-1A centrifuges went into small-scale operation in September 1979 and full operation with 7000 sets of centrifuges in March 1982. During the period from 1979 to 1982, improvement in separative works was achieved. As a result, improved centrifuges, i.e., OP-1B and OP-2 with 50% and 100% higher separative works than OP-1A, respectively, were installed. The pilot plant stopped operation in March 1990.

The objectives for the demonstration plant were to establish a mass production technology for a GCEP. The plant using DOP-1 centrifuges went into operation in April 1988, and construction started on the DOP-2 centrifuges. The demonstration plant with a capacity of 200 tSWU/y went into full operation in May 1989. The demonstration plant stopped operation in February 2001.

In order to accomplish the decommissioning of a GCEP, technologies are required to recover radioactive material prior to dismantlement and to treat waste material generated during the decommissioning period. Currently, the JAEA develops dismantling engineering technologies at the pilot plant and also develops residual uranium recovery technologies at the demonstration plant.

At the pilot plant, all of the cascades including the piping systems were disassembled. The centrifuges have been removed from the cascades and are stored at a designated location. The system for dismantling used centrifuge units, including rotor and inner components, was constructed at the pilot plant in 1998.

In order to provide containment against dispersion of radioactive material generated during disassembling, every component of the centrifuge is enclosed in a housing with negative pressure maintained by local exhaust systems (see figure-6). Every operation will be through gloves from outside the housing.

The disassembled centrifuges are transported by a truck to the dismantling system. The parts, segregated into large and small parts, are subjected to a clinical decontamination. The parts are placed in a container and stored at a designated location. Currently, the JAEA *voluntarily accepts* verification activities on the stored parts.

Feed system, product recovery system and other auxiliary systems, including mass spectrometers still remain at the plant. Currently, these systems are verified through DIV activities. Before disassembling the centrifuges, the seal systems were applied to the heads of pipe lines from feed and product recovery system and tops of piping systems for the cascade. However, after the cascade was disassembled, these seal systems were removed.

In the demonstration plant, residual uranium removal/recovery technology has been developed since 1996. As mentioned before, the residual uranium is the uranium adhering to the surface of components inside the centrifuges and piping system over a long period of operation. The DOP-2 cascade was used for the residual uranium removed/recovery test.

The residual uranium formed as solid uranium fluoride, UF_x ($2 \le x \le 5$), reacts to IF_7 gas or ClF_3 gas to produce UF_6. Therefore, before dismantling the cascades, the residual uranium should be removed and recovered as UF_6 by feeding IF_7 or ClF_3 gas.

Junctures between the feed system, the product recovery system and the cascade hole have been disconnected, and seal systems have been applied to the heads of pipe lines from feed and product recovery systems and the top of piping systems for the cascades (see figure-7 and figure-8).

An additional test system to feed the IF_7 gas to the cascade has been installed and connected to the DOP-2 cascade.

The seal systems are verified at routine inspections and LFUAs. At every LFUA, the IAEA confirms the absence of reconnection of cascades and utilities. The demonstration plant continues to hold a certain amount of UF_6 in the storage area, and the IAEA carries out several routine inspections including an annual PIV and shipment verification.

The pilot plant at Ningyo-Toge is under decommissioning at step e. of the process described above, while the demonstration plant at Ningyo-Toge is under decommissioning at steps c. and d. of the process. The current status of decommissioning of the facility is shown in figure-9 and figure-10.

6.2 The JAEA Tokai R&D center

The Uranium Enrichment Development Facility is a part of the JAEA R&D center at Tokai and was active in the early development of centrifuge enrichment in Japan. The facility was constructed as a R&D facility for centrifuge development using only a small amount of nuclear material. The Facility is being decommissioned. Most of the sensitive enrichment related equipment has been removed from the process area. Some contaminated equipment is stored at the facility. The facility will be renamed as a “Uranium Waste Area”.

Most of the especially designed equipment and the auxiliary system are dismantled and the waste of the equipment is stored in containers and drums. The especially designed equipment is dismantled and stored at a central location.

The JAEA submitted several design information change declarations according to the steps of decommissioning of the facility, and the IAEA carried out DIVs at the facility.

The facility is under further decommissioning process step e. DIVs will continue and the location is designated as a LOF with corresponding routine inspection requirements

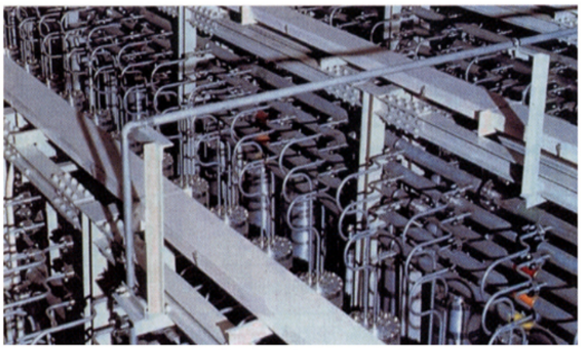

Figure-1 OP1-A centrifuge

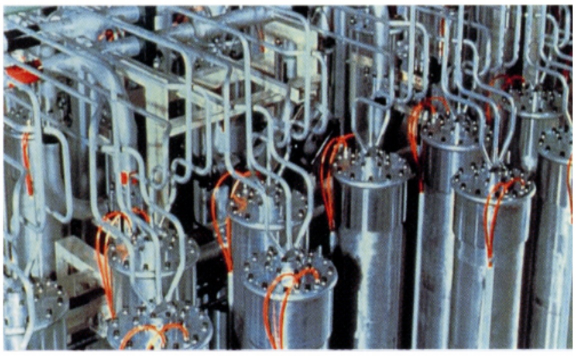

Figure-2 OP1-B centrifuge

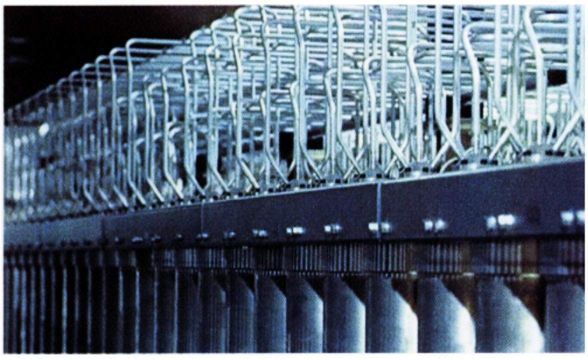

Figure-3 OP-2 centrifuge

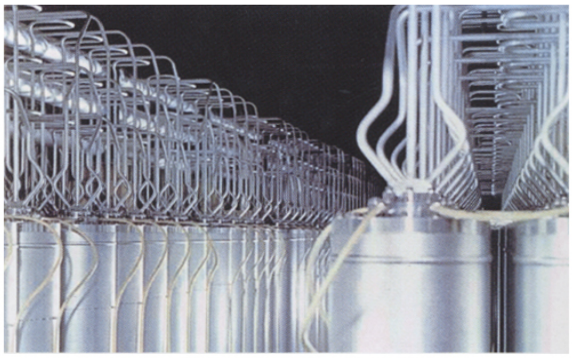

Figure-4 DOP-1 centrifuge

Figure-5 DOP-2 centrifuge

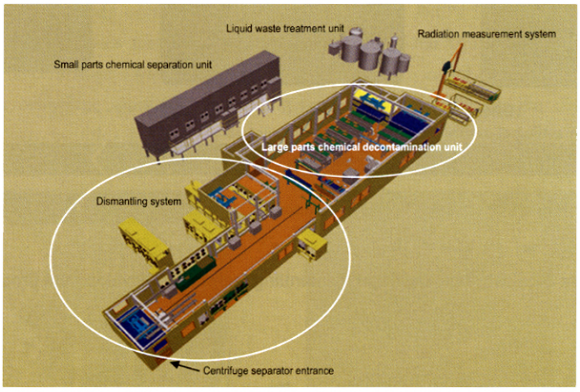

Figure-6 Dismantling system

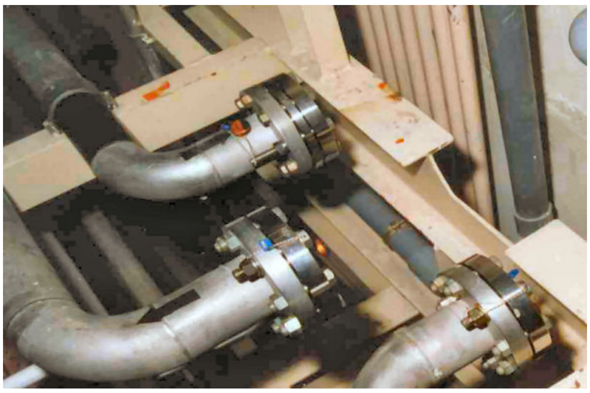

Figure-7 The seal applied to the heads of pipes

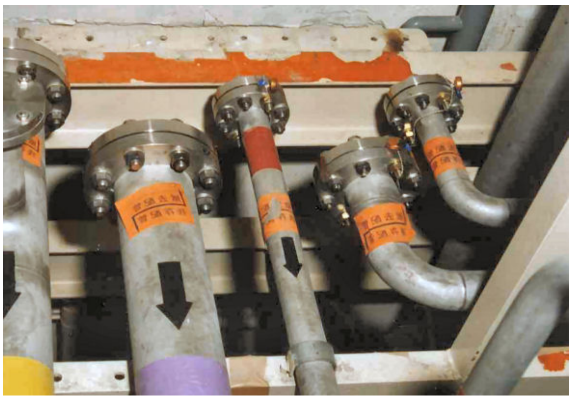

Figure-8 The seal applied to the heads of pipes

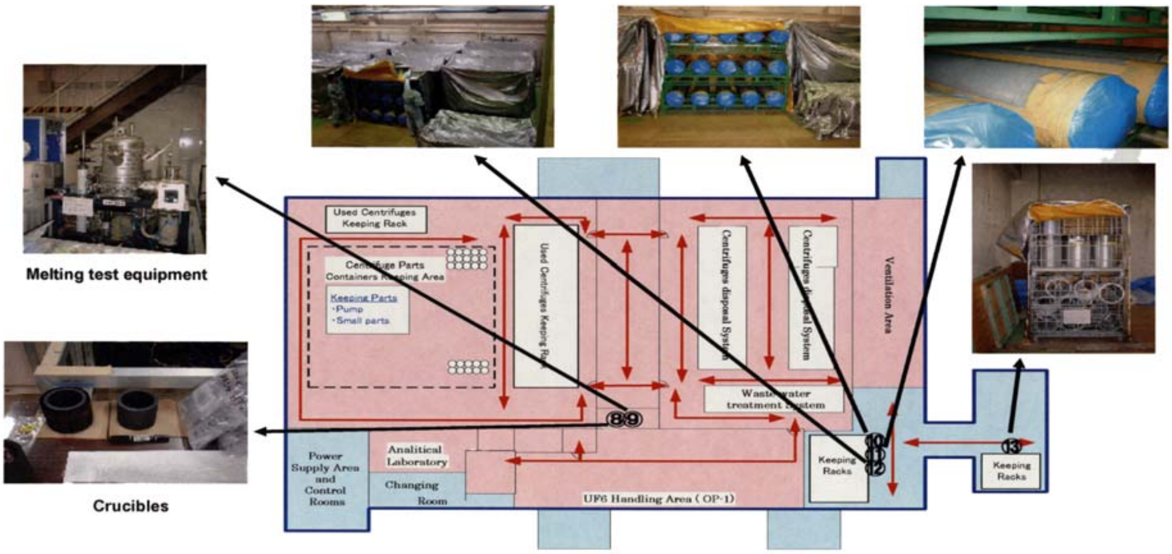

Figure-9 Site visit report of centrifuge dismantling and disposal

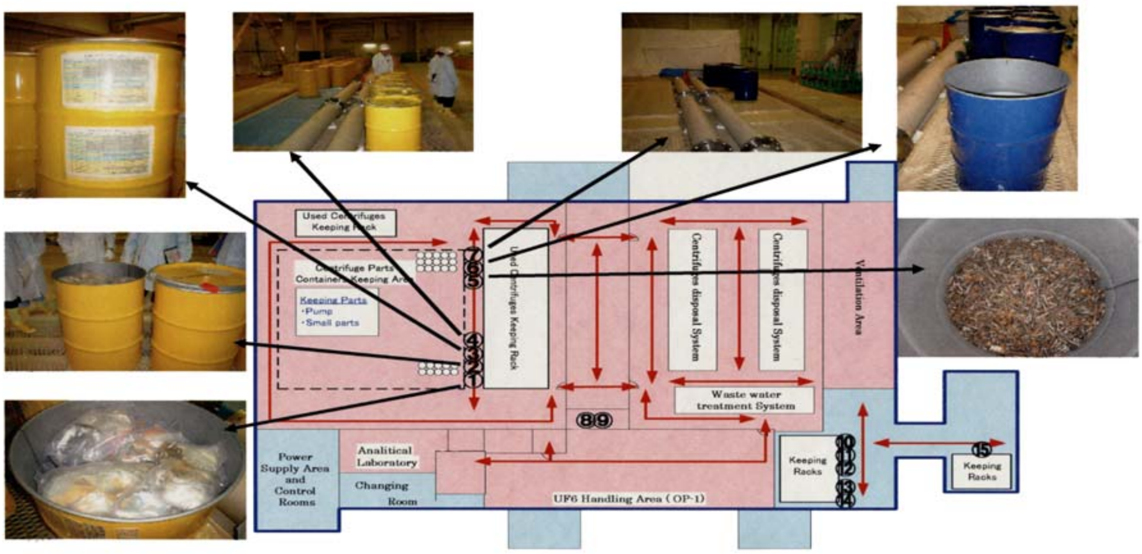

Figure-10 Site visit report of centrifuge dismantling and disposal

ACKNOWLEDGEMENTS

The authors would like to acknowledge the assistance of the GCEP operator as well as NMCC staff who assisted them in this study.

REFERENCES

[1] The Physical model suggests that a large building with great width and flat must be typical indicator of the GCEP.
[2] The Article 79 of INFCIRC/153 corrected states that "The Agreement should provide that in the case of facilities and material balance areas outside facilities with a content or annual throughput, whichever is greater, of nuclear material not exceeding five effective kilograms, routine inspections shall not exceed one per year."
[3] Article 79 of INFCIRC/153(Corrected)
[4] Article 18.i of INFCIRC/540 (Corrected)
[5] Article 48 of INFCIRC/153 (Corrected)
[6] Article 83.(a) of INFCIRC/153(Corrected)

The on-site laboratory for the Rokkasho Reprocessing Plant in Japan

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Abstract. Rokkasho Safeguards Analytical Laboratory (On-Site Laboratory or OSL) for the Rokkasho Reprocessing Plant (RRP) was established in order to analyze, in a timely manner, safeguards samples with high precision and accuracy for safeguards samples taken at RRP. The installation at OSL of the analytical instruments, based upon joint user requirements between the Japan Safeguards Office (JSGO) belonging to the Ministry of Education, Culture, Sports, Science and Technology (MEXT), the Nuclear Material Control Center (NMCC) and the International Atomic Energy Agency (IAEA), was completed in 2003. Among the instruments are two Hybrid K-edge/X-ray Fluorescence densitometers (HKED), a Thermal Ionization Mass Spectrometer (TIMS), some Density Meters and others. The NMCC as the organization for safeguards inspection is mandated by the MEXT to carry out analytical operations to safeguards samples like spent fuel dissolving solution and plutonium nitrate solution taken in RRP and transferred to OSL for joint analyses with the IAEA analysts. The NMCC has been licensed to use nuclear material samples at OSL by the Japanese competent authorities. The OSL was set as radiation controlled area by the NMCC at the time of the start of the uranium commissioning at RRP on December 2004. After start of the uranium commissioning, the functional tests and relevant training on the instruments with uranium samples and the analyses of safeguards samples were undertaken together with the IAEA analysts. NMCC analysts have shown skills in treating and analyzing safeguards samples, in communicating and coordinating technical points with the IAEA analysts and the facility operators in the daily operations. Since start of the active commissioning at RRP on March 2006, plutonium is being handled and analysed at the OSL. This paper describes NMCC's experience and current status of the OSL working stations and analytical instruments. Also working arrangements between IAEA and NMCC are presented.

1. INTRODUCTION

The Tokai Reprocessing Plant (TRP) in Japan was the first reprocessing to come under international safeguards and state system of accounting for and control of nuclear material (SSAC) in Japan. TRP has an operating throughput of about 90 tons heavy metal (HMt) per year with plutonium nitrate as final product. During more than 25 years of safeguards experience at the TRP, the IAEA and the JSGO have significantly strengthened the safeguards approach and in doing so learned how to overcome some of the inherently technical factors limited at a complex flow facility.

The IAEA and the MEXT need to design a new safeguards approach more appropriate to a large commercial scale reprocessing facility when the decision was taken in the 1980s to construct the RRP in northern mainland of Japan, with a throughput of 800t HM/year. The RRP is managed and operated by Japan Nuclear Fuel Limited (JNFL). There was a concern within the international community whether the IAEA could overcome these challenges. In order to address such challenges, a multinational forum referred to as LASCAR (Large Scale Reprocessing Plant Safeguards) was established to provide recommendations for an effective safeguards approach with resource control. This forum, which was sponsored by the Government of Japan during the period of 1988 through 1992, was composed of more than 50 experts with safeguards and reprocessing technology experiences. One of the primary recommendations was on-site laboratory capabilities. The planning to establish OSL in RRP was considered by the IAEA and Japan with recommendations from the LASCAR forum. It is necessary to apply effective and efficient safeguards for RRP. OSL presents benefits in comparison with off site safeguards laboratories as described in references [1] to [3].

The OSL at Rokkasho is the third laboratory of its kind after the two laboratories operated by the Institute for Transuranium Elements (ITU) for the European Commission in the reprocessing plants of La Hague and Sellafield, and it is the only one laboratory for international safeguards to be jointly operated by the IAEA and the SSAC at present [2][3]. Figure 1 shows a lay out on three levels of the OSL.

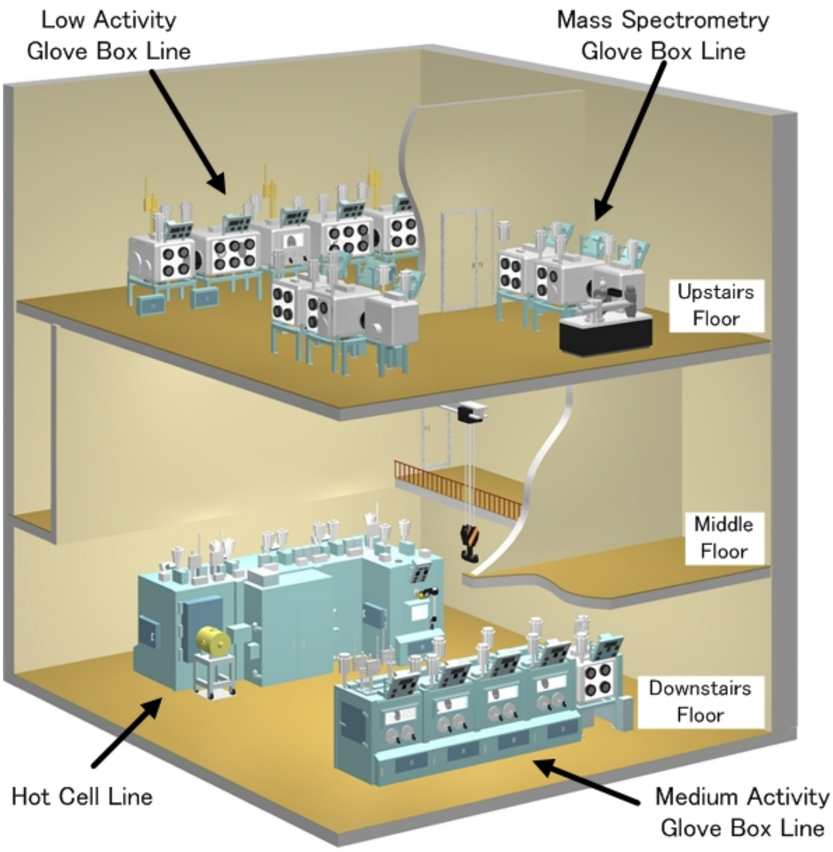


Fig. 1 Rokkasho Safeguards Analytical Laboratory (On-Site Laboratory)

2. OVERVIEW

2.1. General view of OSL

The OSL was located within the operator analytical building at RRP. A common ventilation system to the building is used for OSL. Power and water are provided by the operator. Safeguards sample treatments and analyses are conducted in the upstairs and downstairs floor of the OSL.

- Floor: each part of three floors for the analytical building of RRP

  Upstairs floor: low activity analysis area

  Middle floor: OSL access control, changing and radiation control rooms

  Downstairs floor: high activity analysis area

- Main equipments

  Hot Cells, Glove Boxes (for medium, low activity and mass spectrometry analytical lines), pneumatic transfer network (PTN) for sample, equipments for electricity, ventilation, radiation control, OSL access control and changing cloths, conduits.

2.2. Sample transfer system

The LASCAR recommendation for the OSL was to take independent samples from the facility operator's ones and to obtain timely results. One of the difficulties encountered in bulk facilities, particularly with flow processes, is the taking of representative and valid samples. Since inspectors are normally not allowed to take directly the samples from bulk materials, they must rely on the facility operator. In some cases, human observation is adequate to assure continuity of knowledge (CoK) of the sample. However, in a large scale reprocessing plant, the taking of solution samples is normally a remote operation and therefore not observable. Therefore, it was necessary to develop the means how to ensure CoK on samples taken by the operator's sampling system.

Having obtained valid samples, a timely analysis manner should be performed. This is particularly important in a high throughput facility such as the RRP. Analytical results are reported within a few days so that inspectors can evaluate also timely material flows and inventories. It also allows the inspectors to identify questionable results and to define appropriate follow-up activities, such as repetition of the sampling or of the analysis. Strict measures at the OSL are implemented to ensure the independency of the results obtained by both inspectorates. The means of authenticating samples by the operator's sampling system is achieved using of Inspector Jug Passage Detectors (IJPD) installed on the PTN [2]. The sensors track the empty sampling jug from its origin to the sampling bench and then to the inspectorates OSL. Information from the IJPD is correlated with information from the Automatic Sampling Authentication System (ASAS), the OSL and the operator declared data.

3. ANALYTICAL INSTRUMENTS AND PROCESSING

3.1. Analytical instruments

The Hot Cell Line has been set up on the downstairs floor and receives input solutions and samples contained fission products in the radiation protection cells. The analytical techniques and methods with highest sensitivity and reliability have been selected and introduced such as HKED and TIMS for IDMS. Table 1 shows the instruments and corresponding determination. The IDMS method with LSD (Large Size Dried) spike has been introduced with the benefit of the experience and performances obtained on the TRP safeguards samples. Measurements of solution density with oscillation method using of a U-tube, Uranium concentration with K-edge densitometer, U/Pu ratio with X-ray fluorescence of HKED and fission product (FP) separation work are carried out. The Medium Activity Glove Box Line has been set up on the same floor. Main analytical techniques and methods are applied for the measurements of Pu in solution without FPs and in MOX powder. The HRGS has been introduced, too. The Low Activity Glove Box Line and the Mass Spectrometry Glove Box Line have been set up on the upstairs floor. The TIMS is used for measurement of isotope composition for IDMS method.

Table 1   Analytical Instruments

Line	Analytical Instruments	Determination
Hot Cell Line	HKED	U and/or Pu Concentration [g/L]
	Neutron Counter	Cm-244 [n/s]
	Density Meter	Density [g/cm^3]
	Balance	Weight [g]
	Pu(VI) Spectrophotometer	Pu Concentration [g/L]
	FP Separation Unit	–
Medium Activity Glove Box Line	HKED	U and/or Pu Concentration [g/L]
	Density Meter	Density [g/cm^3]
	Balance	Weight [g]
	HRGS	Pu Isotopic Abundance [wt%]
Low Activity Glove Box Line	U/Pu Separation Robot	–
	Alpha Spectrometer	Pu-238/(Pu-239 + Pu-240) Ratio or Pu Low Concentration [g/L]
Mass Spectrometry Glove Box Line	TIMS	U or Pu Isotopic Abundance [wt%] (U or Pu Content [wt%] for IDMS)

The measurement uncertainties on the OSL NDA and DA methods are satisfactory in viewpoint of the ITV (International Target Value) as shown in Table 2 [4]. Measurement samples are expected to about 640 a year from input solution, output Pu solution, MOX powder and waste samples.

Table 2   International Target Values

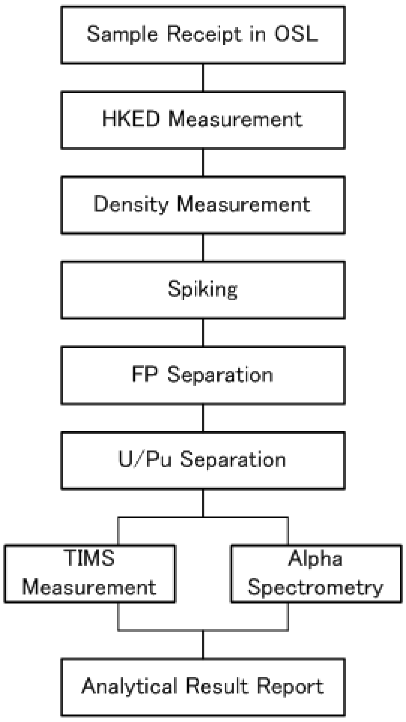
Method	Material		Uncertainty Component (% rel. Std. Uncertainty)			
			U-Concentration		Pu-Concentration	
			u(r)	u(s)	u(r)	u(s)
IDMS	U & Pu Compounds	Hot Cell Condition	0.2	0.2	0.2	0.2
		Glove Box Condition	0.15	0.1	0.15	0.1
KED	U in solution		0.2	0.15	–	–
	Pu in solution		–	–	0.2	0.15
HKED	Spent Fuel Solution		0.2	0.15	0.6	0.3

3.2. Analytical processing

The main analytical processing is devoted at the OSL for the treatments of dissolved spent fuel solution sample from the Input Accountancy Tank (IAT) and Pu nitrate solution from the Output Accountancy Tank (OAT). 100% of solution transfers from those two tanks will be subject to sampling for safeguards analyses.

3.2.1. Input solution

The sample solution issued from the IAT will primary be measured for U and Pu concentrations with HKED and then for density. About 20% of the input solution samples will be measured by a second more precise and accurate analytical technique: IDMS for the determination of U and Pu concentrations. After being spiked (LSD spike is used), separation work between U and Pu on the sample is carried out, and measure for isotopic abundance is performed by the TIMS. The alpha spectrometry is used for the correction of Pu-238/Pu-239 ratio obtained from the TIMS. The above scheme is showed in Figure 2.



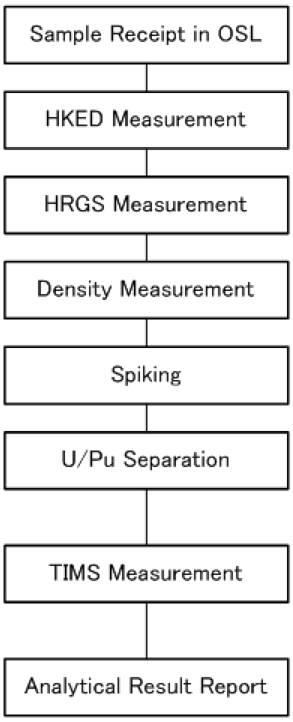


Fig. 2 Basic Analytical Flow for Input Sample   *Fig. 3 Basic Analytical Flow for Pu Nitrate Sample*

3.2.2. Plutonium nitrate solution

The plutonium solution transferred from the OAT will be measured quickly for the Pu concentration with KED function of HKED and Pu isotopic abundance with HRGS, and then the solution density. After that, about 20% of the solution samples will be measured for the Pu concentration by IDMS method for more accuracy and precision as shown in Figure 3.

4. JOINT OPERATION FOR OSL

With the start of active commissioning at RRP, the NMCC entered into a step further to the full-scale of joint operations with the IAEA. In the joint operations of this laboratory, a cooperative work that recognizes both roles and responsibilities is essential for the success of the OSL in order to achieve smooth and proper analytical operations. Such arrangements were defined in a legal document [2].

- Independency of IAEA Safeguards

  IAEA has a role and responsibility to obtain analytical results as an inspection activity with enough reliability and timeliness keeping the IAEA safeguards independency.

- Safety management and proper accomplishment of NMCC designated activity

  NMCC, as a license applicant and responsible management body of OSL based on the domestic law, has responsibility of safety management and role to obtain analytical results as a designated inspection activity with enough reliability and timeliness.

Through the discussion in On-Site Presence Meeting at Rokkasho and analysis work experienced in the uranium commissioning at RRP, common understanding of basic matter of joint use has already been promoted, NMCC has arranged the following basics for common knowledge to analysts (including IAEA) in OSL at starting time of the active commissioning at RRP, and established as Notes for the Joint Operation.

4.1. Work Management

The NMCC and the IAEA set up organization with functions and point of contact for the OSL joint operations. The NMCC and the IAEA hold periodical coordination meetings in order to achieve smooth operations based on the responsibility and duty of both sides. Both field managers (or coordinators) of the respective organisation on a daily basis control the job, such as sample receipt, analytical work, maintenance work for equipment/instrument, administrative related issues, and so on, and properly inform both organisations' OSL analysts. The analysts start the operation work according to the directions of the respective field manager.

4.2. Safety Control

To prevent radiation related contamination or exposure events at the OSL, the NMCC and IAEA establish a safety management organization and communication line, and cooperate mutually with a concerted effort. The NMCC has been licensed to use the nuclear material samples in the OSL from the Japanese safety competent authorities and measures the samples independently to the IAEA under the contract with the JSGO. The IAEA analysts also measure the samples independently with the obligation following the safety rules as defined by NMCC. The NMCC analysts (one or more staffs) attend the IAEA analytical work at OSL in view of the safety management. In case of trouble or emergency, NMCC attendant explains the IAEA analysts the situation and promptly lead them to escape as an example. For the Radiation Safety monitoring, the NMCC provides required registration procedure and education program to the IAEA analysts working at the OSL. Exposed dose records of the IAEA analysts are notified periodically. The NMCC attend the radiation and contamination check of take-out goods by the IAEA analysts and keep the records.

4.3. Quality Control

Both analysts of the NMCC and the IAEA keep the quality of sample analysis based on agreed analytical schemes and procedures. Both analysts check normal state of the equipments cooperatively before starting the job, keep the best operation condition through arrangement, and clean up of hot cells and glove boxes inside.

4.4. Preserving the independency of IAEA activities

Bearing in mind that the analysts can get access to equipments with prior consent of the field manager for starting the action, NMCC and in particular, its field manager makes proper measures to ensure that the IAEA's CoK on its samples or on its analytical activities is maintained. NMCC provides the analysts sufficient education and training to that respect. In case of troubled situation leading to the loss of CoK on samples, NMCC analysts contact immediately to the field managers so that they arrange for recovering the CoK with the IAEA field manager.

4.5. Physical protection

The NMCC issues the entry permit for OSL access-control to the IAEA analysts after an application form being accepted. Entry permit is kept at specified place in RRP office and at OSL. If IAEA needs to access to information (photos of the laboratory, equipment design for instances) that may be related to physical protection concern, it should get the permission of NMCC according to the physical protection information control rules.

4.6. Material accountancy

The OSL analysts keep the material accountancy data based on the OSL material accountancy rule. IAEA, in a timely manner, provides to NMCC necessary accountancy information from their handling and transfer of samples.

5. EXPERIENCES

Since the start of the uranium commissioning at RRP in 2004, NMCC has put in place a shift worker organisation for the safety control at the OSL, and has conducted with the IAEA various functional tests on each analytical instrument. Also, the analysis operations of nuclear samples were jointly achieved with the IAEA staff in the daytime. These experiences are as follows.

5.1. Benefits and Effectiveness of OSL

- Analytical results available within a few days

  The results by HKED and density meter can be currently obtained within a few days.

- Authenticity measures can be more easily applied at the OSL

  Authenticity of the origin of the sample received at the OSL is primary verified by the IJPD installed on the pneumatic sample transfer network lines. The hot cell and glove box that receive safeguards samples are kept under surveillance camera (Server Digital Image Surveillance: SDIS) by the IAEA. Analytical operations are carried out for especially common samples jointly used by both IAEA and NMCC analysts. For specific need to ensure CoK on unplanned situation, portable camera (All In One Surveillance: ALIS) was installed by IAEA for limited period in coordination with field manager.

- Residual Safeguards samples can be recycled into the facility operator’s process or be integrated in the facility waste management

  The residual safeguards samples have been recycled into the operator’s process through pipes of the hot cell and glove box lines.

- Reduction of safeguards sample preparations in the operator’s laboratory

  Additional sample preparations action for the transportation to the safeguards laboratory outside of the facility has been needed. Since the solution samples are automatically taken, transferred by PTN from various tank in RRP to OSL, and the IAEA and NMCC analysts perform directly the sample preparations in the OSL, sample preparations action in the operator’s laboratories was reduced. The administrative formalities and access times of the inspectorate, and the time dedicated for operator’s escorting inspectors for sample preparations were reduced.

- Sufficient nuclear material in sample to repeat analytical measurements

  The volume of solution in samples was automatically taken satisfactory to achieve when needed repeatable analysis.

- Decreasing of sample transportation from facility for off-site analysis

  To the benefit of the operator as opposed to many other nuclear facilities operator in Japan, the operator at RRP has not to prepare for off site sample shipments. Administrative procedures for requesting approval from the Government and the local government for sample shipments were therefore not necessary. Further, the analysis operation at the OSL reduced the risk for sample losses or accidents during shipment.

5.2. Analysis Operation

During uranium and current active commissioning at RRP, the adjustment and calibration work of analytical instruments using nuclear material samples have extensively been carried out. IAEA and NMCC agreed that standard samples used for quality control and calibration are prepared at IAEA-Safeguards Analytical Laboratory in Austria and then shipped to Japan. Several shipments of standards occurred. NMCC has performed or coordinated regular maintenance and repairing checks for the instruments. Monitoring of QC samples must be recorded for further evaluation and their number is expected to be raised. The situation on the main analytical instruments is the following.

- HKED: The adjustment and calibration works for the two HKEDs using nuclear material samples and standards have been carried out by ITU and vendor company's technicians. However, the instability for high voltage of X-ray unit was observed frequently. Since the samples treatment shall be IDMS (precise and accurate but a long analytical technique) during the time of HKED troubleshooting. It was decided to have a thorough review of the HKEDs maintenance plan in order to increase their reliabilities.
- TIMS and IDMS: TIMS had a function failure on the detector control circuit board. It was repaired and it is satisfactorily in operation at the present. For the IDMS, a hot plate for preparation of samples had the function failure and it was replaced. The IDMS scheme with uranium standard sample was tested and implemented on safeguards samples and it has achieved good results. The performances obtained on isotope abundance of the plutonium on standard samples by TIMS after the IDMS scheme have also presented good results.
- Density Meters: The QC based on organic standard samples (Dichlorotoluene – Bromobenzene) was good. The density meters are fully operational.
- Pu (VI) Spectrophotometer, Alpha Spectrometer and HRGS: The adjustment and calibration works for these instruments were satisfactory achieved and agreed with their specifications.
- Separation Unit/Robot (for FP removal or U/Pu separation of the IDMS scheme): The functional tests were conducted. The separation unit/robot is satisfactorily in operation. The manual separation device was prepared as back-up.
- Neutron Counter: The Cm neutron counter was developed by ITU as part of the contribution of the Euratom Support Program to the IAEA for the estimation of the content of plutonium in the hulls and in the high active liquid waste, and was successfully installed under the safety control of NMCC in 2005. It is under full operation.
- Tools for handling samples in hot cells and in glove boxes: During the uranium commissioning at RRP, the tools were not sufficiently prepared. The analysts needed careful analytical handling and more time. Many tools have been prepared up to now, and are subject to continuous review and improvements.

5.3. Joint Operation

The analytical operation modes between the IAEA and the NMCC are defined as "Common Operation" and as "Parallel Operation" [1]. For example, the confirmation of the arrival of samples taken and confirmation of the samples ID in the hot cell and glove box under surveillance camera of IAEA that NMCC cannot operate freely except emergency status is carried out by a "Common Operation" between IAEA and NMCC. "Parallel Operation" is to be carried out independently for each party. However, NMCC has confirmed always the analysis work situation of IAEA in viewpoint of safety regulation and maintenance. So, each staff has arranged the work daily in detail. Through these experiences, they have got the ability for working arrangement. Although procedure for the working arrangement has been established as Notes for the Joint Operation, detailed procedures continue to be reviewed regularly with future experience.

6. CONCLUSION

OSL at Rokkasho is the only one laboratory for international safeguards to be jointly used by respectively the IAEA and the SSAC at present and the first safeguards laboratory that was established in a reprocessing plant in Japan. NMCC has achieved the project of OSL with intensive and extensive cooperation with MEXT, IAEA and JNFL since the early phase of the designing. Analysis work for routine use of the equipment and analytical scheme has successfully been implemented with the commissioning at RRP. NMCC will continuously commit in the improvement of analytical capabilities while preserving the safety rules.

The effort and resources, which have been required to develop and implement the safeguards approach for the RRP, have been shared between JSGO/NMCC, IAEA and JNFL. The most important factor leading to the success of the OSL challenges is the open and full cooperation between all parties, JSGO/NMCC, IAEA and JNFL.

ACKNOWLEDGEMENTS

From uranium commissioning to the current phase of active commissioning at RRP, analysis work of safeguards samples taken at RRP has been carried out satisfactorily in OSL. The authors appreciate deeply MEXT, IAEA, JNFL and many persons concerned for the fruit of efforts, support and cooperation, prepare to take all possible measures for safety, quality assurance in future, and would like to ask the assistance continuously by the persons concerned. The authors would like to express their graduate to Messrs. H. Tomikawa (MEXT) and G. Duhamel (IAEA) for useful comments and helpful suggestion to this paper.

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Improvement of analytical activities in the Tokai Reprocessing Plant, Japan, by measuring destructive and non-destructive assays

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Abstract. We have been analyzing nuclear materials at the Tokai pilot reprocessing plant, Japan, since 1977. To obtain reliable measurements for nuclear material such as uranium and plutonium, we have developed various kinds of measurement techniques and implemented effective ones for accountability and verification analyses in a nuclear material accountancy system. One of our role as a pilot plant has been successfully accomplished with the effort put into various analytical activities. Now, it is time to transfer the experience gained with our technology to the next large-scale commercial plant in Rokkasyo. This paper presents our analytical methods and their results obtained using analytical techniques we have applied over recent years.

1. Introduction

The Tokai Reprocessing Plant (TRP), as the first pilot plant in Japan, started a series of operational testing for reprocessing spent fuel in light-water-reactors in 1977. Since then, we have developed many measuring techniques to establish precise and effective accountability analysis as well as safeguards analysis for use in the verification of nuclear materials.

In nuclear material accountancy and safeguards, our key role was to develop a wide range of analytical technologies in a spent fuel reprocessing plant. We have developed numerous safeguards technologies under various international cooperation programmes[1] over the past 30 years. The Tokai Advanced Safeguards Technology Exercise (TASTEX) was one of the safeguards technology improvement programmes for TRP in cooperation with four parties, Japan, the United States, France and the International Atomic Energy Agency (IAEA), and was inaugurated in February 1978. The purpose of development of safeguards technologies was to implement the safeguards measures more effectively and efficiently by technical measures so that nuclear materials, particularly special fissile materials, are not diverted during the various stages of reprocessing. The proposed research item involved the development of analytical equipment for safeguarding technologies. For example, the first k-edge densitometer (KED) [1][2][3] system for measuring plutonium product concentration was installed in our analytical laboratory. Fruitful results were obtained from the research work and demonstration tests at TRP, thus concluding TASTEX in May 1981.

The Japan Support Program for Agency Safeguards (JASPAS), where Japan supports the development of safeguards technology for the IAEA, started in November 1981. Japan was convinced to improve the reliability of safeguards system in active cooperation with the IAEA not only for promoting the development of safeguards technologies but also for the practical use of research and development results. We have actively participated in the programme as one of the major implementating organisations in Japan. The JASPAS programme covers a wide range of safeguards issues including measurement methods and techniques. The KED system, which measures the concentration of plutonium product in a non-destructive manner, finished its demonstration testing and was presented for inspection in September 1982. An automatic gravimetric sampling system[1] for pre-treatment of inspection samples that was to be shipped to safeguards laboratories, and a high-resolution gamma spectrometer system capable of providing the isotopic composition of plutonium products[1] were successfully applied during the routine inspection at the beginning of 1988. Despite only minor nuclear material accountancy, further safeguards requirements against waste streams of reprocessing have encouraged us to study some measurement methodologies for detecting small amounts of plutonium in waste matrices, especially highly active liquid waste (HALW). Through a series of research work in JASPAS[4][5][6], we have investigated various candidates to fulfil the safeguards purpose and found that conventional spectrophotometry is suitable for on-site verification as an inspection tool [7][8][9][10].

As mentioned above, we have developed safeguards technologies mainly for non-destructive assay (NDA) through international cooperation to enhance reliability in the area of safeguards. On the other hand, we have also improved our vast experience in destructive assay (DA) with isotope dilution mass spectrometry (IDMS). As a result, various effective and efficient analytical techniques have been developed, some of which were presented to the IAEA and the Japanese government as inspection tools, and were actually used for routine inspections as well as for accountability analyses. In the present paper, we review recent improvements in IDMS, currently updated KED of plutonium products, robotised sample preparation for safeguards off-site analysis and on-site verification measurement of plutonium in HALW, which are all recently established and utilised in the system of safeguards and accountability at TRP.

2. Isotope dilution mass spectrometry for spent fuel dissolver solution

IDMS[11][12][13] for accountability analysis of uranium and plutonium in reprocessing solutions has been demonstrated for many years at the TRP. In the early years, we used to apply an isotope dilution spike containing 1 mg ^{233}U and 10 μg ^{242}Pu in liquid form until 1989. The sample preparation using the spikes requires an accurate dilution of sample solution before spiking, which might be a reason for error propagation. On the other hand, the safeguards sample was diluted, aliquoted and sent to safeguards analytical laboratories for their independent measurements using a different type of spike. Dealing with such individual analysis, there was a possibility of a significant difference between the declared and inspected analytical results by using separate reference materials, diluting and spiking at both parties. As an attempt to prevent such an unexpected result, an IDMS analytical procedure using different type of spike was introduced in 1990, which allows spiking with an undiluted sample in a common procedure at a laboratory. The spike consists of different isotopic compositions from the previous one and contains rather larger amounts of dried reference materials—40 mg uranium with 20% ^{235}U and 2 mg plutonium with 97% ^{239}Pu, a so-called large size dried (LSD) spike[14][15][16][17]. In the IDMS analysis for input solutions, LSD spikes have been provided by the IAEA and used in common mode since then to ensure error-free spiking in the preparation of each separate spike. Subsequently each laboratory has performed mass spectrometric measurement using their own equipment. There are three thermal ionization mass spectrometers (TIMSs) for high-precision isotopic measurements in our laboratory. To improve the precision of isotopic ratio measurements, a technique using total-evaporation was employed in an effort to eliminate mass discrimination in conjunction with a multicollector mass spectrometer. An experience of over 10 years with remarkable success have confirmed better performance data here with an average precision of 0.1%, as shown in FIG. 1, where straight-line approximations for uranium and plutonium indicate a step-by-step improvement in their precisions. To ensure the current performance, we also estimated the expanded uncertainty of IDMS for input analysis followed by the International Standards Organisation (ISO) Guide for the expression of uncertainty in measurement (GUM) [18][19] and found it to be within 0.1%.

IDMS has also been applied to accountability samples of plutonium nitrate solution and highly active liquid waste, etc. According to the type of sample, we have prepared different kinds of isotope dilution spikes, optimising appropriate amounts of uranium and plutonium required in each measurement. It can therefore be said that high-performance measurement techniques using IDMS is certainly established in the field of nuclear reprocessing research. The quality of IDMS has also been proven by inter-comparison analyses with the IAEA and the National Nuclear Material Control Center (NMCC), and by an external evaluation of laboratory performance by regular inter-laboratory programmes in the field of safeguards (Qualité du Résultat d'Analyse dans l'Industrie Nucléaire: EQRAIN).

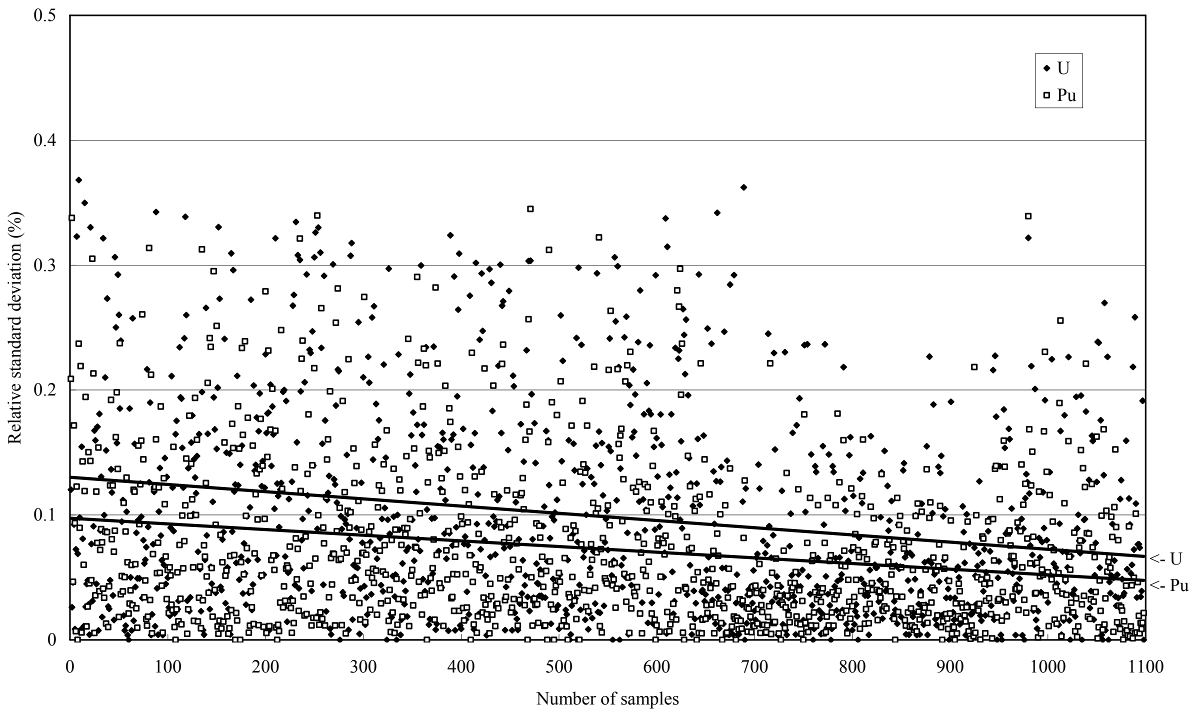

FIG. 1. IDMS uncertainty for U and Pu in input samples observed over a period of 10 years

3. K-edge densitometry for plutonium product solution

A series of development and installation of KED instrumentation for plutonium product solution was executed through TASTEX and JASPAS, for on-site verification measurement as an inspection activity by national and international inspectors. After successful implementation, a further updated KED[20] was installed in the laboratory to improve its performance, focusing on the stability of the X-ray source, in 1994. The KED system was equipped with an X-ray generator that differs from the previous system with gamma-ray sources using radioactive isotopes such as ^{57}Co and ^{75}Se. To apply the improved KED system to safeguards measurements, one of the more difficult problems for authentication of the measurement system was the control of reference material of the plutonium in the liquid state because of its evaporation and radiolysis properties. Thus we examined a solid-type plutonium reference material in a glass matrix that can be expected to retain stability for a long-term. Our experience of using the vitrified reference material helped us in confirming its effectiveness and long-term stability from numerous measurements in actual verification activities for plutonium nitrate products during reprocessing, physical inventory verification and interim inventory verification. Figure 2 shows the most recent trend of measurement control bias testing using the reference material, which is calibrated with the instrument calibration constant in each period of time. At TRP, we performed accountability analysis for each batch of plutonium nitrate product as well as the KED measurements, for safeguards purpose. As part of the authentication of KED and verification of operator’s measurement system, a sample is taken by an inspector and sent to the IAEA and NMCC for DA on the 10% random basis. For the calibration of KED, we used DA results obtained from three parties compared with the same set of KED measurement results. Consequently, the KED has greatly contributed to safeguards analysis of detecting gross defects in plutonium with a measurement uncertainty less than 0.5%, and played one of the most important roles in the safeguards activity in a timely manner at TRP.

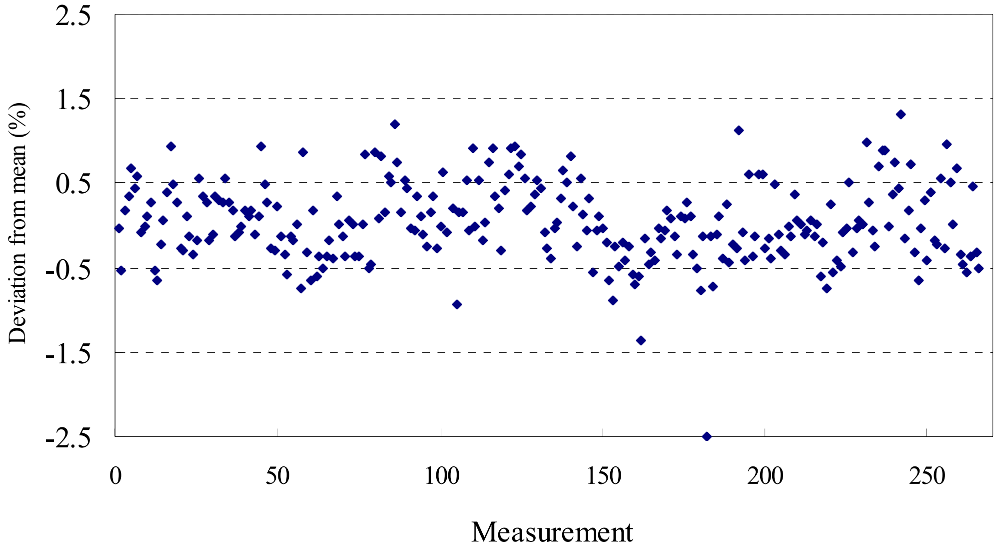

FIG. 2. KED results obtained from a control sample observed over a period of 6 years

4. Robotised sample preparation

Plutonium product is basically measured by KED in the analytical laboratory, but a DA sample for off-site analysis is also taken on a random selection basis or in case of outliers of KED measurement. To achieve highly accurate sample treatment in a glove box with aliquots of a few milligrams and to minimise operator dose from radiation, a robotised system for the automatic preparation of plutonium nitrate solution has been developed and applied to routine inspections. The first robotised system, whose main part was a specially designed robot arm, was installed in 1985 as a JASPAS task[21]. However, the movement of the arm was too slow for routine inspection to fulfil the requirement of safegurads sample preparation in a limited time. Eventually, an analytical operator was used to perform the sample treatment by hand in the presence of an inspector. Also, we met with some mechanical problems and found them difficult to repair in the glove box, because the system was made up of parts that were not commercially available. Taking faster operation and simpler maintenance into consideration, we have designed and developed an improved system applying commercially available apparatus as much as possible. The new system has a six-axis industrial robot, which is mounted at the centre of the glove box floor, and other equipments such as an analytical balance and autoburette at the circumference as shown in FIG. 3. The robot arm weighs an inspection sample in a flask, adds nitric acid to dilute the sample to a permissible concentration for shipment and takes an aliquot in a vial. We demonstrated a series of sample preparations on actual plutonium product solutions using the system and found that the operation time required was the same as that by a skilled operator. Through an acceptance test for inspection use, the system was then implemented for inspection for DA preparation. This system also has a function of sample preparation for IDMS analysis, which is used for the operator's declaration of plutonium product concentration as well as preparation of LSD spikes. Using this system, the measurement uncertainty was evaluated to be less than 0.1 rel.% (95% confidence level), which is the same as that expected of a talented operator. Therefore, the robotisation in analytical chemistry makes it possible to treat safeguards samples with a high degree of accuracy.

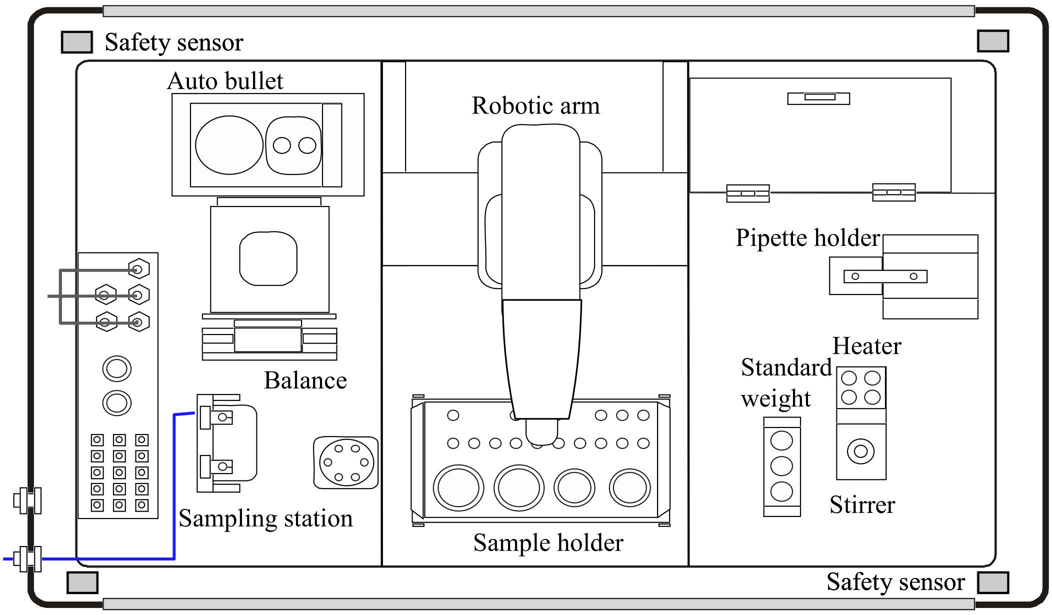

FIG. 3. Layout of robotised sample preparation system in a glove box

5. Measurement of plutonium in highly active liquid waste

In accordance with the safeguards criteria, much of the attention was paid on the amount of plutonium in HALW generated from spent fuel reprocessing. The plutonium in HALW of TRP has been specifically examined as an object to be inspected. Since 1993, HALW samples have been taken for inspection and transported to safeguards laboratories for verification analyses using IDMS. Sample preparation at TRP requires spiking, dilution and taking aliquots in a transportation vial for subsequent independent measurement by an inspector's own equipment. This method is very time-consuming for obtaining analytical results. Therefore we proposed on-site verification analyses for plutonium in HALW and have studied some analytical methodologies to achieve rapid, simple and sensitive measurements. As one of the candidates, a spectrophotometric determination method for plutonium in HALW has been developed for safeguards verification analysis[7][8]][9][10]; this method offers reduced sample preparation and shorter analysis time compared with traditional techniques such as IDMS. In this method we use neodymium as an internal standard to improve measurement reliability and to authenticate the measurement scheme and instrument conditions. This method was validated through a series of comparison experiments with IDMS. The analytical results of plutonium in HALW using this method were in good agreement with those obtained using IDMS. Consequently, the proposed method has been successfully applied to independent on-site safeguards analysis at TRP and a neodymium internal standard has been provided by the IAEA, so that an inspector can check the instrument conditions and the analytical scheme as well. Figure 4 shows data obtained so far in comparison to IDMS. The results indicate a mutual agreement between both the methods and that spectrophotometry is acceptable for verification analysis for plutonium in HALW.
Accordingly, spectrophotometry has been applied to accountability analysis of plutonium in HALW for operator's declaration and has been implemented for verification analysis with neodymium to national and international inspectors, instead of IDMS, since 2003. In a routine procedure of safeguards systems for plutonium in HALW, we determined the concentration of plutonium by spectrophotometry and made a declaration for the HALW sample. The IAEA selected samples on a random basis after the declaration and then verified the measurement using their internal standard and our spectrophotometric system, which has already been authenticated as a common instrument.

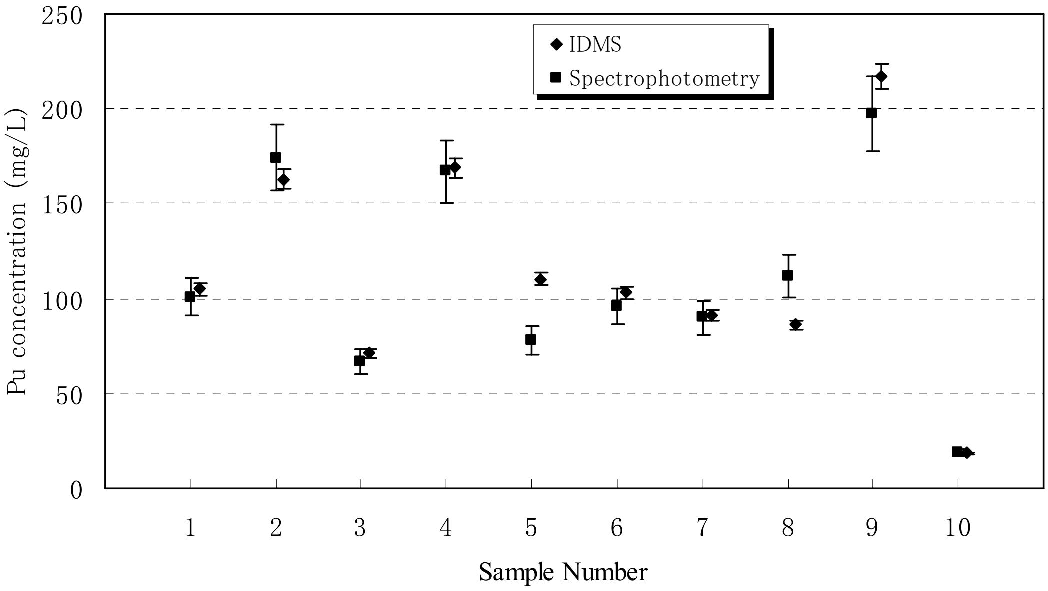

FIG. 4. Comparison between spectrophotometry and IDMS for determination of plutonium in HALW

6. Conclusions

Through research and developmental activities with DA and NDA measurements to meet the safeguards' demands, we have successfully established analytical technologies in each part of the key measurement points for safeguarding. This conclusion has been confirmed by exercises done in the currently operating Tokai reprocessing plant. To maintain our measurement quality at a high level, we have also performed control measurements and calibrations in accordance with an internal quality assurance system as well as an external quality-control programme. Such efforts help us to produce accurate and precise analytical activities under routine conditions, and to bring successful contributions to practical aspects. We have also been working on a series of testing for the on-site safeguards laboratory at Rokkasyo in the JASPAS framework. Our previous experiences have helped us contribute to the analysis performed by the inspection side as well. As a next step, we will transfer our experience to the commercial reprocessing plant at Rokkasho, Japan, and will pursue studies on analysis technology for advanced reprocessing plants in the future.

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Authentication and sample chemistry: A new approach at the Rokkasho Reprocessing Plant on-site laboratory

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Abstract. The On-Site Laboratory (OSL) is committed to providing the IAEA with reliable, accurate and timely results of the inspection samples taken at the Rokkasho Reprocessing Plant (RRP). The OSL is an important part of the efforts to safeguard adequately this large reprocessing plant. It is located on the premises of the RRP, which helps to resolve the timeliness dilemma. The OSL is operated jointly by the IAEA, the Nuclear Material Control Center (NMCC) and Japan Safeguards Office (JSGO). This joint task requires addressing new challenges in destructive analysis (DA) and the sharing of instruments, space and procedures in order to reach the best analytical results possible.

The inspector-analysts make great efforts to achieve excellence in the sample chemistry and to ensure that the procedures and results are adequately authenticated. Because the instruments are jointly used, new approaches for the implementation of measures for authentication and continuity of knowledge have been designed and put into practice. The authentication measures include securing the instruments and the data produced. Additionally, special attention is given to maintaining continuity of knowledge of the samples that undergo chemical analyses, securing the procedures and considering measures of deterrence. All these measures build a relatively solid framework for independent DA. It must be understood that a 100% assurance for a tamper-free operation is a great challenge, and the IAEA aims to achieve the best authentication under the given situation. The implementation of authentication in the routine sample chemistry requires additional efforts on the part of the IAEA and has an impact on the time needed to perform the work, compared to the activities of a normal nuclear laboratory. This paper describes the authentication policy in the OSL, the specific measures implemented and the range of confidence expected in different procedures.

1. Introduction

When Japan decided to build the RRP about two decades ago, it became clear that an on-site laboratory would be needed to fulfil the IAEA commitment to safeguard the large reprocessing facility in a reliable, accurate and timely manner [1]. One of the initial ideas was the set up of a fully independent IAEA laboratory on the RRP site. One reason for this approach was that full sample authentication and independence would have been simple: the IAEA alone would hold the key to the lab. This approach has been put into practice by EURATOM in La Hague and by Sellafield, United Kingdom. Eventually this idea was put aside because of financial and safety reasons. Instead, a jointly operated laboratory, with analysts from the IAEA, the Nuclear Material Control Center (NMCC) and the Japan Safeguards Office (JSGO) was approved and implemented. The operations of the OSL have been described in detail in another paper [2].

During the planning sessions for operating the OSL, the parties came to the agreement that the IAEA would implement special authentication measures to assure the continuity of knowledge of not only the samples but also other components of the analytical process. The authentication requirements for the joint operation as well as most of the laboratory managerial framework are described in a formal document that in spite of having the misleading name “Working Paper” [3] is paramount to the management of the OSL. Additionally, the IAEA proposed and reached agreement with the NMCC on the authentication measures that are in force in the laboratory. These arrangements are reflected in the document “OSL Authentication Measures”, which includes a table with all authentication measures that have been accepted for implementation. The measures include JSGO/NMCC observations because they have the responsibility for the safety operation of the lab. The formal agreement [4] between the IAEA and the Japanese safeguards authorities for the mode of operation of the OSL is a Memorandum of Understanding that has been signed by the responsible directors of both organizations.

The work at the OSL has two important aspects that are continuously followed in daily operations by the inspector-analyst: (a) the quality of the analytical data, and (b) the continuity of knowledge of the samples analyzed for safeguards purposes. Briefly defined:

— Quality control (QC) measures serve to minimize the accidental factors that could alter the true value of a sample.

— Authentication measures serve to minimize the intentional actions that could alter the true value of a sample.

2. The set-up

2.1. Sampling

Sample taking for verification purposes is initiated at the independent jug feeding machine (IJFM) which is installed in the OSL. The machine can hold 40 jugs which have been vacuumed by the inspectors. Authentication of automatic samples taken by the operator is assured by the automatic sampling authentication system (ASAS) to confirm that the requested material is sampled and to detect any tampering with samples during their transfer to OSL. [5]. When a sample is taken, the jug is sent through the pneumatic transport network (PTN) to the sampling bench and then back to the OSL analytical cells. Its path is registered by independent jug passage detectors (IJPD) which send secured information to the IAEA’s I3S regarding position and time. A special application reconciles the data with expected fly paths and gives an authentication approval. The working instructions “Authentication of Samples” is a guideline that the inspectors must follow.

The sample jugs have an identity code known only to IAEA staff and the code is confirmed on receipt of the sample in OSL and cross-referenced to the order in which the jugs were originally loaded into the IJFM.

2.2. Receipt and storage

When the samples return to the OSL, they arrive at one of the two locations which are secured with a server digital image surveillance (SDIS) system (Figure 1). The system is operational at all times and is reviewed every other day. The SDIS consists of redundant cameras for the front of the cells and for the back of the cells which have PADIRAC hatches for waste removal.

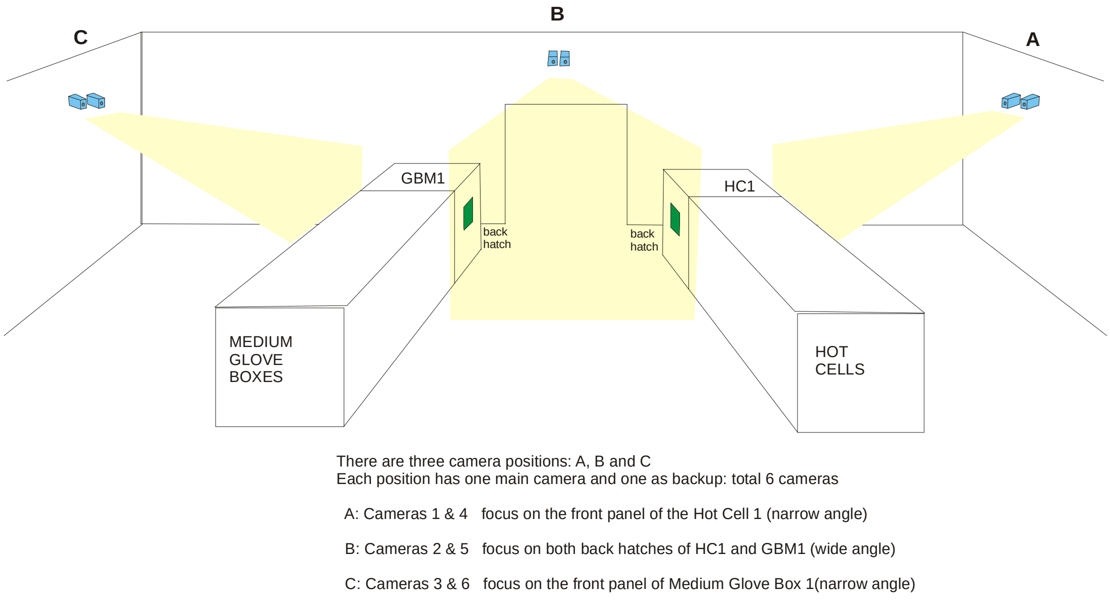

FIG. 1. SDIS camera positions in the OSL.

The inspector-analyst checks the samples at the PTN basket and stores them in dedicated shielded storage boxes inside the cell. For each sample (possibly several jugs) taken, a unique analytical data sheet number (ADSN) is issued by the operator for each jug. He reports the ADSN and the jug identity code for ASAS verification.

2.3. Confirming the density

The density of a sample solution is measured in the OSL with a precision of 0.01%. This value is then compared to the one measured by the solution measurement and monitoring system (SMMS) at the sampling station in order to verify that it is the same solution. Usually, an agreement of 0.1% is achieved, which confirms the origin of the sample with high probability. The density difference of solutions from one tank to another is much greater.

2.4. Sample handling

Some samples are taken and analyzed jointly with the NMCC. Due to material limits in the working cells, interim inventory verification (IIV) and physical inventory verification (PIV) samples are kept under IAEA continuity of knowledge but are analyzed in common mode. The samples from the accountancy tanks are taken in duplicate and the IAEA analyses them alone in parallel mode. These results are not disclosed to the NMCC until the final operator declaration

3. Scenarios

Tampering scenarios have been studied and discussed in working groups that include the safeguards inspector-analysts based at the IAEA Safeguards Tokyo Regional Office (TRO), the task group staff of the Japan Nuclear Fuel Limited (JNFL), experts from the Department of Safeguards, Division of Technical Support (SGTS) [6] and analysts from the IAEA Safeguards Analytical Laboratory (SAL) at Seibersdorf, Austria. Different scenarios have been considered and measures to build a strong authentication framework has been put in practice, taking into account mainly how the sample flows within the working cells and how the instruments are setup.

It must be assumed that tampering will be difficult and extremely risky during the physical presence of the IAEA inspector-analyst at the OSL. Tampering could be considered more probable during unattended hours and especially in the time frame before the inspector-analyst formally acknowledges the sample as correctly received in the analytical cells hot cell1 (HC1) or glove box1 (GBM1).

What does a potential tampering scenario look like? What does the tamperer want to do?

The tampering aim could be to change the elemental concentration, and/or to change the isotopic composition of a sample. Therefore the sample receiving areas are kept under strict surveillance, schematically shown in Figure 1. If there is a breach in the continuity of knowledge through camera failure or unauthorized access at the storage locations, the consequence is that all the samples in the cell are considered lost and useless for safeguards purposes.

As soon as the first analytical procedures are in progress, the sample will be measured for U and Pu concentration and then split for a second analysis. It is less probable that tampering of the sample would be done after this stage because contradictory results would make it evident. Not all scenarios show the same tampering probability. For example, the importance of continuity of knowledge in the analysis of Pu concentration in the OAT is greater than the continuity of knowledge of the analysis of ^{238}Pu by alpha spectrometry

4. Work process

To avoid modifications to the real elemental concentration and/or the isotopic composition, the following has been authenticated:

— The nuclear material,

— The analytical instruments, and

— The analytical data.

4.1. Nuclear material

In the initial phase, the authentication of sample taking and sample transport from and to the OSL is the responsibility of the RRP coordinating inspector and the site officer; however, once the sample arrives at the receiving cells HC1 and GBM1 in the OSL, the responsibility lies with the inspector-analyst. The samples and the standard material are kept under strict continuity of knowledge until no more authentication is needed. As soon as the final analytical results of the operator declaration are known, the continuity of knowledge on the sample and its data is relaxed.

Normally, extra work is undertaken by the inspector-analyst to hide the true identity of the nuclear material. This has to be done in a way that a tamperer would have serious problems in judging which vial to manipulate. He should not risk tampering the wrong one that could betray his activities. For this reason each sample receives a new laboratory internal number (LIN) that is related to the ADSN in a logbook kept under IAEA custody.

The analytical procedures require the use of standards and spikes for the analytical measurements. These standards are kept under strict continuity of knowledge and cross-checked on a regular basis because the accuracy of the samples depends on them. Additionally, a special IAEA safe is used to store this material where up to 20 g of ^{235}U and 1 g of Pu have space in the safe room.

4.2. Analytical instruments

The analytical instruments have been jointly agreed upon and were provided by the NMCC for the use of both parties. In this way, the IAEA inspector-analyst can work alone in the OSL, with hands on the instruments, without the presence of an NMCC analyst. The inspector-analyst is also a qualified radiation worker under Japanese regulations. Operating the instruments gives him greater independence than what is possible in similar inspections done at other facilities, as for example at the Tokai Reproccessing Plant (TRP) in Japan. The inspector-analyst is thus able to address the authentication issues with special handling procedures that are described in the laboratory working instructions.

The authentication approach takes into account that the instruments could be potentially tampered with cabled or wireless electronic signals. Therefore, rather than securing the hardware, the authentication is based on hiding the identity of the sample at the measurement position. This is done in the absence of human or video observation. The identity and sequence of the measurements are unknown to others and therefore difficult to tamper with. Security is based on making it difficult for an adversary to know when a real sample, versus a standard or a bogus sample, is being analyzed so that controlling the instrument to give erroneous results would be risky.

4.3. Analytical data

Most of the analytical results should be exchanged at a certain stage with the NMCC, as agreed upon in the OSL working paper. This occurs usually after the final operator declaration. The NMCC and the IAEA check their performance against the operator's report. Of special interest to the chemists are the comparisons of the blank and standard samples which are vital for the good performance of the instruments and for which QC charts are updated.

Other data, such as the laboratory logbook, the working paper with the secret sample ID (LIN) and surveillance disks are kept under "lock" or encrypted because they are part of the authentication system. The inspector-analyst keeps most of the data in paper form in a secured steel cabinet in the data room. Additionally, information is stored/backed up on laptops on pretty good privacy (PGP) encrypted drives and on safeguards servers at the RRP and the TRO.

4.4. Redundancy

Redundant measurements are a powerful authentication tool. Some measurements will be repeated using different analytical techniques on a random basis or when required. For example, most samples are first measured by hybrid K-edge (HKED), and on a random basis 20% of the samples will be analyzed by isotope dilution mass spectrometry (IDMS). Additionally, a few samples will be randomly selected and shipped to SAL for further verification. In both cases there is an authentication purpose in addition to the QC enhancement of the measurements done in the OSL.

4.5. Hidden identity

Because of the complex setup of the instruments and the need to account for potential tampering with cabled or wireless electronic signals, the authentication approach has been directed toward keeping the real identity of the sample hidden until the measurement has been completed. The inspector-analyst will not only measure one or more safeguards samples in one 'working batch' but will also measure standard and old samples in order to avoid disclosing the material in the measurement position. In case only one sample has to be measured, one or two additional bogus samples of the same type will be included in the working batch.

This handling places an additional burden on the inspector-analyst who has to keep track of the samples with false names and at the same time be careful about not making them public until the end of the process. A set of working papers and the daily logbook that are filled carefully by hand assure that no sample identity mix up occurs. These papers are kept under strict continuity of knowledge at all times.

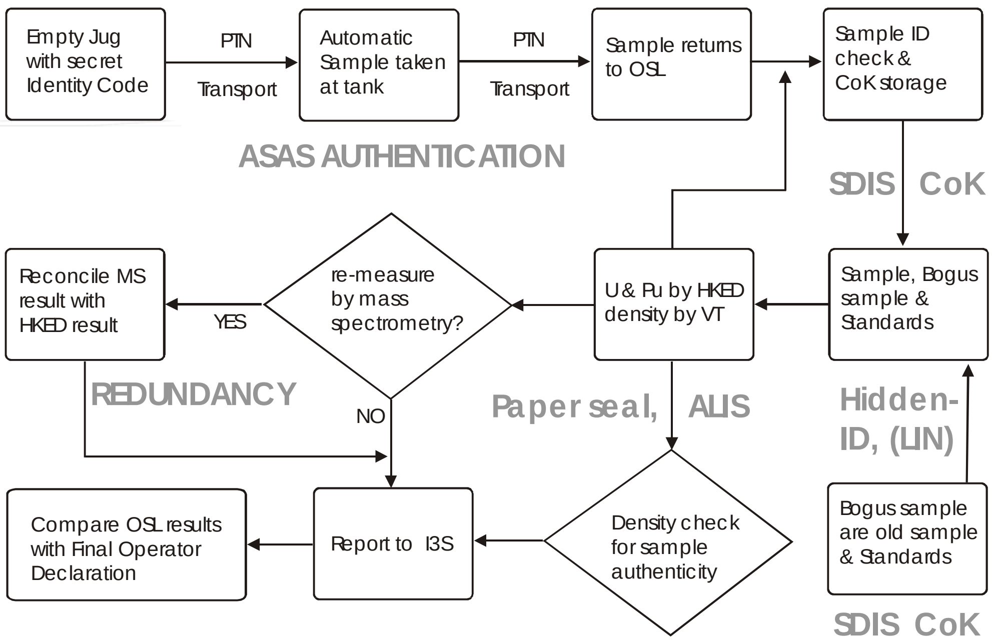

FIG. 2. Scheme of authentication measures.

4.6. Reporting time

From one same sampling point, three sub-samples are taken: one for the IAEA, one for the NMCC, and one for the operator. The reporting time or time for approval of the final operator declaration is crucial for authentication of the data and is done at the end of the process. The operator will not know the result of the IAEA's analysis until he has made the final declaration. In this way he will not be able to tailor his declaration using measurement inaccuracies in order to divert material.

4.7. New situation

The analyst must work accurately on the chemistry and the instruments and also keep an eye on the authentication requirements. This is a new situation in laboratory activities and requires that the inspector-analyst be alert at all times. Figure 2 shows schematically a typical workflow and the authentication measures involved. The authentication procedures have been documented and every new chemist needs to understand the approach and undergo a training period to be fully proficient in their implementation. A short paper "OSL Authentication Basics" is provided to each newcomer.

5. Set-up check

The setup has been discussed in various stages. At an early stage Keith Tolk, an authentication expert in at the IAEA/SGTS, pointed out that given the complex setup of the instruments and shared space with the NMCC, hiding the identity of the samples would be one of the best achievable approaches. The concept has been put into practice and gives good assurance of independency.

In October 2005, a group of experts from the Institute for Transuranium Elements (ITU) and SAL made an auditing visit of various days at the OSL. Their evaluation was positive [7] and some important recommendations were made. The recommendations included that the IAEA work independently to obtain results for IAT and OAT samples, that analysis results be discussed only after the final declaration, and that the analysis of samples be done in batches.

6. How reliable?

The RRP safeguard approach similarly addresses partial and gross defects on sampling points with small and large material quantities. According to the basics of material diversion, random sampling and the study of systematic bias are tools used to practice reliable safeguards without having to sample the whole facility. At the RRP all input and output accountability tanks will be sampled and analyzed: they are vital to the material verification. Otherwise, IIV samples in all material balance areas (MBA) are taken on a random basis in spite of the fact that some tanks contain relative large quantities of material. The same is valid for sampling from other strategic points (OSP) to confirm the operational status of the facility and the PIV.

The IAEA's human resources at the OSL include the dedicated safeguards inspectors that are experienced radiochemists and the SAL chemists that have extensive experience in nuclear analysis; there are also expert advisers from network laboratories such as ITU and the Los Alamos National Laboratory (LANL). Thus material verification can be performed with accuracy and precision. In doing so, a great step towards averting material diversion at the RRP is achieved.

The work in the OSL requires the full attention of the analyst who must follow the authentication procedures while performing the analysis. The working procedures have been designed in such a way that these two activities can be managed simultaneously. However, just as a mistake could occur during the analysis, the analyst can also make a human mistake in the continuity of knowledge: the complexity of the whole process makes 100% authentication a great challenge. The best achievable authentication measures have been designed and are being implemented. Improvements to such measures that are less dependent on human factors are continuously being considered for future implementation.

7. Conclusions

— The sample taking is authenticated and secured through the automatic sampling authentication system (ASAS).

— The sample receipt and storage in the OSL is authenticated through the server digital image surveillance (SDIS) system and the sealed material safe.

— The inspector-analyst can perform chemical measurements independently.

— The analytical measurement is secured with hidden sample identity, bogus samples, redundant analysis and the use of *ad hoc* surveillance.

— The analytical process requires an extra effort by the inspector-analyst to keep track of the authentication measures while doing the analysis.

— The authentication measures are the best achievable for the operation of the joint laboratory and are continuously subject to improvement.

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Improvements in the evaluation setting for the behaviour of solution transfers between vessels in a reprocessing facility using TETRA

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Abstract. Extensive experiments on solution transfer have been carried out at NMCC using the **T**est **E**quipment for solution **TRA**nsfer (TETRA) which simulates part of the main process of reprocessing plants such as cylindrical and annular type tanks, solution transfer equipment and mixing devices. **D**ata **A**nalysis and **I**nterpretation (DAI) software and the **P**rocess **I**nformation (PI) database were developed to analyze the TETRA experiments for solution transfer and stirrer operations. The solution transfer experiments were comprised of 3 types of operations; one batch transfer (batch mode), a series of batch transfers (batch receipt and batch shipment mode), and continuous transfer (batch receipt and continuous shipment or vice versa). It was confirmed that the DAI software detected the start and stop time of solution transfer for all types of operations. Detection of a transfer event requires that DAI have the appropriate parameter for each transfer operation. After detection, the liquid volume difference of receipts and shipments between tanks was calculated under a continuous mode operation that combined batch transfer and continuous transfer. In this paper, we will present DAI results and the analyses of the solution transfer experiments.

1. Introduction

Simultaneous application of continuous solution monitoring (SM) and near real time accountancy (NRTA) is strengthens the safeguard approach for large, flow-dominated reprocessing plants. The continuously monitoring of liquid level, density and temperature are essential elements in SM to identify and evaluate a series of solution transfers between the various types of liquid tanks.
Detailed calculations of the difference between shipper/receiver volumes are very important for accurate SM. The linkage setting between the shipper vessel and the receiver vessel must be identified for each solution transfer operation. The volume shipper/receiver difference (VSRD) have been calculated (simulated) and discussed [1-3]. The log data from solution transfer experiments by using the TETRA test facility was evaluated. The start/stop hours of transfer operations were successfully recognized. This paper provides a description of the experiments, the results and the conclusions.

2. Experiments

Extensive experiments of solution transfers have been carried out at NMCC using the test equipment for solution transfer (TETRA) which simulates a main process of a reprocessing plant with vessels of various types, solution transfer equipment and mixing devices. The Process Information Universal Database System (PI-UDS) and Data Analysis and Interpretation (DAI) software for solution monitoring were introduced to analyze the TETRA experiment results for solution transfers. The PI-UDS records the logging data at intervals of 1 second. For logging, the three dip tubes for level and density measurements and the temperature of all vessels were selected. And DAI accesses the records of the PI-UDS.

TETRA has 4 main vessels V11, V12, V13 and V14 as shown in FIG. 1. Vessels V11 and V14 are cylindrical in shape and vessels V12 and V13 are annular. Each vessel is linked with pumped transfer lines (from V11 to V12, V12 to V13, V13 to V14 and V14 to V11). Transfers can be done via siphon (form V11 to V12), airlift or gedeon equipment (from V12 to V13).

The three types of solution transfer operations in the TETRA were carried out:

Case1 a batch solution transfer operation (in this case, the pump transfer is basic.);
Case2 a B/B solution transfer operation; and
Case3 a B/B + C/B solution transfer operation.

Here, the B/B mode is batch shipment and batch receipt mode and the C/B or B/C modes are continuous receipt and batch shipment or vice versa. In case 1, the pump transfer (from V11 to V12) was carried out in the B/B mode. In case 2, the siphon transfer (from V11 to V12), the gedeon transfer (from V12 to V13), the pump transfer (from V13 to V14) and the pump transfer (from V14 to V11) were carried out in the B/B mode. A hold period of about 10 minutes was maintained between each transfer.

In case 3, the siphon transfer (from V11 to V12) was done first, then the gedeon transfer (from V12 to V13) was started. The pump transfer (from V13 to V14 and from V14 to V11) was started or stopped according to the level setting of the pump (ON/OFF) at V13 and V14. Thus, V11 shipped to V12 by batch siphon transfer and received from V14 by batch pump transfer (B/B mode). V12 shipped to V13 by continuous gedeon transfer and received from V11 by batch siphon transfer (C/B mode). V13 shipped to V14 by batch pump transfer and received from V12 by continuous gedeon transfer (B/C mode). V14 shipped to V11 by batch pump transfer and received from V13 by batch pump transfer (B/B mode).

The behaviours of transfer operations were logged to the PI-UDS. The logging data was treated by the solution monitoring software DAI. The DAI software can detect operation hours by checking with predefined parameters. The main observable is the behaviour of the level pressure by the three dip tubes. The DAI software makes a comparison between a predefined slope range and the mean slope of the time window. When the mean slope of the time window is within the predefined range, DAI flags it and displays it with the observed behaviour. The solution volume in the vessel was calculated based on the recorded hours of operation. Using a presetting which was the relationship between the sending vessel and the receiving vessel, the shipper/receiver difference volume was calculated.

3. Results and discussion

FIG.2 shows the observed level for a pumped transfer from V11 to V12 (case 1). This logging data was extracted from PI-UDS's records. The gradual step behaviour means shipping from or receipts to the vessel. In this figure, the batch pump transfer from V11 to V12 was done 3 times and the batch pump transfer from V12 to V13 was done 2 times.

The mean slope of the observed levels accompanying a solution transfer was calculated from the level difference and the time of the transfer. Table 1 shows the mean slope for the TETRA's transfer operation. These mean slope values were calculated from the logging data of the batch transfer pretest. These values were used for one of the DAI parameters.

FIG.3 shows the transfer operation program (tank cycle) on DAI software. In this figure, these are the marked up behaviour for the transfer operation of V12. Each square mark denotes one event. This tank cycle consists of five events. The “filling” event is the condition of level increasing; the “end of filling” event is the condition of level stopping after increasing. The main parameter, the mean slope of emptying is -8.68[kPa/h] and the mean slope of filling is +26.87[kPa/h].

Table 1 The mean slope value for solution transfer by TETRA

Equipment	Route From	To	Mean slope [kPa/h] at ship vessel	Mean slope [kPa/h] at receive vessel
Pump	V11	V12	-8.68	+26.87
	V12	V13	-29.56	+27.09
	V13	V14	-35.30	+11.95
	V14	V11	-22.63	+22.27
Siphon	V11	V12	-4.83	+15.16
Airlift	V12	V13	-9.85	+9.25
Gedeon	V12	V13	-1.78	+1.63

The recognized results of DAI are shown as FIG.4. The circle on the plot means the points recognized in the evaluation. The DAI obtains the vessel volume records on recognized hours by accessing the PI-UDS.

It was ensured that level changes accompanied transfer into both shipping and receiving vessels. However, according to details, there is latency time of about 5 ~ 120 seconds in these transfer operations. It is thought that the latency at the start of a transfer would be a factor in underestimating the receiving payment difference. The latency at the time the transfer is stopped has less influence on the transfer difference if the level is constant. It is important to identify the start hours at the point near the real operation hours.

FIG.5 shows the tank cycle for V12 filling behaviour. FIG.5(a) is the same tank cycle as that shown in FIG.3. The filling behaviour is associated with a single event. FIG.5(b) is the tank cycle improved from that depicted in FIG.5(a). The filling behaviour is associated with multiple events (in this case two events). In FIG.5(b), pump filling event (PF01) and (PF02) use the same value for the mean slope. ”PF01” event has an expanded area of the upper/lower limit on the amount setting.

FIG.6 shows the detected results by the DAI in case 1 with the tank cycle for V12 observed behaviour. In FIG.6(a), the detected points of “PF01” lag the start of shipping. But the detected points of another “PF01” depicted in FIG.6(b) are nearly equal to the start of shipping. FIG.7 depicts the latency time from the operation hours to the recorded hours. In this figure, A0, A1, A2, B0, B1 and B2 are defined as follows,

A0: the real hours of the transfer start, B0: the real hours of the transfer stop,

A1: the recorded hours of the shipping start, B1: the recorded hours of the shipping stop,

A2: the recorded time of the receipt start, B2: the recorded time of the receipt stop.

The negative latency time means that the recorded hours is before the operation began. By using the multiple event setting, the latency time “A1-A0” and “A2-A0” was negative.

Using these recorded hours, the shipper/receiver difference was calculated under the following three conditions: (a) based on operation hours, (b) based on recorded hours with a single event; and (c) based on recorded hours with multiple events.

In case (a), the maximum deviation in volume was +6 litters.

In case (b), the deviation in volume was -92 litters to +28 litters.

In case (c), the deviation in volume was -3.5 litters to +10 litters.
The range of case (c) was the ninth of the range of case (b), and equivalent to the calculated deviation in volume of case (a). Thus, detection with the multiple event setting is useful for the calculation of transfer deviation.

FIG.8 shows the level behaviour for the case 2 transfer. After the first batch transfer by siphon from V11 to V12, the second batch transfer was carried out by gedeon from V12 to V13. With the starting siphon transfer, the level of V12 increased suddenly. The level of V12 changed from increasing to holding with the stopping of the transfer. These periods correspond to the priming and un-priming periods, respectively. The priming period follows the start of operations and the un-priming period follows the stop of operations.
For the calculation of the transfer deviation, it is necessary that the recorded hours for shipping start is before the operation, and that the recorded hours for shipping stop is after level stabilization. FIG.9 shows the tank cycle setting, which takes into account these requirements. The multiple event setting is used for recording/evaluation of all events of solution transfer among the 4 vessels. FIG.10 shows the detected results of the DAI with multiple events setting as depicted in FIG.9. The circle on the plot means the recognized points. In FIG.10, all recognized points are outputted at almost the same hours as that of the operation.
FIG.11 shows the volume shipper/receiver differences (VSRD) using the multiple events setting. Table 2 shows the difference between the operation's volume and the calculated one. In FIG.11, the volume at the time of the gedeon transfer from V12 to V13 (2G3), which is the translation form V12 to V13 by gedeon, was a large negative value. These negative values are the result of un-priming. The solution returns to the shipping vessel from the sending pipe during the un-priming period. The un-priming begins after the stop of the transfer. The actual shipping volume is less than the shipping volume calculated at the time operations stopped. But, the recorded hours of the end of shipping corresponds with the event setting with the mean slope is calculated for a zero intercept. So, the recorded points are outputted in the period following un-priming. Then the shipping volume calculated at the recorded hours is nearly equal to the actual shipping volume.

Table 2 Difference volumes between the operation hours' volume and the recognized one

Item	Difference volume [L]*
Pump transfer from V11 to V12 (1P2)	-2.31
Pump transfer from V12 to V13 (2P3)	-0.53
Pump transfer from V13 to V14 (3P4)	+4.31
Pump transfer from V14 to V11 (4P1)	-3.33
Siphon transfer from V11 to V12 (1S2)	-2.00
Airlift transfer from V12 to V13 (2A3)	+1.53
Gedeon transfer from V12 to V13 (2G3)	+10.00

* Difference volume = Volume on the recognized hours – Volume on the operation hours.

FIG.12 shows the level behaviour for case 3 transfers. After the first batch transfer by siphon from V11 to V12 was complete (B/B mode), the second transfer by gedeon from V12 to V13 was started (C/B mode). The pump transfer from V13 to V14 and the pump transfer from V14 to V11 were started or stopped in keeping the gedeon transfer from V12 to V13. In FIG.12, the pump transfer from V13 to V14 was done 6 times and the pump transfer from V14 to V11 was done 3 times. For V11 and V14, there was a tank level holding period. There was no tank

level holding period for V12 and V13, the tank's levels changed continuously.
FIG.13 shows the tank cycle settings for case 3 transfers. The tank levels change continuously for V12 and V13. The link setting between filling events and emptying events was added to the tank cycle setting. This linkage prevents the change time from increasing and/or decreasing. The transfer operations in case 3 are a series of transfers, pump transfer from V13 to V14, from V14 to V11 and siphon transfer from V11 to V12. There are not hold periods. This condition is the difference from the transfer operation in case 2 (B/B mode).
FIG.14 shows the recorded results of DAI for the tank cycle associated with transfer operations in case 3. The recorded hours corresponded to the actual operational hours. The difference in time between actual operational hours and those recorded were less than a few seconds.
In case 3, operations were in the B/B and C/B or B/C modes. The recognition with the multiple event setting is useful for detecting the transfer period. The linkage between the start of a shipping event and a receiving event (and the start of a receiving event and a shipping event) are directly useful in detecting a series of transfer operations. It seems that the volume shipper/receiver difference can be calculated using the volume from the recognized points under the multiple event setting and the added linkage for the tank cycle setting.

4. Conclusion

The solution transfer experiments were carried out using a single batch mode, the B/B and C/B modes with the B/B mode. The solution monitoring software DAI was used for the identification of transfer operations.
DAI could identify the transfer period for both the B/B mode and the C/B mode by using the mean slope of B/B and C/B transfers. With the multiple event settings for each transfer, the lag time for recognition based on the operational hours was shorter than that with the single event setting. Therefore, the recognition with the multi event setting is useful for the calculation of transfer deviations. These multiple event settings are useful for adjusting the solution monitoring at large reprocessing plants.

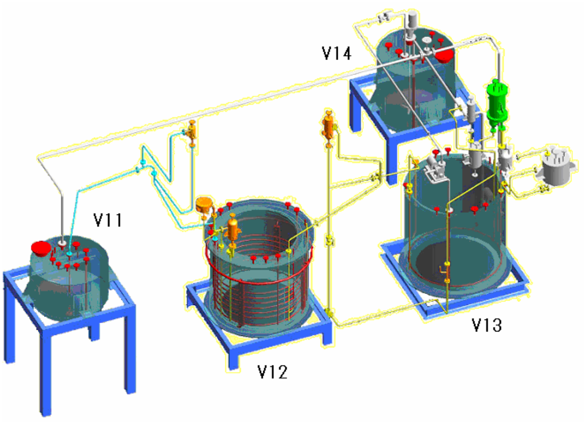

FIG. 1 Test Equipment for solution TRAnsfer (TETRA).
Cylindrical shape (V11 and V14),
Annular shape (V12 and V13).

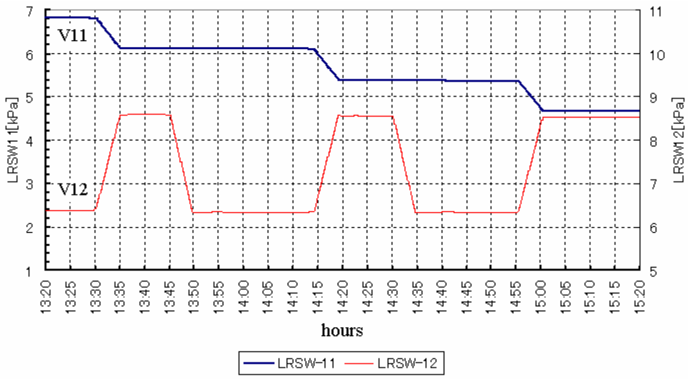

FIG. 2 Level behaviour of pump transfer in case 1.
3 transfers from V11 to V12 (batch mode),
2 transfers from V12 to V13 (batch mode).

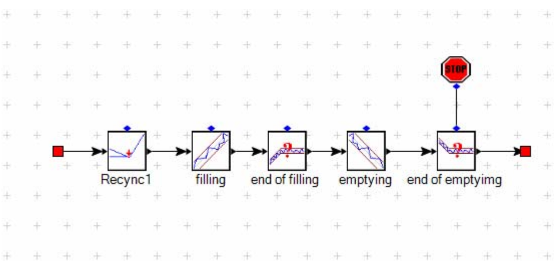

FIG. 3 Transfer operation program (Tank Cycle).
"filling" event encloses the level increasing,
"end of filling" event encloses the level stopping.

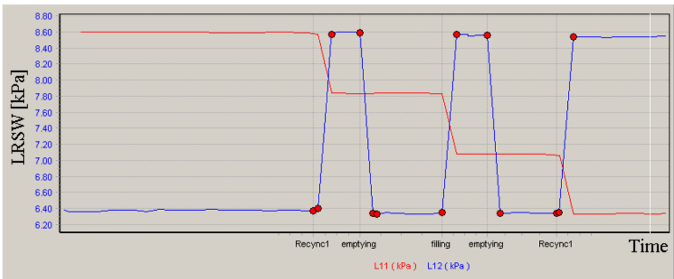

FIG. 4 Recognized results of DAI.
blue line describes the level behaviour of V12,
red circles describe the recognized points.

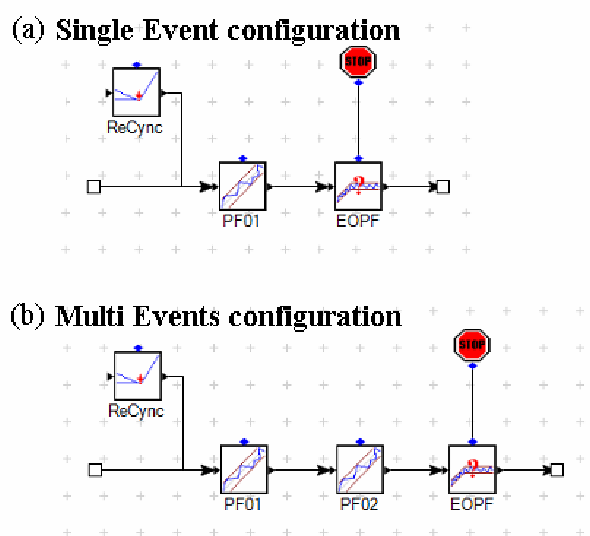

FIG. 5 Tank Cycle for V12 behaviour.
(a) Single filling event(PF01) setting,
(b) Multi filling event (PF01 and PF02) setting.

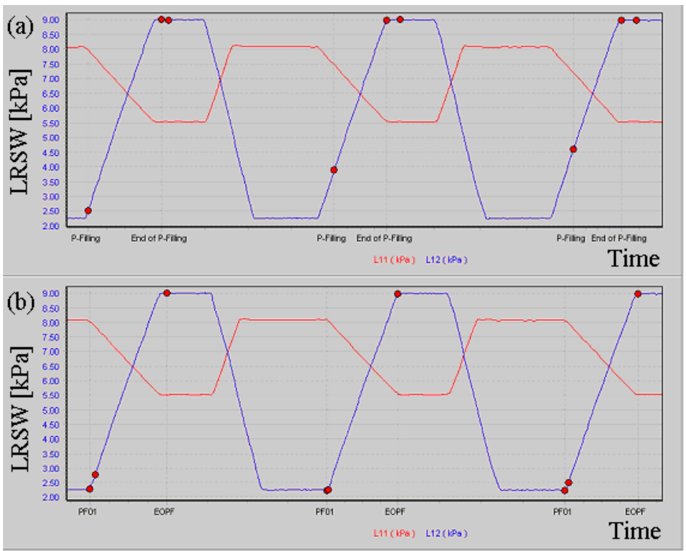

FIG. 6 Recognized results of DAI in case 1.
(a) Single filling event setting,
(b) Multi filling event setting.

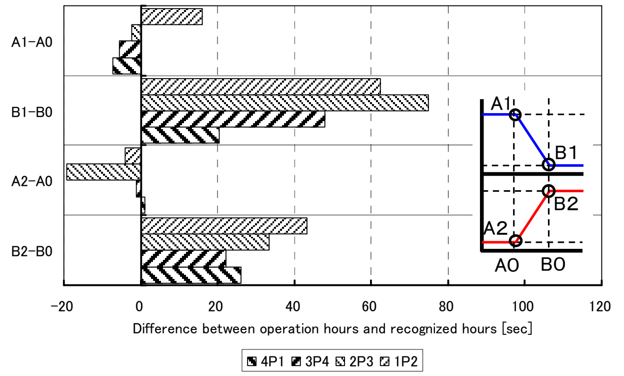

FIG. 7 Difference between operated and recognized hours.

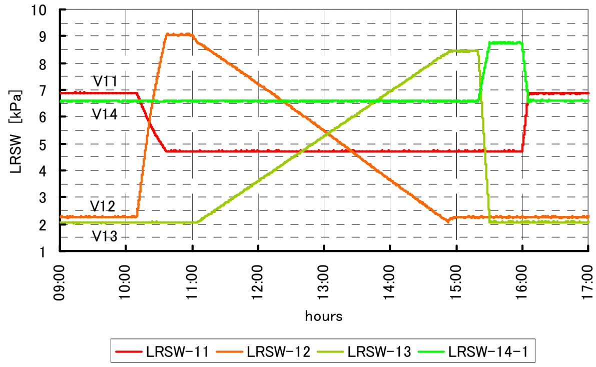

FIG. 8 Level behaviour of case 2 transfer. The operation is series of siphon, gedeon and pump transfer.

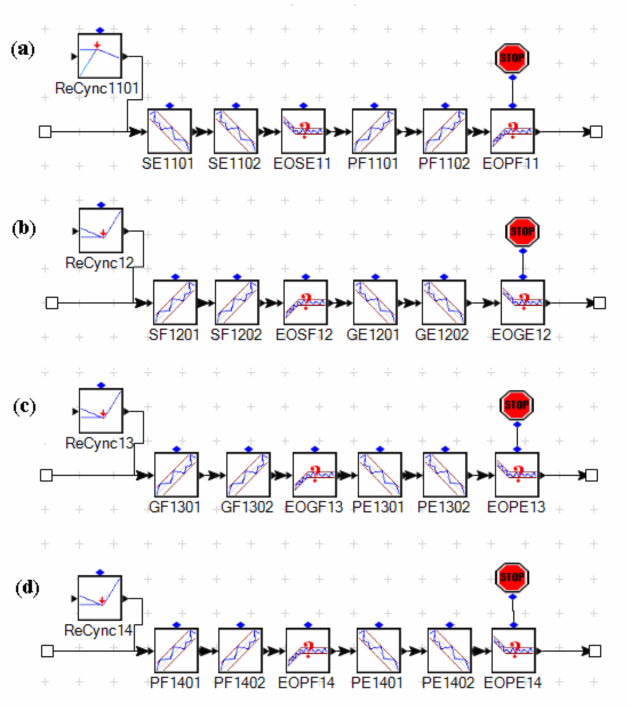

FIG.9 Tank Cycles for transfer in case 2.
(a) V11 contains siphon emptying and pump filling.
(b) V12 contains siphon filling and gedeon emptying.
(c) V13 contains gedeon filling and pump emptying.
(d) V14 contains pump filling and pump emptying.

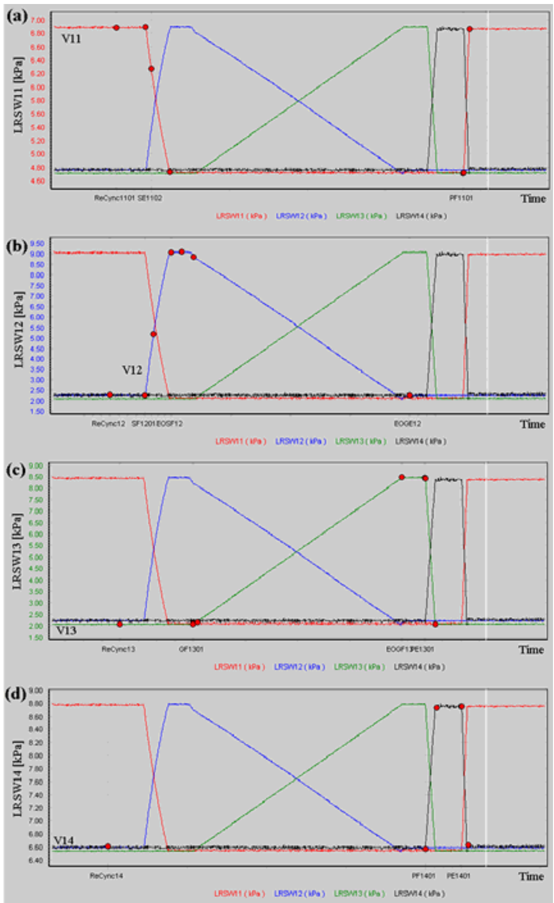

FIG.10 Recognized results of DAI in case 2. Each line describes the behaviour. The circles describe the recognized points.

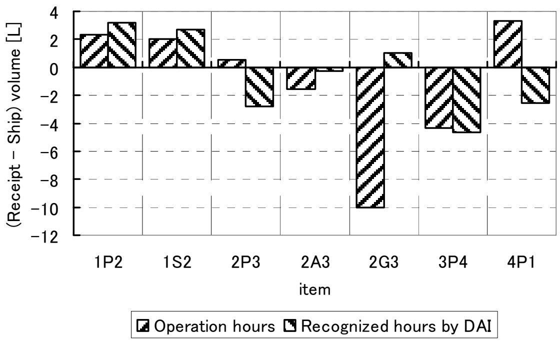

FIG.11 VSRD on the multi event setting.

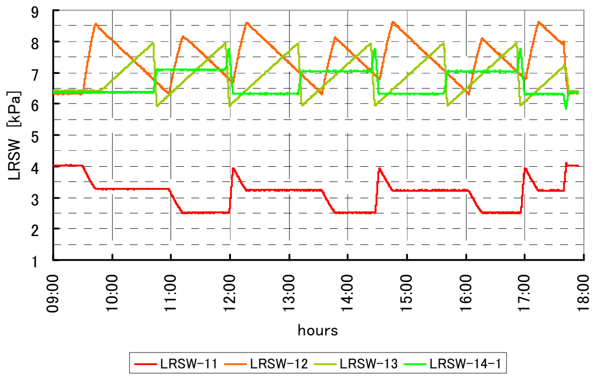

FIG.12 Level behaviour of case 3 transfer.

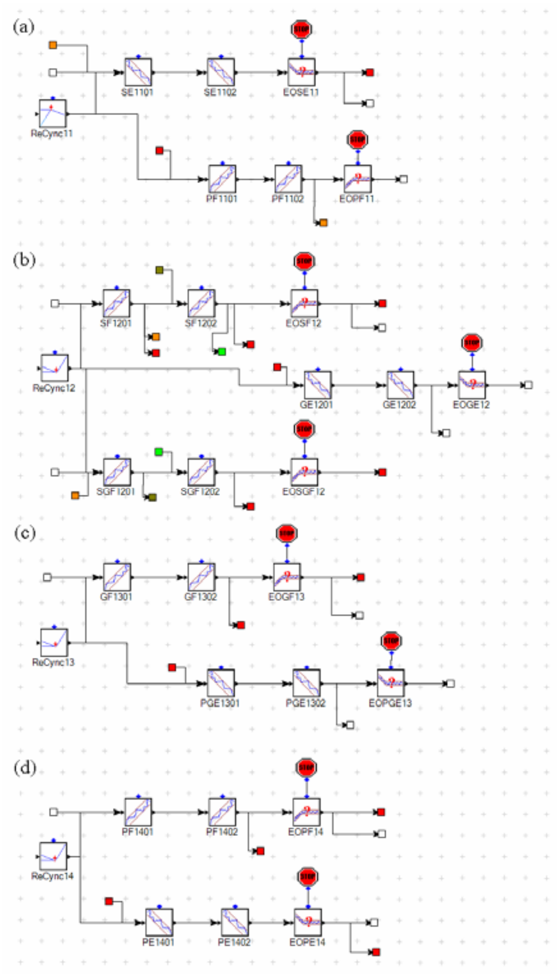

FIG.13 Tank Cycles for transfer in case 3.
(a) V11 contains siphon emptying and pump filling.
(b) V12 contains gedeon emptying and siphon filling or siphon transfer with gedeon transfer.
(c) V13 contains gedeon filling and pump emptying with gedeon filling.
(d) V14 contains pump filling and pump emptying.

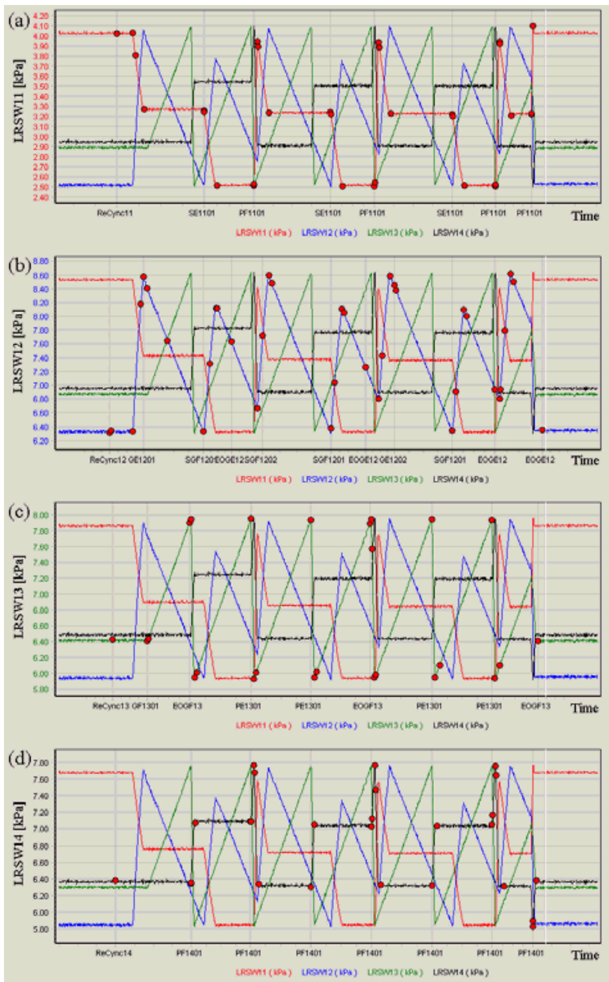

FIG.14 Recognized results of DAI in case 3.
Each line describes the behaviour.
The circles describe the recognized points.

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Reduction of fluctuation and small bias observed in continuous volume monitoring taken in an annular tank for plutonium nitrate

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Abstract. Accurate measurement and continuous monitoring of density, level and volume by dip-tubes in an annular tank for plutonium nitrate are indispensable for material accountancy and safeguards in a reprocessing and conversion plant. The principle of this method is to measure hydrostatic pressure at the tip of the tube, which postulates that solution is resting (motionless) during measurement. However, solution is being agitated for safety reasons except for the short time when the measurement is done, thus solution circulates and causes fluctuation and possibly some small bias on hydrostatic pressure measurement. The effect of agitation is relatively large at the tip of dip-tube far from the bottom of tank. Therefore, the effect does not cancel between two tubes to measure density, and density has a small bias and fluctuation. The change in density causes apparent change in volume to the opposite direction. We focused attention to the solution mass calculated as a product of volume and density measured coincidentally, and found that the small bias and fluctuation is extremely reduced in mass monitoring. This technique enables stable and accurate verification of monitoring data and contributes to smooth inspection activities.

1. Introduction

To establish nuclear fuel cycle for peaceful use in JAPAN, it is indispensable to carry out material accountancy very carefully and strictly, to cooperate with IAEA safeguards, and to carry out technical developments which contribute to non-proliferation of nuclear weapons and to improve compatibility of smooth plant operation and smooth inspection.

In the area of fuel reprocessing and plutonium conversion, over six tons of plutonium has been converted under IAEA inspection since 1983 applying microwave direct denitration method to co-conversion process where no PuO_2 powder exists. Plutonium nitrate solution extracted from spent fuel at the Tokai Reprocessing Plant (TRP) in the Japan Atomic Energy Agency (JAEA) at Tokai-village is transferred to the Plutonium Conversion Development Facility (PCDF) constructed adjacent to TRP, then mixed with uranyl nitrate solution in a ratio of Pu/U=1. The mixed solution is converted directly to mixed-oxide powder using microwave heating, thus there is no PuO_2 in the co-conversion process [1]. The plutonium throughput of PCDF reached 0.6 ton/y (actual base).

There are two annular tanks storing plutonium nitrate and another two annular tanks storing plutonium-uranyl mixed nitrate in PCDF. The first tank is an accountability input tank and annual recalibration has been carried out in the presence of inspector. The plutonium inventory of these tanks is about three quarters of the total plutonium inventory in the process area of PCDF, if these tanks are filled to nominal volume. The nominal volume is 300 liter. Accurate measurement and monitoring of these four tanks are fundamental technique both for operator and for inspector.

2. Volume and Pu mass measurement of plutonium nitrate

The principle of measurement method and procedure to calculate solution density, level, volume and plutonium mass is shown in the FIG. 1. Pressurized air is purged continuously to the dip-tubes and bubbles are released periodically about once every several seconds from the tip of the tubes. The pressure measured by ELTM is the back pressure which corresponds to the hydrostatic pressure. ΔP_D is the differential pressure between the Major and the Minor, and it is linear to the product of gap length Lg (vertical differential length between the Major and the Minor) and solution density ρ_{tank} at solution temperature in a tank. The density is normally 1.4-1.5 g/cm^3 for plutonium nitrate. ΔP_L is the differential pressure between the Major and the Reference, and it is linear to the product of solution density ρ_{tank} and solution level H. The nominal level is 1200 mm. The gap length was determined at initial tank calibration and has annually been confirmed to be constant at recalibration with known density, solution normally ion-exchange filtrated water or diluted nitric acid. Therefore, ρ_{tank} is calculated from ΔP_D, then H is calculated from ΔP_L and ρ_{tank}, then H is converted to volume V using calibration equation F determined at the initial tank calibration with known density solution. The gap length is 200 mm, and ΔP_D is about three kilopascals and ΔP_L is over ten kilopascals for plutonium nitrate. The range of random variation of ΔP_D and ΔP_L is several pascals, so the uncertainty of ΔP_D is several times larger than that of ΔP_L, and occupies over half of the total uncertainty. Therefore, it should be mentioned that measurement of ΔP_D is the key of accurate measurement system. In the practical calculation procedure, corrections such as level difference between the tip of tube and the manometer, bubble effect at the tip of tube, and tank thermal expansion have to be considered.

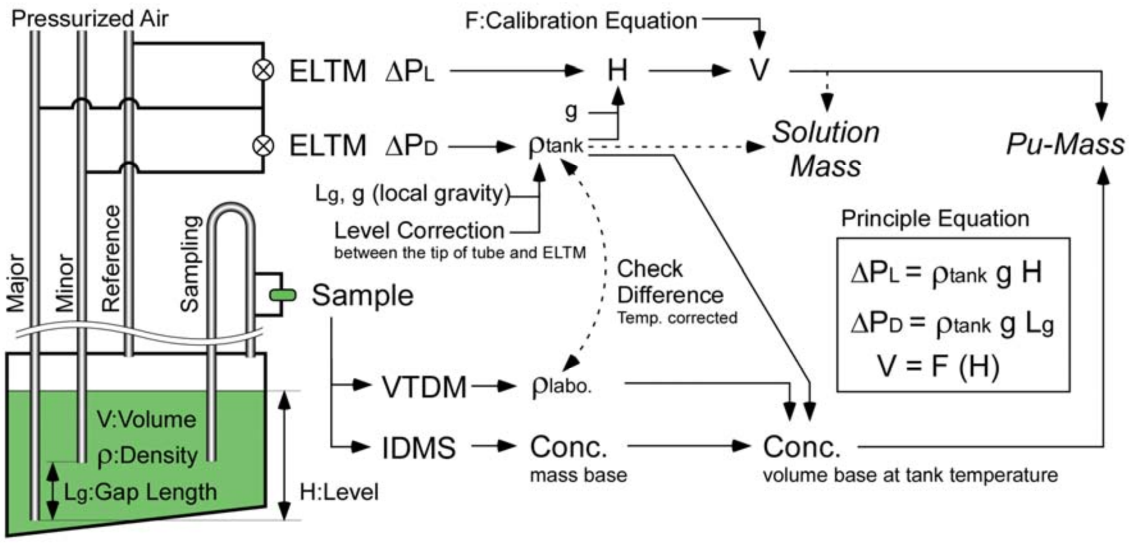

FIG. 1. Procedure to determine solution volume and Pu mass in a tank.

We have developed since the early of 1980s very unique electro-manometer system (ELTM) using digital-quartz pressure transducer with temperature control and automated zero-point control within two pascals [2] to measure accurately the density, level and volume. As a result of demonstration, repeatability of annual recalibration has been successful and we confirmed that coefficients of calibration equation were very stable [3]. In addition, solution density of tank measured by ELTM and solution density of sample measured by vibration tube density meter (VTDM) has been compared, and we confirmed that these two results consisted within 0.1% (1σ) when temperature difference between tank and sample was corrected and level difference between tip of dip-tube and manometer was also corrected [4]. The estimated total uncertainty of ELTM satisfies well the international target value (ITV). This ELTM has been used for determination of plutonium input from TRP, interim inventory verification (IIV) and physical inventory taking/verification (PIT/PIV) both by operator and by inspector. The standard pressure (dead-weight tester) to calibrate the transducer has been verified independently by inspector.

In the late of 1990s, the IAEA installed independent continuous tank monitoring system for these four tanks. For operator, it is expected that continuous accurate monitoring is effective to find unexpected event such as dip-tube clogging, so we improved ELTM to collect data every minute for each tank to enable continuous monitoring. As a result, we found the essential difference between the conventional discrete measurement and the continuous monitoring.

3. Difference between discrete measurement and continuous monitoring

As mentioned above, the density, level and volume measurement is based on the true hydrostatic pressure measurement at the tip of dip-tube, which postulates that solution in a tank is resting (motionless) during measurement. This requirement is satisfied at conventional discrete measurement for material accountancy, because the measurement time is short and the agitation by air is temporarily stopped during measurement.

However, it is very difficult to stop agitation except for the short time of measurement, because the solution have to be circulated for the safety reason to diffuse and remove hydrogen gas generated in the solution by alpha particle. As a result, the solution flows and causes some fluctuation and possibly some small bias on hydrostatic pressure measurement. Conventional discrete measurement when agitation is stopped is free from such effect.

Typical effect of solution flow on ΔP_D is shown in the FIG. 2a and FIG. 2b. FIG. 2a is the example of failed tank calibration where ΔP_D changes dependently on solution level due to agitation, and FIG. 2b is the example of successful tank calibration where ΔP_D is independent from the level because there is no agitation and thus hydrostatic pressure is measured correctly. The effect appears when solution level is over half of the nominal level (1200 mm), and reaches 5 Pa which is 0.25 % of ΔP_D (2 kPa at tank calibration). At the measurement of plutonium nitrate with larger density, this bias will increase to about 6-7 Pa which is 0.2 % of ΔP_D (about 3 kPa). This bias is not sensitive to the existence or nonexistence based monitoring, but sensitive to quantitative monitoring of large plutonium inventory.

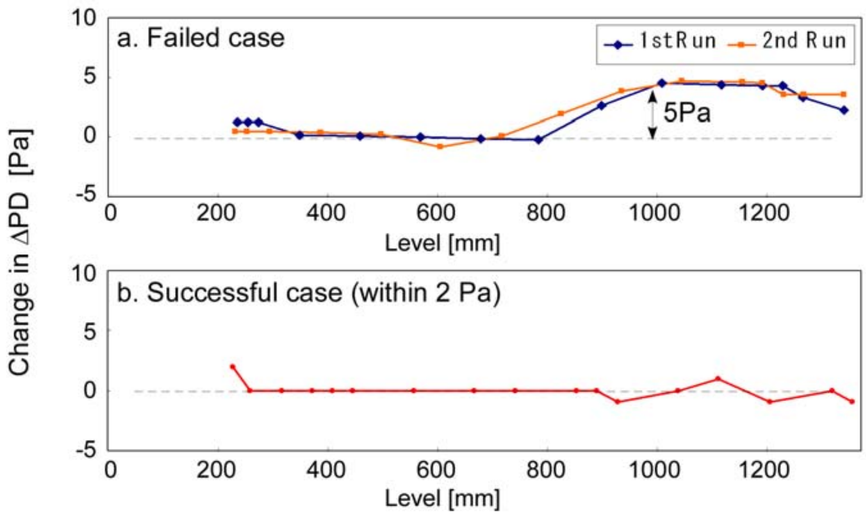

FIG. 2. Example of change in ΔP_D during tank calibration.

Example of continuous monitoring data taken by ELTM is shown in the FIG. 3. During this period (four days), the amount of solution in a tank did not change except for evaporation around 0.1 liter per day and temporally move to the circulation line for sampling at IIV on the third day.

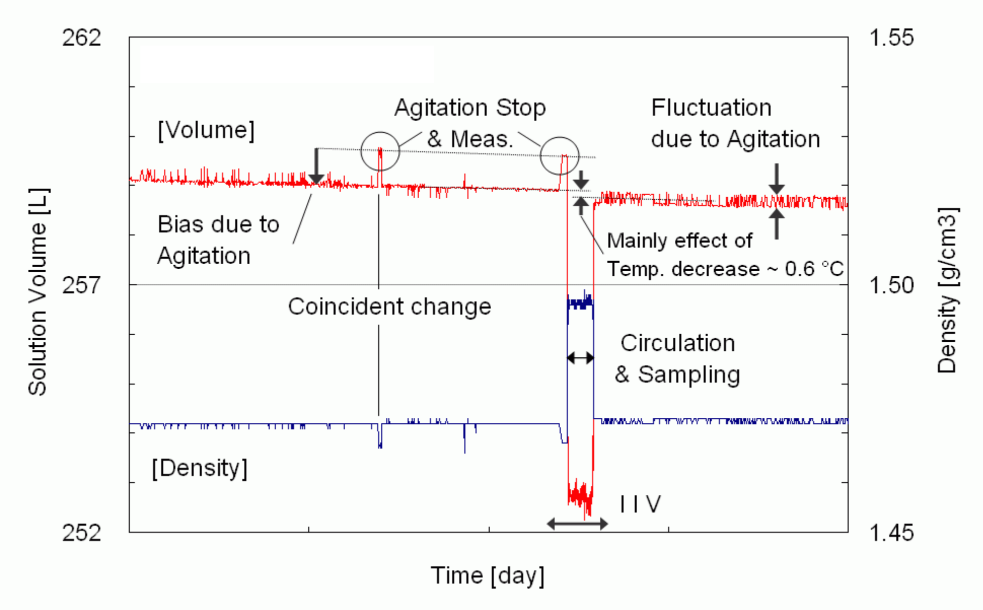

FIG. 3. Example of continuous volume monitoring data.

Here, we can observe changes related to agitation :
a) The day before IIV, density decreased and volume increased coincidentally and inversely around 0.2 % when operator carried out conventional discrete measurement as a preparation for IIV.
b) Just before IIV, density and volume changed coincidentally and inversely again when operator carried out conventional discrete measurement at IIV.
c) Density and volume fluctuated through this period and fluctuation width is around 0.1 %.

We can observe another change in density and volume caused by temperature decrease about 0.6 ºC after circulation & sampling. It should be emphasized that measured density, level and volume mentioned at a) and b) above are correct because agitation is stopped during measurement and almost of all continuous monitoring data have small bias and fluctuation due to agitation.

4. Estimated reason of the small bias and fluctuation

The piping used for agitation by air is installed along with the inclined bottom plane of the tank, thus air bubbles rise up not from entire circumference but from the shallowest point shown in FIG. 4a and 4b which is the projected plane of the annular tank. The tip of the major tube is set very close to the deepest point most far from the rising point of air. The tip of the minor tube is set about 200 mm higher from the bottom where solution flow due to agitation is faster than the flow at the deepest point. It is estimated from these figures that :
a) The down flow reaches to the tip of minor tube and effects on ΔP_D when level is high (FIG. 4a);
b) The down flow does not reach to the tip of minor tube, so ΔP_D is correct when level is low (FIG.4b);

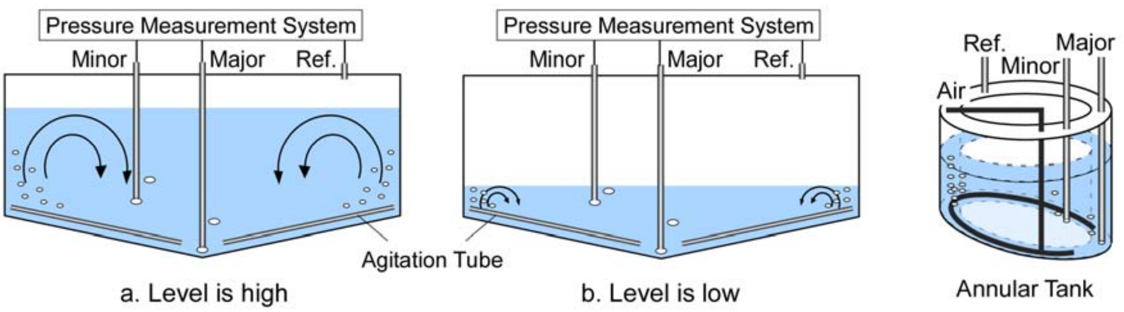

FIG. 4. Estimated solution flow in a tank.

It is difficult to distinguish clearly the stage a) from stage b) because the change is progressive and depends on many parameters such as thickness of solution in the annular tank, flow resisting piping or supports, physical solution properties and so on. However, it is possible to explain qualitatively the experience shown in FIG. 2a.

The reason why ΔP_D becomes large when down flow reaches the tip of minor tube is explained in the FIG. 5. FIG. 5a shows the bubbles at the tip of major and minor dip tubes without solution flow, and FIG. 5b shows the bubbles with solution flow. It is important that the bubble height and shape is theoretically and experimentally independent from solution depth [5]. So, the difference between the backpressure and the hydrostatic pressure at the tip, called overpressure, is exactly same at the tip of major tube and at the tip of minor tube, regardless of asynchronous timing. Therefore, the overpressure cancels each other and measured average of ΔP_D is exactly same as the differential hydrostatic pressure if there is no solution flow. For reference, the height of bubble just before release is 5.5 mm and average of overpressure is 46 Pa in water when outer diameter of the tube is 15 mm. The average of overpressure estimated in plutonium nitrate at the same condition is 54 Pa [6].

However, this balance is lost when solution flows around the tip of minor tube that is set far from the major tube shown in the FIG. 5b. The solution flow yields a certain type of bubble vibration and the bubble is released before the overpressure reaches maximum. Therefore, the average of overpressure at minor tube decreases, and thus measured ΔP_D increases. The change in overpressure is 5 Pa in water or diluted nitric acid shown in the FIG. 2a, and estimated to be 6-7 Pa in plutonium nitrate because the density is 1.5 times larger and the bubble becomes more flat. When the change in overpressure is assumed to be 7 Pa and ΔP_D is 3 kPa, change in density of plutonium nitrate is 7/3000 = 0.23 % or 1.47 g/cm^3 × 7/3000 = 0.004 g/cm^3, which corresponds to the observed change shown in the FIG. 3. On the other hand, observed change in volume due to agitation is about 0.6 L which is 0.23 % of 260 L, that corresponds to the change in density. When agitation is very strong during circulation & sampling, observed change in density is 0.025 g/cm^3 which is 1.7 % of 1.47 g/cm^3, and the change in overpressure is estimated to be 3000 Pa × 1.7 % = 50 Pa which is close to but smaller than the average overpressure of plutonium nitrate. Therefore, the reason of small bias and fluctuation is considered to be the change in overpressure at the tip of minor dip-tube.

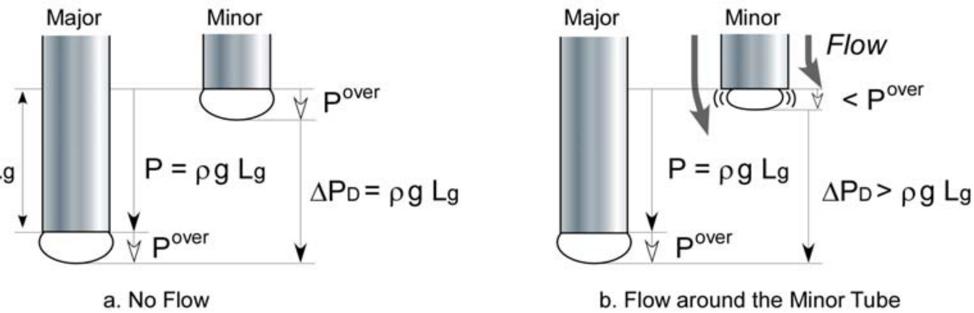

FIG. 5. Effect of solution flow on measurement of ΔP_D.

In addition, the observed change in volume after circulation & sampling is 0.1 L which corresponds to the calculation 0.6 ºC × 0.07 % × 260 L = 0.1 L where 0.07 % is the typical temperature coefficient of density for plutonium nitrate [4].

5. Mass monitoring for continuous monitoring

As mentioned above, solution flow during continuous measurement effects on ΔP_D measurement and increases ρ, decreases H and decreases V according to the principle equations. Here, we consider a small variation of density δρ and related equations.

$$\rho = \rho_0 + \delta\rho \quad (1)$$

$$H = \Delta P_L / (\rho \times g) \quad (2)$$

$$V = a_0 + a_1 \times H \tag{3}$$
$$M = V \times \rho \tag{4}$$

where, ρ_0 is the density without flow, $\delta\rho$ is the variation of ρ due to flow, g is the local gravitational acceleration, a_0 and a_1 are coefficients of the function V=F(H), and M is solution mass we introduced here. It is supposed that contribution of the 2nd term or more of the function F on evaluating error propagation is small and negligible. When $\delta\rho / \rho_0$ is sufficiently small, $\delta H / H_0$, $\delta V / V_0$ and $\delta M / M_0$ are approximated as :

$$1 / \rho = 1 / (\rho_0 + \delta\rho) = (1 / \rho_0) / (1 + \delta\rho / \rho_0) = (1 / \rho_0) \times (1 - \delta\rho / \rho_0) \tag{5}$$
$$H = H_0 + \delta H = [\Delta P_L / (\rho_0 \times g)] - [(\delta\rho / \rho_0) \times \Delta P_L / (\rho_0 \times g)] \tag{6}$$
$$\delta H / H_0 = -(\delta\rho / \rho_0) \tag{7}$$
$$V = V_0 + \delta V = [a_0 + a_1 \times \Delta P_L / (\rho_0 \times g)] - [a_1 \times (\delta\rho / \rho_0) \times \Delta P_L / (\rho_0 \times g)] \tag{8}$$
$$\begin{aligned} \delta V / V_0 &= [-a_1 \times (\delta\rho / \rho_0) \times \Delta P_L / (\rho_0 \times g)] / [a_0 + a_1 \times \Delta P_L / (\rho_0 \times g)] \\ &= -(\delta\rho / \rho_0) / [1 + a_0 / [a_1 \times \Delta P_L / (\rho_0 \times g)]] \\ &= -(\delta\rho / \rho_0) \times (1 - a_0 / V_0) \quad \text{when} \quad a_0 << V_0 \end{aligned} \tag{9}$$
$$\begin{aligned} M = M_0 + \delta M &= [V_0 + \delta V] \times [\rho_0 + \delta\rho] \\ &= [V_0 \times \rho_0] - [\rho_0 \times a_1 \times (\delta\rho / \rho_0) \times \Delta P_L / (\rho_0 \times g)] + [V_0 \times \delta\rho] \\ &\qquad - [\delta\rho \times a_1 \times (\delta\rho / \rho_0) \times \Delta P_L / (\rho_0 \times g)] \\ &= [V_0 \times \rho_0] - [\rho_0 \times a_1 \times (\delta\rho / \rho_0) \times \Delta P_L / (\rho_0 \times g)] + [a_0 + a_1 \times \Delta P_L / (\rho_0 \times g)] \times \delta\rho \\ &\qquad - [\delta\rho \times a_1 \times (\delta\rho / \rho_0) \times \Delta P_L / (\rho_0 \times g)] \\ &= [V_0 \times \rho_0] + a_0 \times \delta\rho - (\delta\rho / \rho_0)^2 \times a_1 \times \Delta P_L / (\rho_0 \times g) \\ &= [V_0 \times \rho_0] + [a_0 \times \delta\rho] \quad \text{when} \quad (\delta\rho / \rho_0)^2 \quad \text{is supposed to be zero} \end{aligned} \tag{10}$$
$$\delta M / M_0 = (\delta\rho / \rho_0) \times (a_0 / V_0) \tag{11}$$

As a result, the relation shown below is derived when $a_0 << V_0$ that is normally approved.

$$|\delta M / M_0| << |\delta V / V_0| < |\delta H / H_0| = |\delta\rho / \rho_0| \tag{12}$$

Therefore, it is expected that solution mass shows smaller bias and variation caused by agitation. The result is shown in the FIG. 6. The original data is same as the FIG. 3.

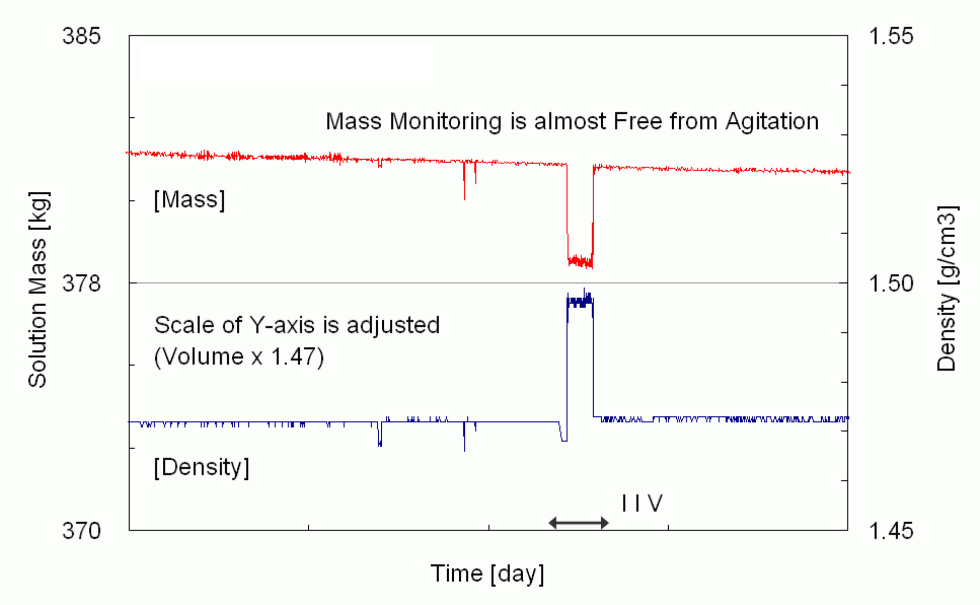

FIG. 6. Example of continuous mass monitoring data.

1) Change in volume by appearance observed in the FIG. 3 was amazingly reduced.
2) Change in volume due to temperature observed in the FIG. 3 was disappeared.
3) Mass monitoring data shows correctly the amount which is gradually decreasing due to evaporation.
4) Decrease of the amount during circulation & sampling was reduced and became to reasonable value because the change in volume by appearance was reduced.

Therefore, it is expected that mass monitoring is very helpful for inspector to compare smoothly the operator's declaration with the monitoring results. For operator, it is also expected to find unexpected event timely and precisely such as a small leak of solution during sampling. In addition, it will be possible to simplify the current procedure to determine Pu mass (shown in the FIG. 1) by the way of calculating Pu mass as a product of solution mass by ELTM and mass based concentration by IDMS, if the density measured by ELTM is compared and checked with the density measured by VTDM.

6. Conclusion

For density, level and volume measurement of plutonium nitrate in a tank, essential difference between the conventional discrete measurement without agitation by air and the continuous monitoring with agitation was observed and considered. In PCDF, continuous monitoring data showed change in density and volume by appearance which was not negligible. The estimated reason of the change was the solution flow around the tip of minor tube and resultant change in overpressure related to ΔP_D. Error propagation was evaluated and solution mass as a product of volume and density measured coincidentally was expected to be most stable. After these considerations, mass monitoring was confirmed to be stable and showing correctly the amount of solution regardless of agitation by air and temperature change.

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Non-proliferation technology development study for UREX

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Abstract. The United States recently increased its Research and Development efforts in the reprocessing of commercial spent nuclear fuels. This change is taking place in the context of the Global Nuclear Energy Partnership (GNEP) program. One research aspect under the GNEP program involves changing the reprocessing chemistry from the more common PUREX (plutonium and uranium recovery by extraction) process to the $UREX^{+1a}$ (uranium recovery by extraction) process. An initial report looking at the broad changes to safeguards technology was written by M.M. Pickrell et al. This present publication is a follow-up to Pickrell's initial work. Because changes to the chemistry will impact the safeguards systems, investigating the impact of these changes on the passive neutron albedo reactivity (PNAR) concept is the purpose of this work. Particular emphasis is given to the applicability and sensitivity of the PNAR technique. Isotopic data from the Argonne Model for Universal Solvent Extraction provided by Argonne National Laboratory personnel was used as input data. The Monte Carlo transport code MCNPX was used to perform the necessary neutron transport.

1. Introduction

Since publication of the report by M.M. Pickrell et al. [1], the UREX+ (uranium recovery by extraction+) process has changed from $UREX^{+4}$ to $UREX^{+1a}$. The present work uses the updated $UREX^{+1a}$ chemistry. The purpose of this publication is to investigate, in detail, one of the measurement situations resulting from the change in the reprocessing chemistry. Given the wide breadth of relevant research topics, an additional filter was applied in the selection process: it is desirable that the research be of general interest to international safeguards. Neutron interrogation using the passive neutron albedo reactivity (PNAR) concept is studied in this publication. The results of K-edge densitometry in the context of $UREX^{+1a}$ will soon be published as a separate work [2]. A third publication of general assistance to those interested in nondestructive assay measurements of the $UREX^{+1a}$ material streams will contain the spontaneous fission and (α,n) neutron data for 18 material streams of a $UREX^{+1a}$ facility [3]. In all of these publications, the relative abundances of the different isotopes were obtained from the Argonne Model for Universal Solvent Extraction (AMUSE) code [4].

1.1. *PNAR Concept in $UREX^{+1a}$ Context*

1.1.1. *Introduction to the PNAR Concept*

The PNAR concept is described in detail in a publication by Menlove and Beddingfield [5]. A brief description is given here. The intrinsic neutron emission [primarily the spontaneous fission of curium with some plutonium and (α,n) neutrons] from the sample is used to self-interrogate the fissile material in the sample itself. Two separate measurements of the sample are made, and the ratios of the count rates obtained are analyzed. The only difference between the two measurements is that in one case, the sample has a thin layer of cadmium surrounding it (located between the sample and the moderating walls about the product), while in the other case; cadmium is not present as all. An illustration of this approach is given in Fig. 1.

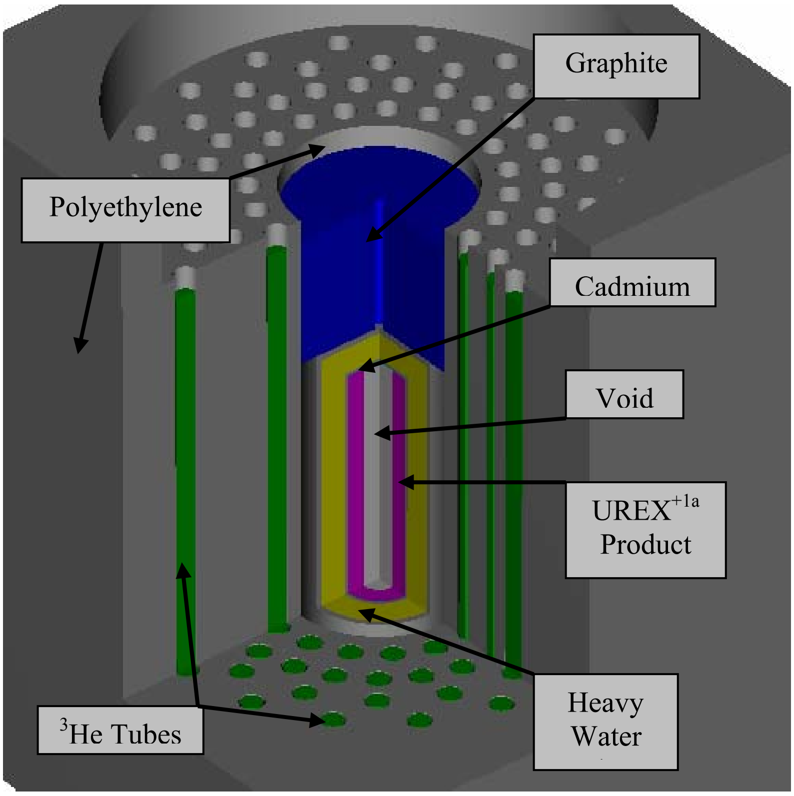

FIG. 1. The conceptual design of the PNAR diagnostic with UREX^{+1a} material depicted as an annulus.

The innermost volume in Fig. 1 is a void inside of an annulus of the sample material. The sample material for UREX^{+1a} contains neptunium, plutonium, americium, and curium. Depending on the measurement location, the assay material may be in liquid or oxide form, or the assay sample may contain fission products or uranium. The cadmium layer is outside the sample material with heavy water next. The heavy water was selected as a non-absorbent moderator to reflect a substantial fraction of the neutrons leaving the sample back into the sample to induce fission in the sample. Beyond the heavy water are polyethylene moderated detector banks containing ^{3}He tubes.

Since the cadmium layer efficiently absorbs all neutrons with energies below ~1 eV, the ratio of measurements made with and without the cadmium quantifies the effect (change in induced fissions) of low-energy neutrons interrogating the sample. Effectively, the PNAR concept is a low-energy (<1 eV) neutron-interrogation technique; as such, the thermal reactivity is measured. Since the thermal reactivity is a function of fissile mass, fissile mass can be determined.

Time-correlated neutron data is collected (doubles and triples) in additional to the singles. The time-correlated data can give a better signal relative to background, and the additional measured parameters allow analysis of multiplication and (α,n) influences. Unlike most measurement methods, the PNAR concept benefits from the high neutron source strength of a sample such as spent fuel. Both the count rates taken with and without the cadmium present are directly proportional to the source strength; as such, the ratio of these count rates is independent of source strength. In fact, the relative precision improves with source strength for singles. At very high count rates, the increase in the accidental rate is such that the relative uncertainty of the doubles is essentially independent of count rate, whereas elevated source strength causes the uncertainty in the triples data to increase significantly.

The PNAR concept is of particular use in the UREX^{+1a} aqueous reprocessing of the Global Nuclear Energy Partnership (GNEP) program, since plutonium and curium remain co-mingled at all times, and since curium is the primary source of neutrons. For this reason, the technique is of potential use for every stream containing plutonium. If PNAR is applied to cases in which uranium and plutonium are both present, then the fissile content measured is expected to have significant contributions from both elements. If isotopic resolution is needed, data from other diagnostics (destructive analysis, gamma spectroscopy, etc.) is required. If PNAR is applied to a case in which fission products are still present (flows after CCD-PEG [the Cs and Sr removal process] or TRUEX [transuranic elements extraction]), then lead shielding can be added to the detector. The fundamental approach still works. For this reason, the PNAR concept is applicable to spent fuel. Spent fuel combines the two cases just described (uranium and plutonium with fission products). The ability to measure spent fuel is of particular interest to the possible pyrochemical reprocessing path of the GNEP, since pyrochemical reprocessing does not have an accountability tank at which to determine fissile content with destructive analysis. Since uranium is removed in the first process of UREX^{+1a}, PNAR is particularly useful for quantifying plutonium, since it is the dominant fissile material remaining.

1.1.2. Benchmarking Past Experiment

In the work by Menlove and Beddingfield [5], the PNAR concept was studied experimentally and with modelling. A californium source was used as the neutron source in place of curium. Low-enriched uranium (LEU) fuel rods (3.19 wt % ^{235}U) and depleted uranium (DU) fuel rods (0.23 wt % ^{235}U) provided the fissile material. The experimental setup is pictured in Fig. 2. A polyethylene insert was designed and put inside of the plutonium scrap multiplicity counter (PSMC) [6] for the experiment. The californium source fit inside of the inner polyethylene cylinder. A thin, removable, cadmium layer surrounded the polyethylene cylinder followed by the fuel rods and then another, removable, layer of cadmium. The final layer in the insert contained more polyethylene to reflect low-energy neutrons back into the fuel (this was necessary due to the cadmium layer on the inside of the PSMC). By varying the mixture of LEU and DU fuel rods, the cadmium ratio was measured as a function of average enrichment. Since polyethylene was located both inside and outside the fissile material, two annuli of cadmium (one inside and one outside) were positioned about the fissile material.

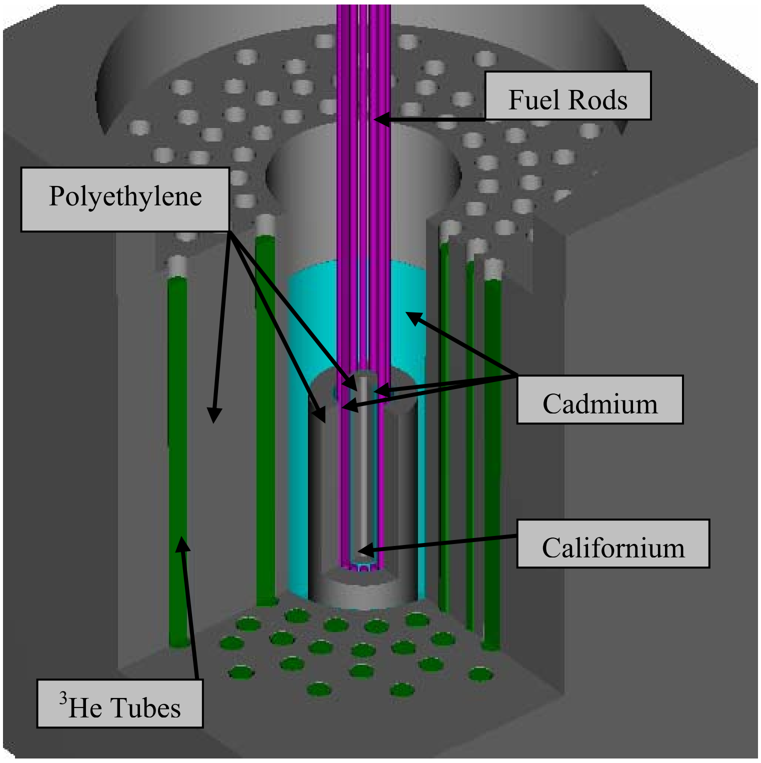

FIG. 2. The experimental setup used to benchmark the PNAR method. LEU and DU fuel rods are placed inside the "PNAR insert." Together, the insert and the fuel rods are placed inside the PSMC counter with the counter lid remaining off.

The measured data taken for the experiment described above is shown in Fig. 3. The measurement time was 1000 s, and the resulting uncertainties were 0.04% for singles, 0.16% for doubles, and 0.52% for triples; a predelay of 3 µs and a gate width of 128 µs were used. For the calculated cadmium ratios, the uncertainties were 0.06% for singles, 0.22% for doubles, and 0.65% for triples.

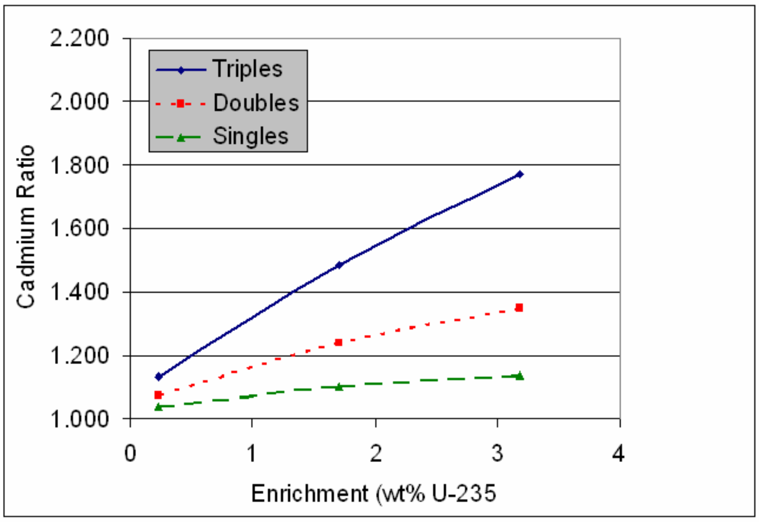

FIG. 3. The measured cadmium ratio results for the LEU fuel-rod experiment illustrated in Fig. 2.

In the publication by Menlove and Beddingfield [5], the Monte Carlo N-Particle (MCNP) transport code was used to model the results. Good agreement was obtained for the singles count rates, but a large disparity, greater than 100%, was determined for the doubles and triples count rates as compared to the experimental results. It was anticipated in their publication that this disagreement was because of the violation of point-model assumptions used to interpret the MCNP results. Recently, multiplicity capabilities were added to the Monte Carlo N-Particle eXtended (MCNPX) [7] transport code so that it is no longer necessary to use the point model. The neutron transport can be modelled with full spatial and temporal accuracy. Since the goal of this present work is to use the MCNPX capability to predict the capability of PNAR with UREX^{+1a} materials, the earlier experimental results were benchmarked first to illustrate that the new modelling capabilities in MCNPX can accurately predict the experimental results of a PNAR experiment. The ratio of the measured-to-modelled results using MCNPX are given below in Table 1.

Table 1. Ratio of the Measured-to-Modelled [a] Results for the Cadmium Ratio Data of Fig. 3

Fuel Enrichment (wt %)	Cadmium Liner	Singles Count Rate (measured/modelled)	Doubles Count Rate (measured/modelled)	Triples Count Rate (measured/modelled)
3.19%	No	1.01	0.99	0.96
1.71%	No	1.01	0.99	0.97
0.23%	No	1.01	0.99	0.97
3.19%	Yes	0.96	0.86	0.72
1.71%	Yes	0.98	0.90	0.79
0.23%	Yes	1.01	0.99	0.95

[a] The statistical uncertainty of the modelled singles, doubles, and triples were typically 0.05%, 0.15%, and 0.32%, respectively.

From the data in Table 1, good agreement is generally found between experiments and modelling. Two points are noted: (1) There is a trend in the measured/modelled ratio that gradually reduces from singles to doubles to triples. This may be because the singles count rates is directly proportional to the efficiency whereas, using the point model as a general guide, the doubles count rate varies as the efficiency squared and the triples varies as the efficiency cubed. The published experimental efficiency of the PSMC was 55.0% and 54.1% [6], whereas the efficiency of the MCNPX modelled detector was 52.4%. (2) The agreement is poorer when more fissile material is present in the presence of the cadmium liner. Further investigation is needed to resolve this discrepancy.

1.1.3. Applying the PNAR Concept to the UREX^{+1a} Product Material

The final product of the UREX^{+1a} stream was selected to investigate the application of the PNAR technique to UREX^{+1a}. For simplicity, the same PSMC counter from the LEU benchmark study was used again. In the modelling of UREX^{+1a} materials, the cadmium on the PSMC walls was removed, and only one layer of cadmium was needed around the fissile material. Fig. 1, which was used to explain the basic concept of PNAR, also depicts the UREX^{+1a} product case as modelled. No effort was made to optimize the fission rate in the sample by altering the sample geometry or the surrounding moderator. An overall product mass of 2.5 kg was selected (1.9 kg of plutonium) since this mass is similar to the mass of one can of product typically used at a large-scale reprocessing plant [8]. The annular shape of the product was selected for two reasons: (1) Since the PNAR measurement effectively interrogates with thermal neutrons, it is desirable to have a relatively thin sample. And (2) criticality concerns encourage an elongated design. The cadmium layer, when present, enclosed the entire annulus of fissile material.

The isotopic mix of the product was obtained from AMUSE data and was for a burn-up of 33 GW-d/tonne and a cooling time of 5 years. The MCNPX-determined cadmium-ratio results are depicted in Fig. 4. The 100% case represents the expected results for the product, and the other data points are for the diversion cases with varying amounts of the plutonium removed. As plutonium was removed, no change was made to the neptunium, americium, or curium content. The density of the oxide was maintained at 2 g/cm^3 in all cases. As such, the fill height of the container changed with the plutonium content.

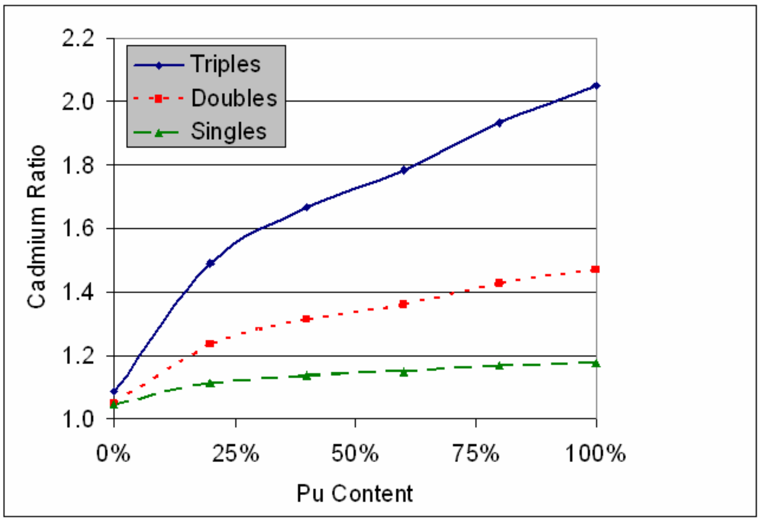

FIG. 4. The modelled cadmium-ratio results for the UREX^{+1a} product material for the experimental setup depicted in Fig. 1.

The modelled count rates, when cadmium was not present in the detector and when 100% of the plutonium was present, were 1.93×10^8 n/s, ± 0.06% for singles; 1.38×10^8 n/s for doubles; and 8.96×10^7 n/s for triples. When the cadmium was present, the rates obtained for singles were 1.64×10^8 n/s, ± 0.03%; for doubles were 9.49×10^7 n/s; and for triples were 4.37×10^7 n/s. The singles uncertainty value is for counting statistics only. Given the high count rate, it may be desirable to reduce the product mass or reduce the efficiency of the detector, or both. The PSMC detector has 80

^3He tubes. As it is, these count rates are above what a typical multiple tube multiplicity system can process. However, this count rate can be processed with list-mode data acquisition, with each ^3He tube having a dedicated amplifier.

1.1.4. Discussion of PNAR Results

If the PNAR technique is used in the field, the sensitivity of the singles, doubles, and triples must be quantified. Since the case, as modelled, has such high count rates, it is noteworthy that the precision for singles continually improve with count rate whereas those for doubles and triples do not. The relative uncertainty in the doubles is essentially independent of count rate, but the relative uncertainty of the triples value increases with count rate for the examined case, where the accidentals are much larger than the time-correlated events. For singles with a count time of 1000 s, the three-sigma uncertainty in the cadmium ratio value of 1.176 is 0.001% (statistical error only). This uncertainty would not be attainable even if there were no systematic uncertainties. In ideal measurement conditions, stability in the electronics limited the uncertainty to 0.005% [9]. Using 0.005% for the uncertainty, as the plutonium content is changed from having 100% to 80% of the plutonium present, the cadmium ratio value changes by ~380 sigmas. Hence a three-sigma change in the plutonium mass of ~2000 g is ~3 g. Given these results, the efficiency of the detector could be reduced by a factor of 100, which would make a more practical in-plant application.

In order to examine PNAR uncertainty further, the uncertainty of the benchmarked results presented in Fig. 3 and Table 1 were studied. The count rate difference between the UREX^{+1a} product just examined and the LEU-benchmarked case was more than 4 orders of magnitude, from 2×10^8 n/s to 7×10^3 n/s. The uncertainties used in the following analysis were experimentally determined, so the uncertainty from deadtime and electronic instability are included with the counting statistics.

As indicated in the last two columns of Table 2, the PNAR technique is very sensitive even at these relatively low count rates. As such, uncertainty introduced by the movement of cadmium or uncertainties in the oxide density are likely to dominate overall uncertainty. Furthermore, such systematic uncertainties will impact the uncertainty of singles more than doubles since the doubles cadmium ratios have a wider dispersion, as illustrated in Fig. 3 and Fig. 4. To quantify this point with the data from Table 2, if a 0.40% uncertainty in both the doubles and singles count rates resulted from putting the cadmium layer in place, this would introduce a 0.40% uncertainty in the cadmium ratio values. Combining this systematic uncertainty with the uncertainty from Table 1 provides a total uncertainty of 0.41% for the singles and 0.46% for the doubles. A 0.41% uncertainty (up from 0.067%) in the singles reduces the 55-sigma value in the last column of Table 2 to 9 sigma. In contrast, the 0.46% uncertainty (up from 0.22%) in the doubles reduces the 37-sigma value in the last column of Table 2 to 24 sigma. The greater dispersion in the doubles cadmium ratio results in a more sensitive measurement with the doubles, as compared to the singles, for realistic measurement conditions.

Table 2. Cadmium Ratio Uncertainty Examination for the LEU-Benchmarked Case of Fig. 3 and Table 1.

	Count Rate with Cadmium (counts/s)	Count Rate without Cadmium (counts/s)	Cadmium Ratio	One Sigma of the Cadmium Ratio (%)	Number of Sigma Values between 1.71% and 3.19%[a]
Singles	7.46×10^3	6.56×10^3	1.14	0.067	55
Doubles	4.40×10^3	3.27×10^3	1.34	0.220	37
Triples	1.65×10^3	9.40×10^3	1.76	0.655	23

[a] The variation in the units of sigma for the cadmium ratio as the measurement case changed from a LEU enrichment of 1.71% to 3.19%, not including systematic uncertainties.

Another systematic uncertainty that will need future attention upon implementation is the variation in the powder density. Higher density (or concentration for liquids) would increase multiplication for a fixed fissile material. Although the singles, doubles, and triples will all increase, the ratios of their values are indicative of a change in multiplication. To what degree changes in multiplication can be detected needs further investigation.

In the case of spent-fuel assay, an uncertainty in the density would be less likely since ceramic spent-fuels pellets' density should not vary much. However, the presence of significant lead shielding will increase the size of the detector; the die-away time will be slightly increased, and there will be some (n,2n) reactions in the lead. The shielding issue is also present when assaying UREX^{+1a} streams before the fission products are removed. These streams (after UREX or CCD-PEG processing) have the added issue of density variation due to dilution or evaporation. Once again, the multiplicity data may assist in quantifying these deviations.

The oxide UREX^{+1a} case studied here represented a very short cooling time, since this is what was available for the AMUSE data. If the material had cooled for much longer, say 40 years, the relative strength of the curium-244 (18-year half-life) would have reduced relative to the other actinides and (α,n) neutrons. The general reduction in the neutron flux reduces the sensitivity of the technique, but given that the sensitivity for a 5-year cooling time was extremely high, the sensitivity is expected to remain excellent. The change in the composition of the neutron source, from spontaneous-fission dominated to more (α,n) neutrons, is beneficial to the PNAR technique. The (α,n) neutrons are not correlated, so multiplicity counting will have a better signal relative to background.

If the fissile material is diffuse, highly self-moderating or large relative to the mean free path of thermal neutrons, the PNAR technique will perform less well. The change in the cadmium ratio from a diffuse quantity of fissile material will generally be small since the signal induced by thermal neutrons in the sample will be small relative to the other neutrons detected. The signal from a highly self-moderating sample will be poor as well since the cadmium ratio technique works by comparing two measurements, one with and one without low-energy neutrons incident from outside the object. If the low-energy neutrons incident from outside the sample are few in number relative to the low-energy neutrons inside the sample, then the two measurements will have almost identical signals. A large amount of moderator in the sample reduces the significance of the returning low-energy neutrons. If the sample is large relative to the mean free path of thermal neutrons then the signal from a given mass of fissile material will vary with location in the sample. Since the measurement situation (shape, concentration, etc.) of the sample has some flexibility, by taking the issues listed above into consideration, the effects of concern can be minimized. Further study is need after the key measurement points are determined for a UREX^{+1a} facility.

2. Summary

The PNAR concept was studied. This concept involves measuring the singles, doubles, and triples count rates from a fissile material twice; once with cadmium around it and once without cadmium present. By taking the ratio of these two measurements, a diagnostic is created that effectively interrogates the fissile material with thermal neutrons; the interrogating neutrons originate inside the sample containing the fissile material and are reflected back into the sample. Measurements described in a previous publication were modelled to show that the new multiplicity modelling capabilities of MCNPX accurately predict the multiplicity count rates whereas previous calculations based on the point model failed. Because there was confidence in the validity of the modelling, the product of the UREX^{+1a} process was selected for study. The PNAR technique was demonstrated to be very sensitivity in both the UREX^{+1a} case and the benchmarked experiment, so sensitive that the uncertainty in the assay technique is expected to be dominated by systematic uncertainties or instability in the electronics. The PNAR technique is of particular interest for the UREX^{+1a} process since curium and plutonium are never separated. There will always be plenty of neutrons available from the curium to interrogate the plutonium. The PNAR technique is also of use in the assaying of spent fuel, a measurement which will be of particular importance to the GNEP if pyrochemical reprocessing is used, since there is no accountability tank with pyrochemical reprocessing.

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Calibration check in accountancy tanks
Rokkasho-mura Reprocessing Plant

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Abstract. Accurate determination of volume is a fundamental component of nuclear material accountancy in a facility that processes such material in liquid form. In the Rokkasho-mura Reprocessing Plant (RRP), the inlet and outlet accountancy is performed in tanks that have been initially calibrated during the 'cold' tests in 2002. To ensure that the calibration curve V = f(H) remains valid during the entire lifetime of the plant, the operator performs a specific test once a year during a long shutdown period. This test involves pouring known amounts of water into the tanks and measuring the resulting height of the liquid with dip tubes (fast bubbling rate). The IAEA is measuring independently the levels and densities in the tanks using its own manometers. The temperature and mass values are the operator's data. During the test, inspectors randomly survey the data collected by the operator. Samples are taken in the tanks before and after calibration, and also in the server pot. These samples are independently measured in the On Site Laboratory (OSL). In order to limit errors, the procedure and conditions of the calibration check are similar to those of the initial calibration. Yet some differences cannot be avoided, such as the presence of the heel in the bottom in the tank, the presence of a hydraulic seal on the feeding line, the operation of the ventilation off gas system and the use of humidified bubbling air. The IAEA's evaluation of the calibration check results must take into account all possible errors or biases in order to detect any significant change in the way volumes are measured in this tank.

1. Introduction

The direct use of the calibration curve for volume determination in the inlet accountancy tank (IAT) and in the outlet accountancy tank (OAT) in the RRP requires a check of the validity of the calibration curves that were initially drawn up, in 2002, during the commissioning of the plant.

Initial calibration checks were performed in the IAT and in the OAT in November 2005, after completion of the uranium tests. The active commissioning of the plant started in April 2006, and both the IAT and the OAT have received nuclear material before the second series of calibration checks were performed in September 2006.

2. Purpose of the calibration check

The purpose of the calibration check is to make sure that the relationship between V and H has not been modified because of structural changes inside the tank or any physical/chemical phenomenon (e.g., fouling, corrosion, partial clogging of the dip tubes).

The basic principle of the calibration check is to pour known amounts of solution in the tank and to compare the corresponding volumes with the ones deduced from the calibration curve. The analysis of the residuals (Run – Curve) allows assessing the validity of the calibration curve used in normal operation. The calibration check is performed, when possible, with the same liquid as that used for the initial calibration.

The objective is to compare two volumes: the first is deduced from the mass of liquid introduced in the tank (V_{poured}); the second is calculated with the calibration curve $V_{curve} = f(H)$. To allow for the comparison, both volumes must be given at the same temperature, generally the tank temperature at the time of the measurement.

If the comparison is performed at the temperature of the tank, the calculation is as follows:

$$(V_{poured})_{Ttank} = \frac{\Sigma M}{\rho_{Ttank}}$$

where ΣM is the cumulated mass of liquid poured in the tank (after buoyancy correction of the masses measured in the server pot) and ρ_{Ttank} is the density of the liquid at the tank temperature (T_{tank}).

The second term of the comparison is calculated as follows:

$$(V_{curve})_{Ttank} = f(H_{Tref}).\ (1 + 3.\ \alpha \cdot (T_{tank} - T_{ref}))$$

$$H_{Tref} = \frac{\Delta P}{\rho_{Ttank} \cdot g} \cdot (1 + \alpha \cdot (T_{ref} - T_{tank}))$$

where f(H) is the relationship between V and H (calibration curve) at reference temperature (T_{ref}) and α is the linear coefficient of thermal expansion for stainless steel. ΔP is the differential pressure measured by the electro-manometer.

3. Conditions of the calibration check

3.1. Liquid used for the calibration check

The selected liquid must be one whose density can be adequately characterized. This means that the density of the liquid has been or can be determined with sufficient accuracy at all measurement temperatures so that density measurement uncertainty does not have a significant adverse effect on volume measurement uncertainty.

Both the IAT and the OAT have been initially calibrated with water. The density of water can be easily correlated with temperature. The initial checks in 2005 were done with water. In 2006, due to safety constraints, the OAT was checked with nitric acid 1N. The density of nitric acid can also be given by temperature correlations, provided the acidity is known. The in-situ densities determined with the dip tubes can be used to assess ρ_{tank}, but the accuracy of the measurements must be consistent with the target accuracy of the calibration check. It is preferable to have a good estimation of the density and temperature of each increment, especially at the beginning of the test.

A buffer tank of the same capacity as the accountancy tank should be used to feed the pouring station. This buffer tank must be filled with the water, or the acid solution that will be used for the calibration check, at least 12 hours before the beginning of the run in order to reach the temperature equilibrium.

3.2. Tank status before the test

3.2.1. Rinsing of the tank

The tank must be carefully rinsed before the test; the use of the decontamination ring can be considered if the tank is equipped. The rinsing solution can be introduced through the feeding line used during the calibration check. It will thus wet the inner wall of the line and fill up the hydraulic seal. Each rinsing batch must have a sufficient volume to allow sample taking by the lowest sampling airlift. The density and acidity of the solution must be measured, as well as the plutonium concentration (IAT/OAT) and the uranium concentration (IAT) for accountancy purposes.

3.2.2. Characterization of the heel

Heel volume. If the ‘low’ level of the tank after transfer is above the submergence level of the lowest dip tube, the determination of the heel volume is of less importance since the transferred volume will be the difference between two volumes deduced from measurable levels. Nevertheless, for inventory purposes it is recommended to estimate the heel volume with a reasonable accuracy; it might also be needed for dilution calculation. A spiking and dilution method can be implemented for that purpose.

Heel composition. The last rinsing operation should be performed with the same solution as the calibration liquid. Before draining the rinsing batch, a sample should be taken in the tank to confirm the characteristics of the remaining solution (density in case of water; density and acidity in case of nitric acid).

3.2.3. Verification of the instrumentation

The instrumentation must be calibrated or verified before the test in order to confirm that the performances of the measuring devices are still consistent with the required accuracies.

Instrumentation of the tank:

- Electro-manometers used for level and density measurements (verification, calibration if needed);
- Temperature probes (verification, calibration if needed); and
- Air flow meters and manometers on bubbling air (verification if needed).

Instrumentation of the server pot:

- Weighing machine (calibration);
- Temporary temperature probe (verification); and
- Hygrometer, manometer, thermometer for ambient air characteristics (verification if needed).

The densitometer used in the laboratory to check the density of the solution sampled in the tank or in the server pot should show a reasonable accuracy (at least 10^{-4}). The determination of the nitric solution acidity also requires some precautions, and it should be crosschecked with density measurements.

3.3. Conditions during the test

3.3.1. Feeding of the tank

The feeding line has a hydraulic seal (U seal) just before the tank. The flow rate through the pipe should be high enough to limit the duration of the pouring (about 2 m^3/h in half filled pipe conditions).

3.3.2. Ventilation of the tank

During the initial calibration, the manhole of the IAT was open and some pipes were not connected on both tanks. The calibration checks are performed under tight conditions and the ventilation off gas system is operating. Airflows inside the tanks may differ significantly from the initial calibration (injection of scavenging air for instance).

3.3.3. Bubbling system

The bubbling system must operate under the same conditions as those for the initial calibration. The flow rates must be adjusted to their nominal values. In particular, the humidification system should be stopped if the initial calibration had been performed with dry air. If not, the use of humidified bubbling air may result in a drift of the calibration curve (mainly due to the differences in air columns weight in the dip tubes).

3.3.4. Duration of the test

It is advisable to limit the number of checkpoints. Too many increments will increase the risk of operating and measuring errors, but also the duration of the calibration check. Increments should be selected to provide several points around the areas of the cut points and particularly in the areas of significant internal structures that manifest themselves in deviations from the fitted curve.

4. Possible sources of error

4.1. Temperature and density measurements in the tank bottom

Because the tanks are no longer accessible for the setting of temporary temperature probes, the beginning of the calibration check is performed without any temperature measurement of the liquid in the tank. Consequently the densities cannot be determined and the thermal expansion corrections cannot be calculated. The missing data can be estimated with the temperature of the poured liquid. The reliability of this estimate is dependent on the gap of temperatures between the tank and the feeding solution in the early steps of the calibration check. The temperature given by the emerged temperature probes may also help to assess the temperature of the heel.

If nitric acid is used as a calibration liquid, the in-situ density measurements are also not available at the beginning of the calibration check to confirm the density of the solution resulting from the mixing of the heel and the first increments. This is the reason why the temperature and the acidity (or the density) of the feeding liquid have to be measured in the server pot with a good accuracy, especially as long as the in-situ temperature and density measurements are not fully reliable.

4.2. Evaporation rate

The evaporated masses may evolve between the initial calibration and the calibration check, because of the following:

— The airflows inside the tank have increased or decreased because the ventilation changed;
— The temperature of the cell (and consequently the temperature of the tank) has changed;
— The temperature of the calibration liquid has changed; and
— The duration of the test has changed.

Provided that reliable estimates of the evaporation rates have been obtained during the initial calibration and during the calibration check, this error could be corrected. But it is highly advisable to limit the evaporation phenomenon as far as possible (e.g., shortened duration of the test, rather low temperature of the calibration liquid) or to be in conditions comparable to those of the initial calibration.

4.3. Homogenization of the tank solution

Although it might be necessary to homogenize the solution to reach at least temperature homogeneity, the use of a stirring device during the calibration check is not recommended in the case of a stirrer; it is totally prohibited in the case of a bubbling ring.

The use of a stirrer will induce a wetting of the tank wall on a significant height (a few centimetres). It will result in liquid retention and accelerated evaporation. It also tends to heat up locally the solution, adding concerns to the temperature correction. At the very least, the level, density and temperature of the solution in the tank should be measured just before and after stirring if the use of the stirrer cannot be avoided. The use of a bubbling ring poses a risk since the transient steps related to the opening and closing of the valve on the air feeding line may allow the liquid to flow into the ring.

As a consequence, the temperature homogeneity should be guaranteed, firstly, by achieving thermal equilibrium of the feeding liquid (see section 3.1) and, secondly, by minimizing the gap between cell and liquid temperatures.

4.4. Residual amount of liquid in the U-Seal

The presence of the U-seal on the feeding line can induce random errors since the residual amount of liquid in the seal might be a different amount after each pouring, especially for small increments.

5. Acceptance criteria

Because of all the potential errors described above, the calibration checks can never meet the accuracy of the initial calibrations. If the residuals (Run – curve) are beyond the tolerance curves determined during the initial calibration, this should be investigated by the operator. This statement should take into account the order of magnitude of the observed deviations compared to the target accuracy for the volume measurement in the tank.

6. IAEA evaluation

The IAEA uses independent values to assess the validity of the test performed by the operator. Both tanks are equipped with two high precision electro-manometers connected to the dip tubes used by the operator for level and density measurements. The electro-manometers are part of the SMMS-1 system (solution measurement and monitoring system). After digitization, the signals are sent to the IAEA database. The temperature measurement in the tank is shared with the operator and the split signal is sent to the IAEA database.

The chemical analyses are performed by the OSL in the RRP. The solution in the tank is sampled before and after calibration check for accountancy purposes but also for characterization of the heel and consistency check. The sampling is done automatically on a sampling bench and the jugs are sent directly to the OSL through the PTN network. The authentication of the jugs is performed using the ASAS (automatic sampling authentication system).

If nitric acid is used, a few local samples are taken in the server pot to check the density and acidity of the feeding solution. The operator performs the sampling and transfers the jugs in the OSL under the surveillance of an inspector.

All measurements done at the pouring station (weighing, temperature measurement) are performed by the operator with his measuring devices. The IAEA witnesses the verification of the weighing machine just before the test, and surveys randomly the data collected by the operator during the calibration check.

IAEA experience using the 3-Dimensional Laser Range Finder (3DLRF) for design information verification at the Rokkasho Reprocessing Plant

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Abstract. Design information verification (DIV) is one of the cornerstones for the development of a safeguards approach for a facility. The DIV is performed throughout the lifecycle of a facility in order to assure the absence of undeclared design changes and the continuous appropriateness and validity of the safeguards approach. For a large complex facility it is challenging to verify the accuracy and completeness of the declared design information and to monitor the modifications and changes that occur during the lifecycle of the facility. The DIV activities in such a plant require a tremendous investment in human and financial resources and time. Large complex facilities also have inherent design features that impose limits on the accuracy, completeness, and continuity of knowledge achievable using the traditional DIV methodology. Additionally, physical access to places of safeguards interest may be difficult. In 2003, the IAEA introduced an innovative device, the 3-dimensional laser range finder (3DLRF), that is capable of meeting some of the challenges involved in performing a DIV at large complex facilities. This paper describes the IAEA's experience in using the 3DLRF at the Rokkasho Reprocessing Plant (RRP) in Japan.

1. Introduction

The purposes of design information verification (DIV) are:

(a) To provide confirmation that a facility is being/has been built and/or modified to conform to the operator's stated design and purpose; and

(b) To provide one of the bases for the development and continued updating of the safeguards approach.

To these ends, the IAEA has traditionally used methodologies that rely on inspector access to the specific locations of interest. However, access may be difficult due to limited space, height, sensitive equipment or safety concerns. This was the situation in many instances at the Rokkasho Reprocessing Plant (RRP) in Japan. The physical dimensions of most of the process cells and the placement of equipment in the cells imposed limits on the inspector's physical access to locations of safeguards interest. The access restrictions to some essential equipment and the complexity of the design posed a possible compromise to completeness, accuracy and continuity of knowledge.

The DIV of the RRP started prior to construction when the ground was first prepared to lay the foundations. As construction progressed it was necessary to begin verification activities within the cells. However, at that time most of the cells were without adequate lighting. The lack of lighting complicated the use of normal design verification methods, such as dimensional measurements and pipe tracing, and of documentation by sketching or photography. There was a need for an innovative and robust tool that could cope with the challenge of verifying design features in adverse conditions and also provide reference documentation for the lifetime of the plant. Also, a methodology was needed that could simplify the verification activities, provide long-term continuity of knowledge of the verified design and reduce the in-field efforts. The solution was found in the 3-dimensional laser range finder (3DLRF) which was developed by the Joint Research Centre (JRC) of the European Commission, under the umbrella of the Support Program of the European Commission to the IAEA Department of Safeguards.

The 3DLRF system enables the construction of virtual 3D models to be used to confirm the up-to-date validity of design information and to support the assessment of DIV results away from the DIV location. The system enables the inspector to perform dimensional determinations on pipes and vessels and to establish equipment locations relevant to each other and also provides a reference for future re-verifications.

This paper describes how the 3DLRF was incorporated into the DIV scheme used at the RRP and the overall resulting experience.

2. Plant information

The RRP is an 800t U/year light water reactor (LWR) spent fuel reprocessing facility. It is the largest and most complex facility ever submitted for IAEA Safeguards.

The RRP is located on a site of about 4 000 000 m^2 with 38 buildings, 20 of which house process and storage facilities. This represents thousands of cells and rooms, tens of thousands of pieces of equipment and more than 1500 km of piping.

3. Scope of the DIV at the RRP

From the time of receipt of the provisional design information for the RRP, it was apparent that a complete verification of the plant design and its equipment would not be achievable due to size and complexity. Therefore, an approach based on prioritization and random selection of cells, pipe runs and equipment was introduced into the DIV scheme.

The importance of prioritizing DIV tasks was previously highlighted by the LASCAR expert forum at the beginning of the 1990s [1]: “To be practicable, design verification should concentrate on those design features of prime importance to safeguards and must be timed to take place before the particular design features become inaccessible”

The introduction of this approach for the use of the 3DLRF resulted in the scanning, verification and documentation of 60 of the cells classified as safeguards significant.

4. Description of the 3DLRF

The 3DLRF for a DIV was designed to be a portable system for routine inspection use [2]. The system for data acquisition consists of a portable commercial laser range scanner and a portable computer. These can be mounted on a tripod with a dolly and are battery operated (Figure 1).

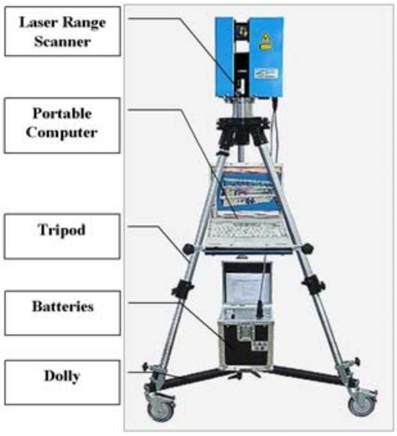

FIG. 1. 3DLRF data acquisition equipment.

The maximum range of the scanner used at the RRP is 53.5 m, measuring distances with accuracy better than 5mm.

The data processing software (DIV tool) has been specially developed by the JRC and adapted to the needs of IAEA safeguards. It allows for the following operations:

(a) Creation of a global 3D model from scan data taken at different positions;

(b) Automatic comparison of the reference 3D model with newly acquired scans;

(c) Viewing of the scanned data in 2D or 3D;

(d) Navigation within the 3D model;

(e) Viewing of global 3D model from different perspectives;

(f) Taking dimensional measurements; and

(g) Including annotation within the 3D models.

The developer of the system, JRC, has also developed a mobile variant of this system: the outside viewing system (OVS) that has a range of 1 km and can be used outside of buildings or in open areas. Based on the success in using the 3DLRFD, the OVS may be evaluated for use in verifying a nuclear site as a whole.

5. Use of the 3DLRF at the RRP

The approach which was used with the 3DLRF was as follows:

(a) Reference 3D models of the most important cells were built by acquiring a number of scans at different positions in the cells;

(b) The initial verification was performed by visually comparing the 3D models of cells with the engineering drawings provided by the operator; and

(c) Re-verification of the cells was performed just prior to their final closing. At that time, new scans of selected areas were taken and compared with the reference models in order to detect any changes since the initial verification.

5.1. Equipment and materials

The equipment and materials consisted of the following:

(a) The 3DLRF,

(b) Targets,

(c) Operator drawings: layout drawings, vertical section drawings and equipment drawings for the selected cell, and

(d) The 3DLRF working paper.

5.2. Site survey

The drawings provided by the operator were used to confirm that the cell was indeed the one selected.

The 3DLRF working paper was used to make a sketch of the cell prior to scanning. The sketch indicated the major equipment in the cell, the location of the entrance and the positions (capture points) selected for placing the laser during scanning.

Surveying the cell for the selection of capture points was an important activity before scanning. Capture points were selected to maximize the overlap between scan images taken from different positions. The quality of the global 3D model that was built depended, among other things, on the amount of overlap between the individual scans.

Site survey was also performed to select positions to place targets around the cells. The inspectors found that for a complex cell with similar looking equipment and piping it was difficult to effectively find common points without the help of the targets.

5.3. Setup

The configuration of the 3DLRF shown in Figure 1 was not possible in the crammed conditions inside most of the cells at the RRP. Furthermore, because the inspector had to carry the 3DLRF up ladders and scaffolding, or crawl with it under piping, the 3DLRF was mainly used without the tripod and the dolly (see Figure 2). The setup of the system is straightforward and takes about 20 minutes to complete.

FIG. 2. Inspector using the 3DLRF at RRP.

5.4. Handling

The dimensions of the 3DLRF are 300 x 180 x 350 mm^3, and it weighs 15 kg. The equipment is therefore quite portable and it was possible to manoeuvre it in the tight environment within the process cells.

5.5. Safety

The laser used in the 3DLRF has a safety classification of 3R. The risk of injury from such a laser is low, but direct viewing of the beam should nevertheless be avoided. The inspectors were always careful to maintain the safety distance of 75 cm.

5.6. Data acquisition

Based on the site survey, scan data were taken from different capture points around the cell. The number of scans depended on the complexity of the cell. On average, about six scans were necessary to build a representative 3D model of one standard RRP cell in about two hours. Each scan consisted of a 2D component and a 3D component of the area viewed. A typical high resolution scan took on average less than two minutes to acquire.

After each scan, the acquisition software provided a first visualization on the computer screen of the acquired data. This gave the inspector an impression of the quality of the data collected and the validity of the capture point used for scanning.

All the individual scan data that were taken during the initial verification, as well as the global 3D model that was later built using the individual scan data taken during the initial scanning, were considered the ’reference scan data’ for the cell.

During the initial scans, it was sometimes not possible to build a complete 3D model of the entire cell because of scaffolding in the cell. Hence raw data were collected only for those parts of the cell that were not obstructed. During re-verification scanning which took place just before the final closure of the cells, no scaffolding was in the cells. The new data could be used to build the global 3D model for the cell.

A convenient feature of the system is that for re-verification scanning prior knowledge of the exact position of the scanner during initial scanning was unnecessary. Documentation of capture points on the 3DLRF working paper was to give the inspector the information of the field of view of the particular scan.

No system failure has been experienced since the introduction of the 3DLRF in 2003.

5.7. Data aalysis

Data analysis consisted of two main steps:

(a) Scan registration: Combining the individual 3D scan data taken from different positions into a global 3D model for the cell; and

(b) Verification: Comparing the scan data with the operator’s drawings for the initial verification, or automatic comparison of the reference scans with new scans.

5.7.1. Scan registration

The combination of the individual scans into a global 3D model consisted of two steps:

(a) A manual pre-registration step that involved identifying common points in the different scans; and

(b) An automatic registration process that is performed by the processing software.

The main challenge in the data analysis was the manual combination of scan data. This required good visual observational skills and good site survey choices. In selecting common points between scans, we found that the quality of the 3D model improved under the following conditions:

(a) The common points were spread around the cell and not clustered;

(b) The number of common points were increased; and

(c) There were common points on the equipment within the cell as well as on all the surfaces (i.e. all walls, the floor and the roof).

During the initial scanning, it was sometimes not possible to build a global 3D model of the entire cell because of scaffolding in the cell. This had no impact on the re-verification because reference scan data from individual scans were used for the comparison.

During re-verification scanning which took place just before the final closure of the cells, no scaffolding was in the cells. In cells which previously had scaffolding and other obstructions, the new data were used to build global 3D models.

5.7.2. Verification

During the initial verification, the data processing software was used to take dimensional measurements in the 3D model and to make comparisons with the cell drawings provided by operator.

During re-verification, the data processing software was used to make automatic comparison between the reference scans and the new scans (see Figure 3).

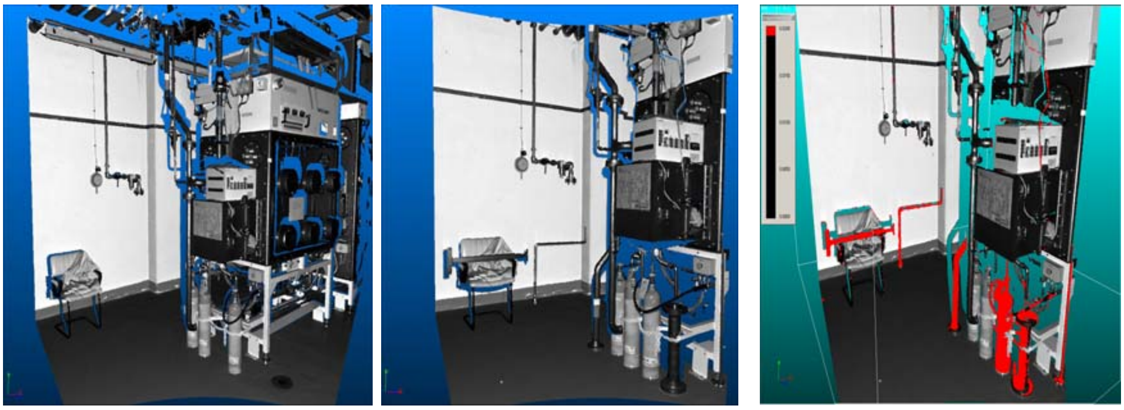

FIG. 3.

(a) Reference model *(b) New scan* *(c) Automatically detected differences (in red)*

5.8. Data security

The data acquisition computer and the data processing computer are stand-alone machines that are kept on site at the facility. In order to protect commercial and security sensitive information, the 3D data (which are stored on CDs or DVDs) are kept at the facility in cabinets jointly sealed by the IAEA and the State.

Note: The system as currently designed cannot be used in contaminated areas. Decontamination of the surface is not a problem, but the system is fitted with a cooling system which could pull contamination into the 3DLRF.

6. Conclusions

The 3DLRF has demonstrated that it is a valuable tool for use in verifying that complex and difficult to access features of a large nuclear facility are designed as declared. It also has become an invaluable tool for documenting the ‘as-built’ design features for future periodic verifications of design.

It will continue to be used at the RRP during the remaining initial verification of the MOX handling areas. Based on the experience at the RRP, the 3DLRF will now be used more widely by the IAEA at other complex nuclear facilities to perform the initial DIV and later DIVs so as to assure that there are no undeclared design changes.

The 3DLRF has permitted the compilation of unprecedented amounts of verification data, which may be used for the following purposes:

(a) Documentation and reconstruction of the areas scanned;

(b) Verification that the main plant configuration and capacity are, or are not, as declared;

(c) Monitoring changes in the area scanned; and

(d) Taking dimensional measurements of large structures (without the need to return to the DIV location).

The 3DLRF was found to have a high data acquisition rate, to be robust, compact and highly transportable. It was also found to be efficient, cost effective and reliable.

Since the introduction of the 3DLRF in 2003, there has been continuous feedback and consultation between the IAEA users and the JRC developers, resulting in user friendly software modifications and updates.

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Extensive cooperation in establishment and installation of safeguards system at Rokkasho Reprocessing Plant (RRP)

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Abstract. The IAEA has never before been experienced with designing a credible safeguards approach for a large scale reprocessing plant. The safeguards discussion was initiated in 1988, but at that time no available model or guideline that could be used as a reference.

The IAEA, the State and JNFL have studied and discussed the safeguards approach and safeguards system, and established effective, efficient and credible safeguards for RRP, under extensive cooperation manner. This paper presents overview of the safeguards system at RRP from the operator's viewpoint.

1. Introduction

Rokkasho Reprocessing Plant (RRP) has an operating throughput of 800tU/y and is under active test using actual spent for final confirmation of equipment and system, and the commercial operation is scheduled in August 2007.

In order to establish an effective and efficient safeguards at RRP, the safeguards discussion with the IAEA was initiated in 1988, but at that time no available model or guideline that could be used as a reference. Therefore, during the period of 1988 through 1992, a multinational forum, referred to as LASCAR (Large Scale Reprocessing Plant Safeguards)[1], provided the following recommendations that how an effective safeguards approach could be implemented to a large scale reprocessing plant with maintaining an efficient use of resources.

- High accurate measurement systems for the nuclear material accountancy;
- Timeliness of verification by advanced nuclear material accountancy techniques
- Redundant and independent containment and surveillance systems;
- Early consultations on facility characteristics especially early and continuing design information verification;
- Authentication of equipment and systems made available by operator;
- On-site verification capabilities (On-site laboratory);
- Data acquisition and transmission; and

- The on-going or needed research and development tasks.

The Project objectives were to plan, coordinate and integrate all activities necessary to ensure that an effective and efficient safeguards would be implemented at RRP on a schedule consistent with construction and commissioning of the plant and resource expenditures within the IAEA, Japan Safeguards Office (JSGO), JNFL, and Member States Support Program capabilities for the IAEA safeguards.

Funding for the safeguards systems have been shared between the IAEA, the State and JNFL. A significant amount of development work and provision of safeguards systems has been made through the Member State Support Programs. The large efforts were made for RRP safeguards establishment in cooperation with IAEA, the JSGO and JNFL have yielded a robust safeguards approach, which employs state of the art technology and innovative methodology.

2. Facility Description and MBA Structure

RRP is comprised of the Spent Fuel Receipt and Storage area (currently in operation), the Head-end, a Main Process which includes uranium oxide conversion, the U/Pu Co-denitration Conversion Process, the Mixed uranium/plutonium Oxide (MOX) and Uranium Oxide Storages, the Waste Treatment areas including the Vitrification process for the high active liquid waste and Storage areas.

For nuclear material accountancy purposes, the facility is divided into five Material Balance Areas (MBA):

MBA-1: Spent fuel receipt and storage area, head-end area
MBA-2: Main process area (including U conversion and laboratories)
MBA-3: Waste treatment and storage area
MBA-4: MOX conversion area
MBA-5: Product storage area

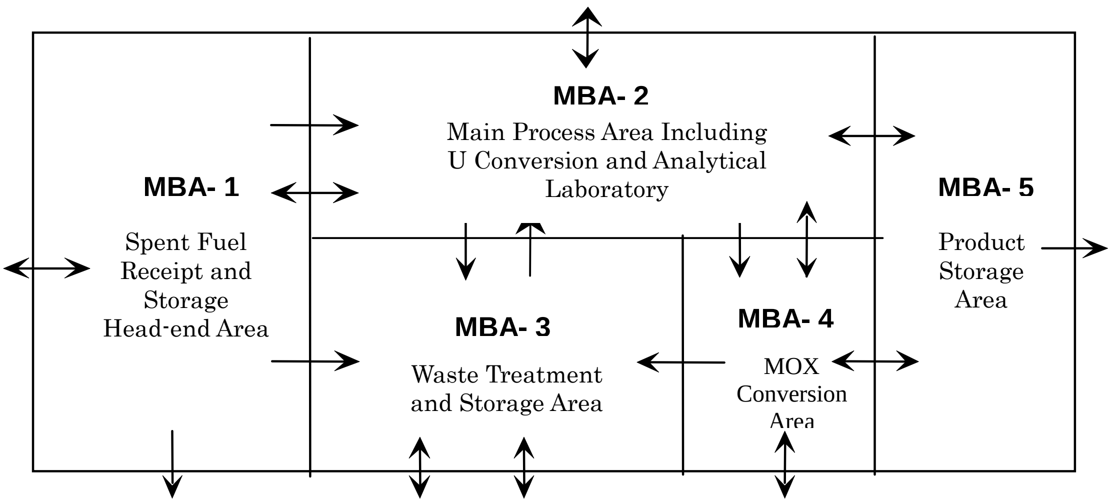

Figure 1 MBA Structure for the Material Accountancy at RRP

3. Development of Safeguards Equipment and the Use of Operator's Equipment

3.1 Development of safeguards equipment and joint use

The IAEA, the JSGO and JNFL have cooperated in the development of unattended verification systems including NDA systems, automatic sampling authentication system and C/S equipment.

Frequent Working Group discussions were held with the IAEA concerning the safeguards systems, cost shearing and responsibilities.

The measurement, monitoring and surveillance systems required for the implementation of the safeguards approach to RRP encompass a wide range of technologies and applications. There are more than 50 measurement and/or monitoring systems and approximately 70 camera systems. This represents not only a large financial burden, but also a huge demand on human resources for preparation of user requirements, installation and testing. Therefore, systems have been developed and installed that will be jointly used by the JSGO and IAEA inspectorates, and in some cases also by the operator.

3.2 Use of operator's equipment for safeguards inspections

The LASCAR recommended that the Inspectorates should utilize independent equipment to the extent possible. However the LASCAR also recommended that the plant operator's equipment would be utilized through appropriate authentication, if the installation of independent safeguards equipment is impracticable due to limited resources of the IAEA, plant space difficulties, plant safety features, etc.

With the cooperation of the operator, utilization of the operator's equipment is widely implemented in RRP based on signal splitting through appropriate authentication.

Considering that they will be operated in unattended and remote mode, there has been a large effort made toward assuring that neither the systems nor the data can be tampered. The authentication measures employs physical containment, use of reference standards and so on.

3.3 Providing timely analytical results by Onsite Laboratory (OSL) with Automatic Sampling Authentication System

The On-site laboratory (OSL), dedicated to analysis of safeguards samples, has been built in RRP analytical laboratory so as to be connected to RRP through pneumatic tubes for sample transfer from sampling benches located throughout the facility. The OSL provides very timely analysis results for the IAEA and the JSGO. This is particularly critical in a high throughput facility such as RRP. Results are needed within days so that the inspectors can evaluate material flows and inventories. It also allows the inspectors to identify questionable results or to request follow-up activities, such as additional analysis. For the reason of timely results and the cost of shipping samples to the IAEA laboratory in Austria (SAL), this lab will be jointly used by the IAEA and the JSGO and strict measures will be implemented to assure that independent results will be obtained.

The sample integrity and authenticity will be achieved by the Automatic Sampling Authentication System (ASAS) [3] which is based on use of Independent Jug Passage Detectors (IJPD) installed on the Pneumatic Transfer Network (PTN) and the Solution Measurement and Monitoring System (SMMS) [4] information during sampling. The IJPD sensors will track the empty sampling jug from its origin to the sampling bench and then to the OSL. The ASAS will provide sample integrity and authenticity by information of the IJPDs, density and solution level fluctuation information from the SMMS during the sampling and the OSL analytical results compared with the operator declared data.

3.4 Cost Shearing and Responsibilities for Safeguards Equipment

RRP safeguards equipment were developed and installed successfully under the extensive cooperation among JNFL, the JSGO and the IAEA with Member State Support Program (MSSP) for the IAEA safeguards. The cost shearing and responsibilities of RRP safeguards equipment is shown in "Table 1 Safeguards Equipment and Cost Shearing at RRP", and safeguards system at each MBA is shown in "Figure 2 Overview of Safeguards System at RRP".

Table 1. Safeguards Equipment and Cost Shearing at RRP.

Equipment Name	Location	Responsibility	MSSP Task ID
ASAS: Automatic Sampling Authentication System	MBA-1, 2, 3 and 4 ; Pneumatic transfer network between OSL	JNFL/JSGO	French FRA-A-01099
CRD: Combined cameras and Radiation Detectors	MBA-1	JSGO	Japan JPN-E-1188
DCPD: Directional Canister Passage Detector	MBA-4 and 5	JSGO	
IHVS: Integrated Head-End Verification System	MBA-1	JSGO	Japan JPN-E-1188
ISVS: Integrated Spent Fuel Verification System	MBA-1	JSGO	Japan JPN-E-0819
MSCS: MOX Storage C/S System (surveillance camera & others)	MBA-4 and 5	JSGO/IAEA	Japan JPN-E-1281 USA USA-A-1351
iPCAS: improved Plutonium Canister Assay System	MBA-4	JSGO	USAUSA-A-351
IPLC: iPCAS Load Cell	MBA-4	IAEA	
OSL: Onsite Laboratory		JNFL/JSGO	
PIMS: Plutonium Inventory Measurement System	MBA-4	JNFL	UK A-01310
RHMS: Rokkasho Hulls Drum Measurement System	MBA-1	IAEA	USA USA-A-1317
SMMS: Solution Measurement and Monitoring System including Solution Monitoring Software(SMS)	MBA-1,2,3, and 4	SMMS-1 : IAEA/JSGO SMMS-2: JNFL/JSGO	(SMMS-1) Japan JPN-D-0730 USA-C-01257 FRA-A-01261 & -01352 (SMMA-2) USA-C-01257 FRA-A-01261 & -01352
TCVS: Temporary Canister storage Verification System	MBA-4	IAEA/JNFL	
UBVS: Uranium Bottle Verification System	MBA-5	IAEA	
USCS: Uranium Storage C/S System (surveillance camera)	MBA-5	IAEA	
VCAS: Vitrified Canister Assay System with surveillance camera & DCPD	MBA-3	IAEA	USA-A1317 and USA-A-1351
WCAS: Waste Crate Assay System	MBA-3, 5	JNFL	USA-A-1351
WDAS: Waste Drum Assay System	MBA-4	JNFL	

In the spirit of cooperation, Japanese side provided more than 3/4 of the initial investment for the safeguards system at RRP.

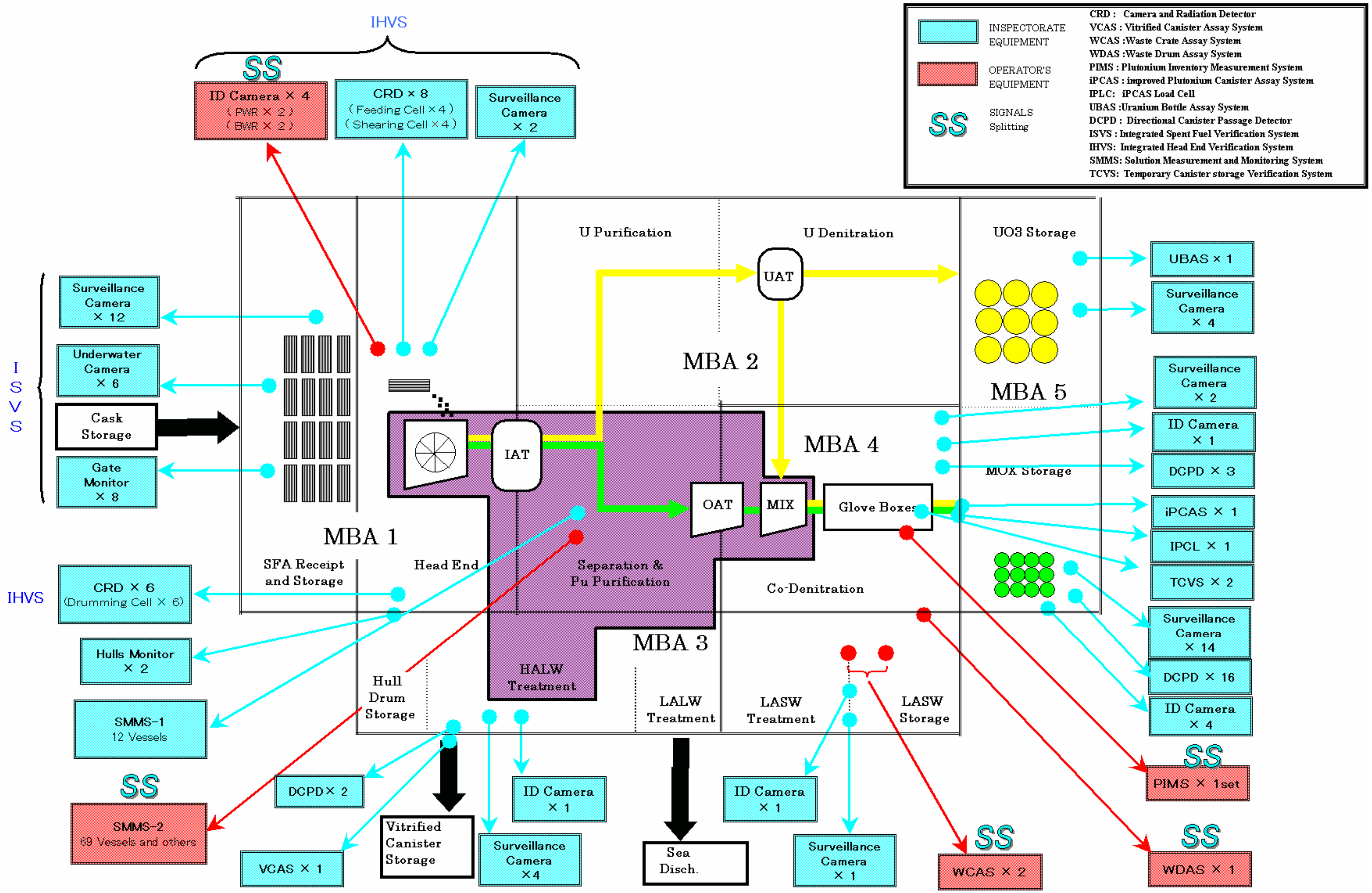

Figure 2 Overview of Safeguards System at RRP

4. Overview of the Safeguards Approach

4.1 Initial and Continuing Design Information Examination / Design Information Verification (DIE/DIV)

The possibility of verifying 100% of the safeguards relevant design features of a commercial size reprocessing plant is beyond the available resources of both the IAEA and the JSGO inspectorates. Therefore, priorities were established for verification of design information in the areas of building layout, cell design, equipment designs, installation and testing, piping and pipe penetrations, active trenches and vessel calibrations.

DIV was performed based on following concept in order to optimize the limited resources.

- Classification of all plant components to be verified based on the safeguards significance in order to prioritize the verification;
- Random verification for the medium and low significant class in order to reduce verification activities.

The tools of DIV such as endoscopes, 3-D Laser Range Finder for DIV (LRFD) [5], wall thickness gauges, digital photography, and portable electromanometers, along with human observation were used to compare the engineering drawing and actual equipment. Due to the sensitive nature of much of design information, it was necessary that it must be kept in the IAEA and the JSGO controlled cabinets at the RRP.

4.2 Key of Safeguards Approach

Considering large throughput of the RRP, the sensitivity and complexity of accepted technologies, and the inherent inaccessibility of nuclear materials, a robust and comprehensive safeguards approach was needed. It was also recognized that by simply adhering to the minimum verification requirements called for in the IAEA Safeguards Criteria, there would be inadequate sensitivity for detecting the diversion of a significant quantity of material or misuse of the facility. In order to overcome these difficulties, additional measures should be needed, which are SMMS and Plutonium Inventory Measurement System (PIMS). The objectives of additional measures are as follows;

- To provide additional assurance that the plant is operated as declared;
- Improve and/or support verification of the material accountancy system for conventional and Near-Real-Time-Accountancy (NRTA) in unattended mode;
- Monitoring of material (solution and powder) flows in process and inventory, and maintain continuity of knowledge on the verified design of the plant; and
- Provide validation of the other safeguards systems, including the automated sampling system.

The primary areas of verification activities can be divided into inventory changes, inventories and flows within the MBA. In addition, Other Strategic Points (OSP) are identified that will provide confirmation of the operational status of the facility as declared.

The safeguards approach is based on material accountancy as the fundamental measures, unattended NDA systems, C/S measures, process monitoring, environmental sampling and continuous inspector presence.

A summary of inspection activities with safeguards equipment in each area is shown Figure 3: “Inspection Activities on Rokkasho Reprocessing Plant (RRP)”.

5. Conclusion

RRP safeguards equipment were developed and installed successfully under the extensive cooperation among JNFL, the JSGO and the IAEA with Member State Support Program (MSSP) for the IAEA safeguards. DIVs during construction period were performed quite smoothly without undue interference of the construction schedule by the IAEA and JSGO made excellent cooperation and efforts.

JNFL has implemented quite accurate material accountancy system even waste stream. However, in case of a large scale reprocessing plant, it is obviously difficult to detect of 1 Significant Quantity (SQ) of plutonium by conventional material accountancy.

Therefore a number of monitoring systems and unattended verification systems have been introduced along nuclear material flow, which is from the spent fuel storage until the product storages. These systems will provide credible safeguards assurance as additional safeguards measures, because any possibility for removal of nuclear material from processes will be eliminated.

Currently, the uranium test was done and the active test is being performed. Through these confirmation tests, evaluation parameters for verification on monitoring will be obtained. Also appropriate inspection accountancy procedure and inspection schema will be established and confirmed.

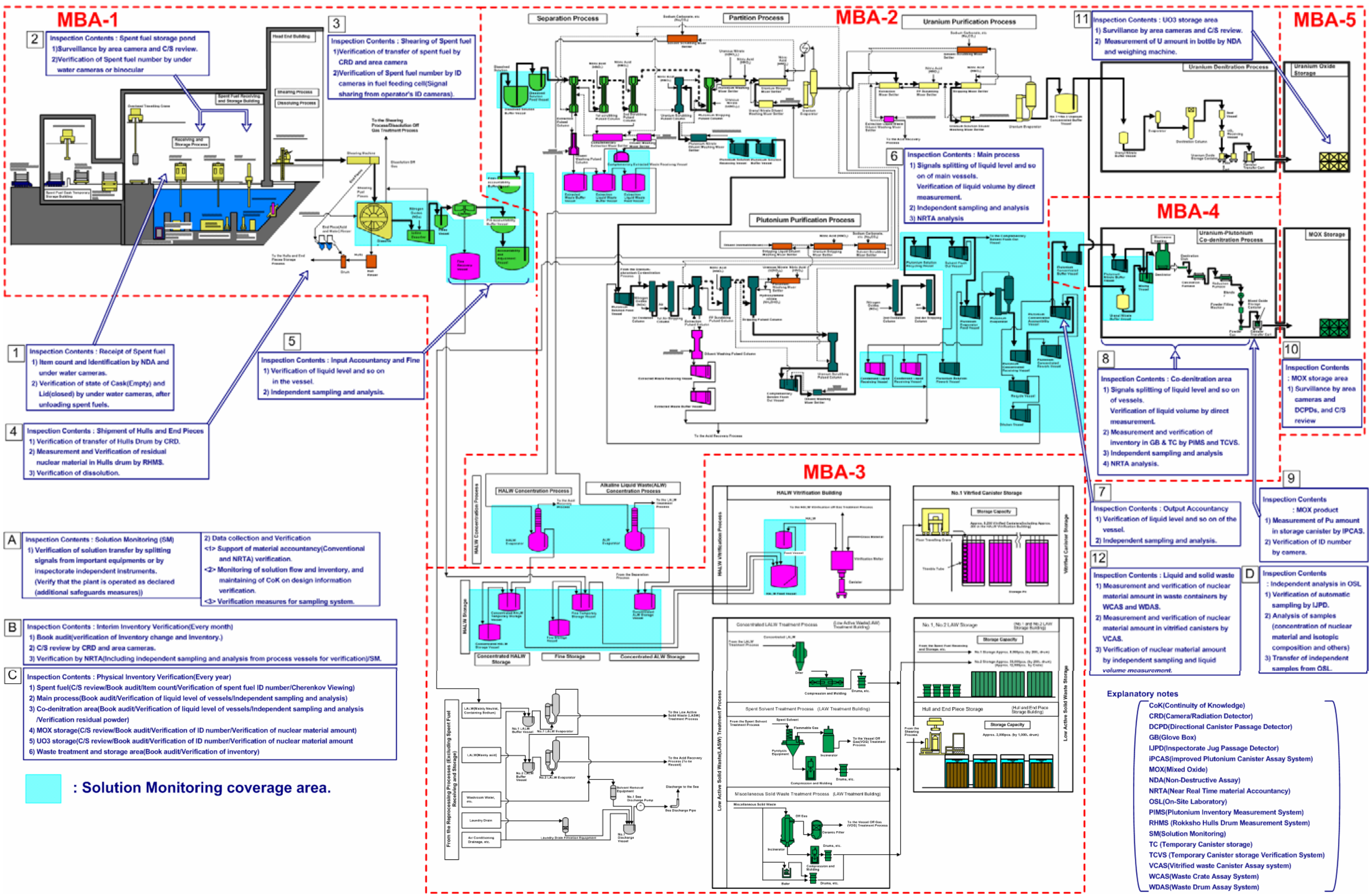

Figure 3 Inspection Activities on Rokkasho Reprocessing Plant (RRP)

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Safeguards improvement for the Tokai Reprocessing Plant (TRP)

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Abstract. Safeguards have been implemented at TRP with cooperation between the Government of Japan and the IAEA. JAEA has actively cooperated with the safeguards activities, and has also tackled the development of safeguards technology for the TRP under the Japan Support Programme for Agency Safeguards (JASPAS) and a cooperation research program with Department of Energy (DOE). IAEA has established new policies in order to improve the strengthening and the efficiency of the IAEA's safeguards in 1993. As a result the IAEA requested the Safeguards Improvement Plan for TRP in 1995. The main improvement areas were: improvement of verification for design information; improvement of reliability of verification for primary tanks; establishment of waste verification methods; and improvement of evaluation methods for material accountancy. JAEA had actively cooperated in these improvement items on the basis of the regular discussion and consultations among three parties. This paper will report the accomplishments for the improvement plan.

1. Introduction

TRP started its hot operation test in September 1977. The total reprocessed amount was ca. 1,120 tons of uranium of spent fuel on the end of July 2006. The operation history is shown in figure 1. TRP had accomplished to reprocess the spent fuels based on the reprocessing service agreement with electric power suppliers on the end of March 2006 and has changed its objective of the operation as research and development. In addition, JAEA has constructed a Low level radioactive Waste Treatment Facility (LWTF) for reducing volumes of low level solid and liquid wastes, and will start a hot operation at the end of 2007.

Safeguards have been implemented at TRP with cooperation between the Government of Japan and the IAEA. During reprocessing campaigns IAEA inspector carries out twenty-four hours inspection activities by having continuous presence at TRP along with inspectors of the Government of Japan. JAEA has actively cooperated with the safeguards activities and has also tackled the development of safeguards technology for the TRP under the JASPAS and a cooperation research program with DOE.

As described[1], a TRP Safeguards Improvement Plan started in 1995 before Japan's ratification of the Additional Protocol with ten tasks for improvements of design information, monitoring of main flow and inventory, waste monitoring and material accountancy.

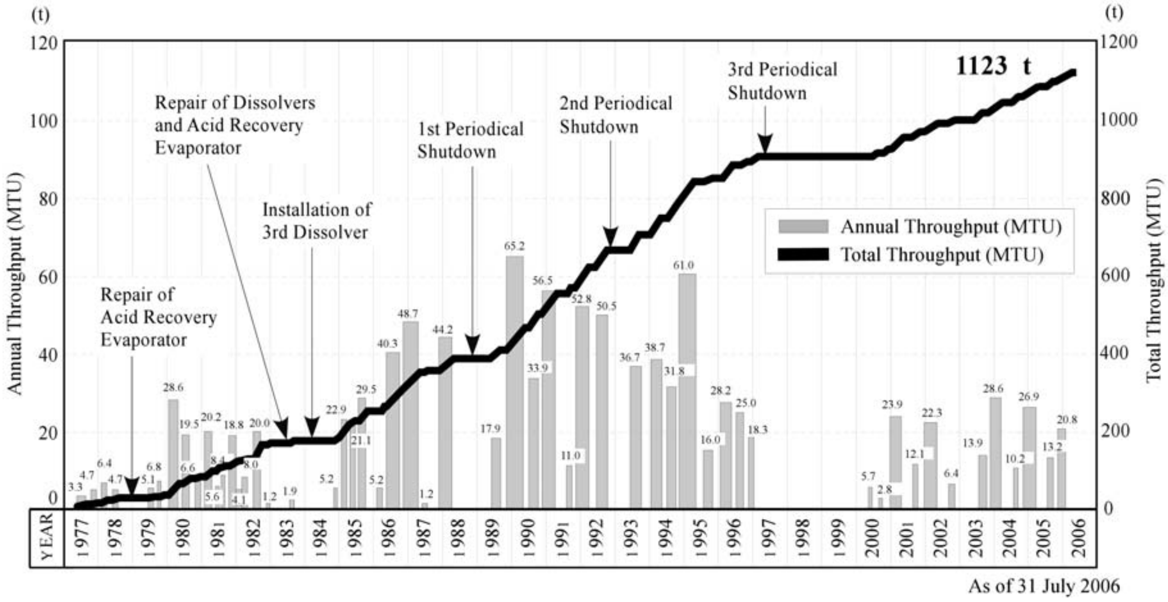

FIG. 1. Operation history of TRP.

2. Safeguards improvement Plan

The Improvement Plan consisted of following four areas and ten tasks.

(1) Improvement of Verification for Design Information
  Task 1 : Revision of the Design Information Questionnaire (DIQ)
  Task 2 : Design Information Verification (DIV)
(2) Improvement of Reliability of Verification for Primary Tanks
  Task 3 : Solution Measurement and Monitoring System (SMMS)
  Task 4 : Sample Integrity
(3) Establishment of Waste Verification Methods
  Task 5 : Verification of Hulls Cask Movements
  Task 6 : Transfer Verification of Transfer of High Active Liquid Waste (HALW)
  Task 7 : Verification of Vitrified HALW
(4) Improvement of Evaluation Methods for Material Accountancy
  Task 8 : Shipper/Receiver Difference (S/RD)
  Task 9 : Near Real Time Material Accountancy (NRTA)
  Task 10 : Provision of Data to IAEA by Diskettes

The summary of the accomplishments for each task is as follows.

(1) Improvement of Verification for Design Information

Task 1 : Revision of the Design Information Questionnaire (DIQ)

In order to evaluate the validity of the current SG approach, IAEA required that JAEA to provide more detailed information. The DIQ of TRP was made in 1978 and has been revised along with additions of new facilities such as the high level liquid waste storage facility and the third uranium product storage facility. Under the Improvement Plan, JAEA rewrote the text of DIQ in detail and attached more detailed drawings and supporting information for the various equipment to the DIQ. In addition, sensitive engineering drawings for each process and primary vessels were provided and have been stored in a on-site joint box controlled by the inspectors. During the Design Information Examination (DIE) carried out by the Government of Japan and IAEA, JAEA cooperated by explaining the purpose of equipment and details of the drawings. In addition, JAEA provided "Procedures for Accountancy Control at each Key measurement point" as the related document of DIQ. These enabled the IAEA to establish more effective and efficient Safeguards for TRP.

Task 2 : Design Information Verification (DIV)

In order to establish the DIV procedure to verify the completeness and correctness of the DIQ, IAEA required to implement the lifetime DIV activities during routine operation, in conjunction with maintenance or modifications, during extended shutdown (including closed-down) and during decommissioning based on a DIV Plan, and also required the inspection activity for confirmation of plant operation status at the Other Strategic Points (OSP). The DIV plan proposed by the IAEA was discussed and agreed among the IAEA, the Government of Japan and the JAEA, and the lifetime DIE/DIV activity have been implemented in accordance with the agreed DIV plan. For the confirmation of the plant operation status JAEA proposed the places and indicators where such activities could be carried out. JAEA also cooperated the IAEA's plant tour in order to evaluate the validities of the places and indicators. As a result, the places were defined as the OSP in the Facility Attachment (FA). The activity for the confirmation of plant operation status has been implemented monthly in a short notice and random basis. The activities for lifetime DIV and the confirmation of plant operation status can provide an additional assurance of the absence of undeclared design changes and undeclared operations.

(2) Improvement of Reliability of Verification for Primary Tanks

Task 3 : Solution Measurement and Monitoring System (SMMS)

JAEA has carried out development of a continuous monitoring system to collect data, such as solution level in the plutonium output accountability tank and the plutonium solution storage tanks through TASTEX and JASPAS. The IAEA obtained monitoring data from the original system at the field inspection. However, the IAEA's evaluation concluded that the system's authentication function was not adequate. In the Improvement Plan, the IAEA required an authentication function and additional monitoring of the input accountability tank. According to this requirement, JAEA carried out further development under the JASPAS program and installed the SMMS that has data transmission function. The system has been used for verification of operator declarations and confirmation of solution transfers, and has saved the effort of the field inspection activity during a campaign by random verification of TRP's declaration on site.

Task 4 : Sample Integrity

In order to provide more assurance of the Continuity Of Knowledge (COK) of the safeguards sample, IAEA required to install Short Surveillance Video Units (SSVU) to the cell and glove box where the samples were handled for analytical work and to use the Sample Vial Secure Container (SVSC) for the safeguards samples. JAEA installed five SSVUs to the cell and glove boxes. These systems enhanced the COK of the samples at the storage, and enable the inspectors to carry out another inspection activity during the sample treatment. JAEA has also cooperated to use SVSCs for the shipment of IAEA's dried sample to IAEA-SAL. In addition, JAEA carried out the SVSC transfer tests in order to use it for the in-plant pneumatic transfer of the safeguards sample. However, it was confirmed that the current SVSC could not be transferred in plant due to the shape or/and weight.

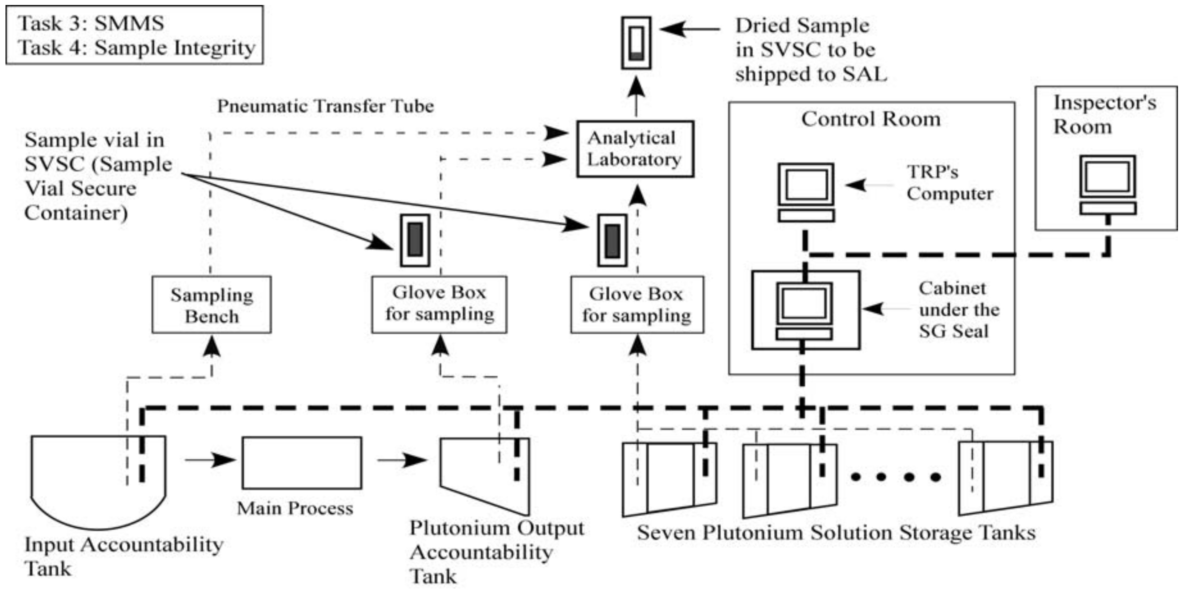

FIG. 2. Concept drawing for Improvement of Reliability of Verification for Primary Tanks.

(3) Establishment of Waste Verification Methods

Task 5 : Verification of Hulls Cask Movements

Leached hulls are transferred with a hulls cask from the spent fuel receipt and storage area to the high level solid waste storage facility. JAEA developed and installed the Hulls Monitoring and Measurement System (HMMS) that is highly accurate and an unattended measurement system in cooperation with Los Alamos National Laboratory (LANL) in 1999 because an attended measurement system had been used before. It detects neutron from the hulls cask to determine Pu amount. Since neutron emission is dominated by Cm, analysis of Pu/Cm in the hulls was carried out. It was found that the amounts of Pu and Cm were very small in the hulls compared with those of little dissolved solution around the hulls that could be dissolved solution, whose Cm/Pu agreed well with those by calculation. Therefore, the calculated Pu/Cm are used to determine Pu amount[2].

Task 6 : Transfer Verification of Transfer of High Active Liquid Waste (HALW)

The HALW is stored in storage tanks after concentration in an evaporator in the TRP main process. The volume of the HALW is measured and samples are taken at the evaporator in order to determine the nuclear material content for accountancy purposes. The HALW stored in the HALW storage tank of TRP is transferred to the HALW receiving tank of TVF for feed to the vitrification process. The volume of the HALW is measured and samples are taken at the receiving tank in order to determine the nuclear material content for accountancy purposes. IAEA required verifying the quantity of nuclear material for the both transfers of HALW in order to satisfy the IAEA’s SG Criteria. JAEA proposed the procedures for the verification of HALW, such as, volume measurement, analysis procedure and shipment of dried sample for IAEA. The process operations are very complicated, so we had many discussions and site tours and we must minimize the effects on the process operations. The procedure of the sample was that dissolution of sludge that contained small amount of Pu. Pu amount was revised to correct an accumulated S/RD.

Task 7 : Verification of Vitrified HALW

In order to measure the nuclear material contents in the vitrified waste prior to the termination of Safeguards, IAEA required to install the temporary measurement system to the hot cell of TVF as the first step and an unattended measurement system as the next step. JAEA cooperated in the installation of the temporary system, and developed and installed the Vitrified Waste Coincidence Counter (VWCC) that is highly accurate and the unattended measurement system in cooperation with LANL. The system is used for the quantitative verification of the nuclear material contents in the canisters and saves the effort of the field inspection activity. It detects neutron from the waste to determine Pu amount in the same way as HMMS, using Pu/Cm data obtained from analytical result of the HALW.

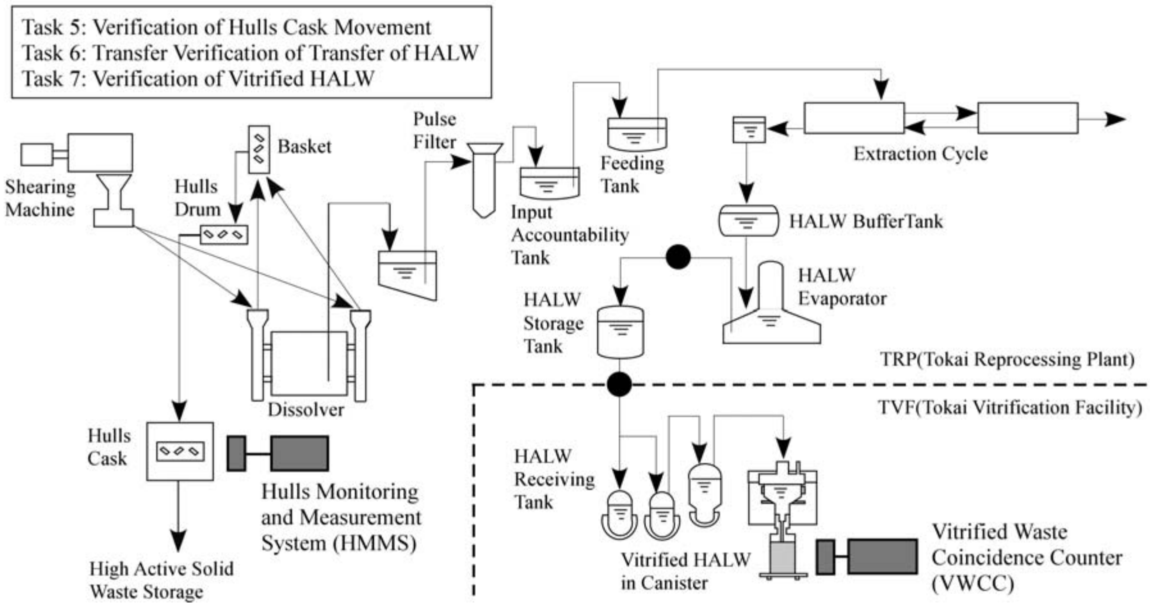

FIG. 3. Concept drawing for Establishment of Waste Verification Methods.

(4) Improvement of Evaluation Methods for Material Accountancy

Task 8 : Shipper/Receiver Difference (S/RD)

An SRD in TRP is the difference between the accounting data of the input accountability tank and that of the power station for a quantity of nuclear material contained in the spent fuel assembly. IAEA required investigating the cause of the accumulated S/RD. The accumulated S/RD had reached about 206 kgPu for twenty-five years. JAEA had tackled to investigate the cause and to improve the accumulated S/RD. As a result, considerable S/RD has decreased by reevaluation of nuclear loss of the stored spent fuel assembly, of nuclear material contained in the sludge of Pulse Filter Rinsing Solution (PFRS) and of nuclear material contained in the reached hull. These made accumulated S/RD reduced by 59kgPu. JAEA corrected the accountancy report under the agreement with the IAEA and the Government of Japan. In addition, inspection activity for PFRS transfer containing small nuclear material quantity that is the inventory change has been implemented.

Task 9 : Near Real Time Material Accountancy (NRTA)

An NRTA scheme is applied to evaluate and verify operator declared data during the monthly IIVs. JAEA declares an inventory data and provides support to the inspectors in such activities as acquisition of the volume data, taking of samples from selected tanks, pretreatment of the sample and shipment to the inspectors' analytical laboratories. In order to make NRTA approach more adequate and complete, IAEA proposed to update measurement errors and to introduce an improved NMAX version of NRTA software in the Improvement Plan. JAEA tackled the following items under the discussion with the IAEA and the Government of Japan.

- JAEA investigated the inventory points for plutonium during operations and shut down and has explained the measurement methods for quantifying plutonium at these points.
- JAEA investigated volume measurement and density analysis methods and the associated uncertainties for the NRTA tanks.
- JAEA reviewed estimate formulas for estimating the quantity of plutonium in the unmeasurable inventory, such as, extractors and pots in each process.
- JAEA proposed an order of sampling of Pu storage process to minimize a variation of a volume of the remaining Pu solution because Pu solution remains in pipes of Pu storage process after sampling activity of each tank.

As a result, the NRTA system has been introduced. JAEA provide the data for NRTA software to enable the IAEA to evaluate the influence of the various measurement uncertainties on MUF and to carry out statistical analysis of sequential MUF.

Task 10 : Provision of Data to IAEA by Diskettes

In order to reduce the time consuming and type error, IAEA required to provide the data related to inventory changes and inventory to inspectorate on diskettes. The data format was discussed and agreed among the IAEA, the Government of Japan and the JAEA. The JAEA has provided the data related to inventory changes and inventory to inspectorate on diskettes. It makes it possible for the IAEA to computerize the data calculation and evaluation, which results in efficient inspection.

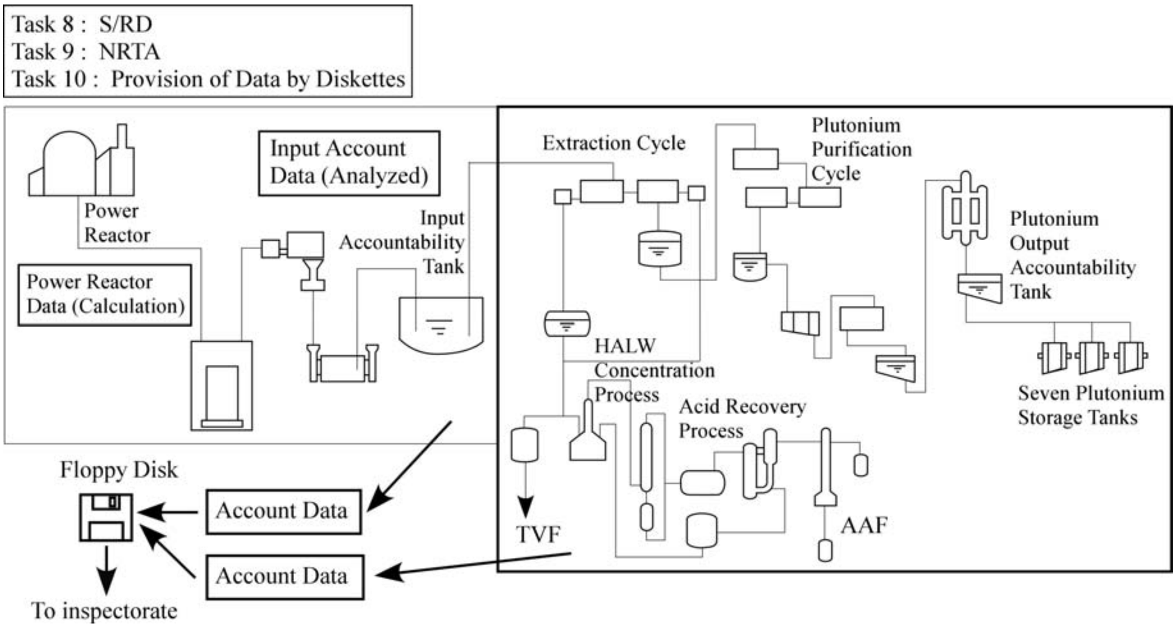

FIG. 4. Concept drawing for Improvement of Evaluation Methods for Material Accountancy.

3. CONCLUSION

The Improvement Plan for TRP safeguards was accomplished and a considerable part of Safeguards Approach has been improved in both efficiency and effectiveness. JAEA provided bird's-eye view figures of buildings, more detailed drawings and supporting information for transparency. Confirmation of the plant operation status for the IAEA's plant tour. The places and indicators are defined as the other strategic points for the confirmation of plant operation status on a FA. The confirmation of plant operation status provides additional assurance of the absence of undeclared design changes and undeclared operations. Three monitoring systems, SMMS, HMMS and VWCC, were installed for transparency and efficiency of inspection activity by unattended mode. More assurance of the Continuity Of Knowledge (COK) of the safeguards sample was obtained by surveillance camera and using the Sample Vial Secure Container (SVSC). Improvements of material accountancy are i) reduction of S/RD by calculation of nuclear loss, ii) measurement of nuclear materials in the hulls and iii) analysis of nuclear materials in the sludge of HALW and update of measurement errors of NRTA. It is expected that these improvements provide a platform for introducing the Integrated Safeguards into TRP.

4. Future challenge

JAEA has been investigating the following points in order to improve the efficiency and effectiveness of safeguards and to improve our accountancy control.

- Introduction of the Integrated Safeguards into TRP
- Transmission of the safeguards system data from the radiation control area to the inspector room
- Development of the measurement system of low radioactive solid waste

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Progress in dismantling of the WAK pilot reprocessing plant: Vitrification of the HLLW

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Abstract. In the past international nuclear safeguards was focused on operating facilities and in these facilities namely on the product streams. The amount of nuclear material in waste was regularly measured or estimated by the plant operator but normally accepted by the IAEA without verification, if certain limits were observed. With the Additional Protocol in force, safeguards measures are applied on (in a technical sense) decommissioned plants and it seems that the conditioning of waste will be safeguarded by the IAEA to a higher degree than in the past. This paper will describe in detail the situation at the Wiederaufarbeitungsanlage Karlsruhe (WAK), where accumulated high level liquid waste (HLLW) will be vitrified in a campaign of 1.5 years. Although technical possibilities of recovering Pu are missing at the site, the IAEA recently asked for installation of a camera and a neutron monitor in addition to former agreed safeguards measures. From the operators' point of view, the effort is not in an appropriate relation to the safeguards significance of the material. Considering that the IAEA is on the way to Integrated Safeguards in the European Union, the Agency may decide to use their limited resources for other safeguards challenges.

1. History

In 1956 the (now) Forschungszentrum Karlsruhe (FZK) was founded in the north of Karlsruhe, Germany. In the beginning the aim of this nuclear research centre was the development of nuclear reactors. In relation to this work also research on reprocessing and waste handling was established. In 1967 the construction of the Wiederaufarbeitungsanlage Karlsruhe (WAK) pilot reprocessing plant started. The aim of this facility was the testing of flow sheet variations and process components developed by institutes of the FZK. Furthermore, also staff training for the planned industrial scale reprocessing plant was necessary. WAK started its hot operation in 1971. During 31 campaigns 207 Mg of uranium and 1.16 Mg of plutonium originating from different German reactors were reprocessed. The average burn-up was approx. 17 GWd/Mg U, the peak value 40 GWd/Mg U. In 1989, the German utilities decided to stop the construction of the industrial scale Wackersdorf reprocessing plant. The result was the final shut down of WAK at the end of 1990. Fig. 1 shows the actual buildings on the WAK site. Furthermore, all the research and development activities concerning reprocessing were stopped at FZK. Today, only waste conditioning and intermediate storage facilities like the Hauptabteilung Dekontaminationsbetriebe (HDB) and the Institut für Nukleare Entsorgung (INE) are in operation. These facilities are necessary for the dismantling of the FZK owned research reactors as well as for WAK.

2. Status of the WAK Dismantling Project

After the final shut down of WAK at the end of 1990 the plant was rinsed and all separated plutonium and uranium were shipped off site. Starting in 1996, the equipment in the process building has been totally dismantled. First 12 systems only having a low activity level could be dismantled manually. In a second step the content of all process cells was dismantled by remote handling (Fig. 2) and also the necessary control systems were removed. Today even most of the pipe penetration blocks between the hot cells are already cut out (Fig. 3). In a first campaign hot spots were removed by abrasive methods. All liquid and solid wastes produced during the dismantling activities were shipped to HDB. The progress in dismantling was regularly verified during the inspections of EURATOM and the IAEA. An overview of the WAK dismantling project has been given in [1], remote dismantling is described in detail in [2].

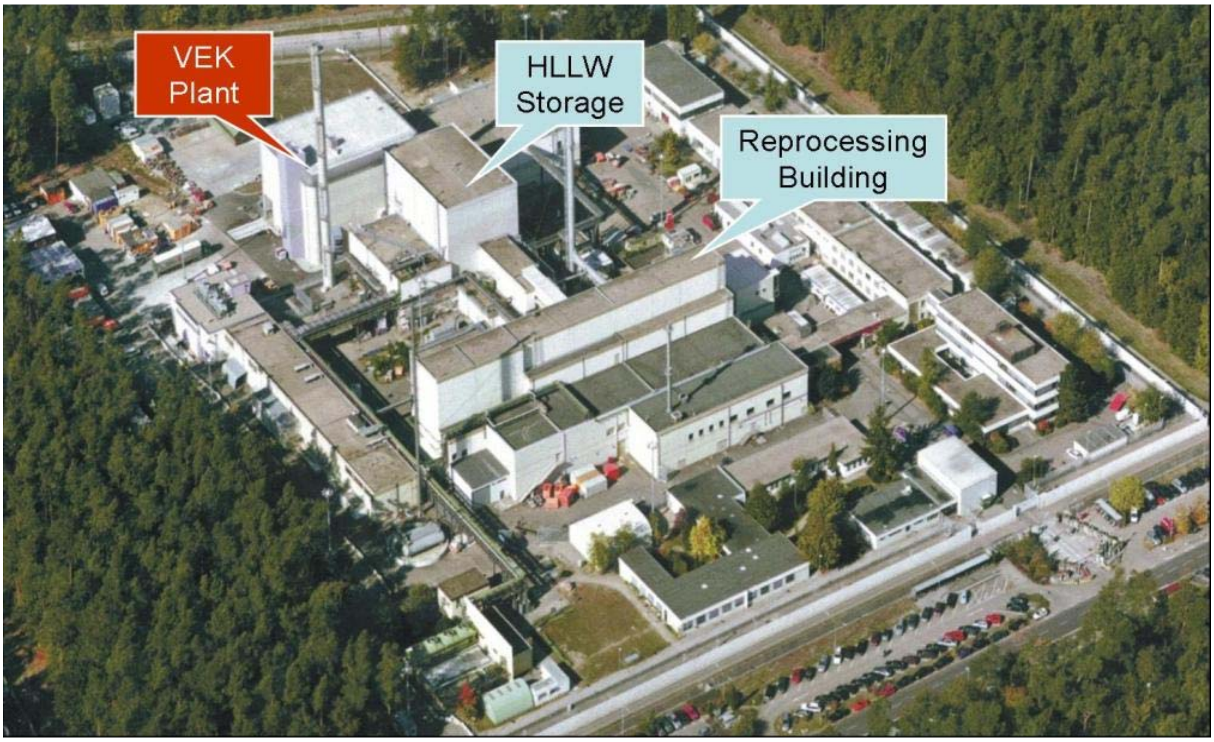

FIG. 1. Aerial view on WAK site.

FIG. 2. Remote dismantling of the HLLW evaporator cell.

FIG. 3. HLLW evaporator cell after removal of the pipe penetration blocks.

Only one part of WAK is still in hot operation: In a separate building (LAVA) approximately 60 m³ of high level liquid waste (HLLW) with a total radioactivity of nearly 8E17 Bq are stored as “retained waste” to be conditioned on site. This vitrification project also has been established in 1996 and a new building for the Verglasungseinrichtung Karlsruhe (VEK) is finished and equipped, now waiting for the license of hot operation in 2007.

3. Design of VEK

3.1. Structure of the building

The vitrification of the HLLW is an essential step for the total dismantling and demolition of WAK. The VEK facility is only planned, constructed and licensed for this specific task that should be terminated within 1.5 years of operation. The equipment of the main process is installed in several hot cells as indicated in Fig. 4, which shows a longitudinal cross section of the VEK building. The HLLW receipt cell (1) contains two receipt tanks as well as the secondary liquid waste treatment. In the melter cell (2) the HLLW feeding vessel, the melter and the first two off-gas components (dust scrubber and condenser) can be found. The two off-gas treatment cells (wet/dry) are located behind the rear wall of

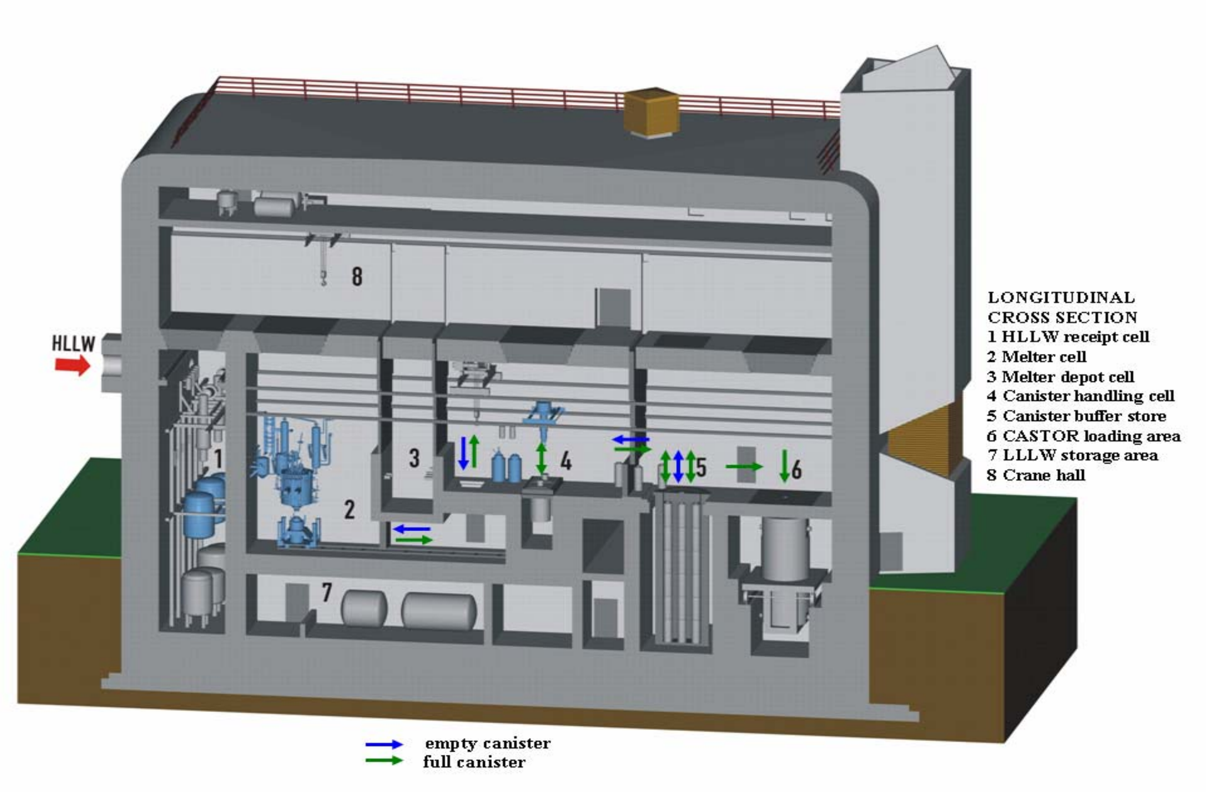

FIG. 4. Longitudinal vertical cross section through the central area of the VEK building including the movement of empty and full glass canisters.

the hot cells and therefore are not visible in Fig. 4. In the unlikely event of a severe melter failure during hot operation, the old melter can be removed remotely and stored in the melter depot cell (3). A second melter was built and is ready for this replacement. In the canister handling cell (4) we find the cooling station, the automatic welding device for the canister lid and the decontamination unit. At the right of Fig. 4 the canister buffer store (5) and the CASTOR loading area (6) can be seen. The LLLW storage area (7) in the basement is used for liquids coming from the off-gas treatment. Heavy components in the cells may be replaced for maintenance reasons by using the equipment of the crane hall (8). The design of VEK has been already described in detail [3, 4].

3.2. *Vitrification Process and Canister Handling*

Looking at the flow sheet of vitrification, it is a straight forward process (Fig. 5): In LAVA a volume of 1.6 m³ HLLW is analyzed and transferred to VEK. In VEK approx. 40 l of ILLW (coming from the wet off-gas treatment) is added and the mixture is analyzed once more to verify the oxide content. The solution is transferred automatically in a small dosage vessel and than poured continuously on the surface of the melted glass. The raw glass itself is dosed batch wise as small pearls of glass frit. The HLLW is dried, calcinated and its chemical elements are incorporated as oxides into the glass matrix.

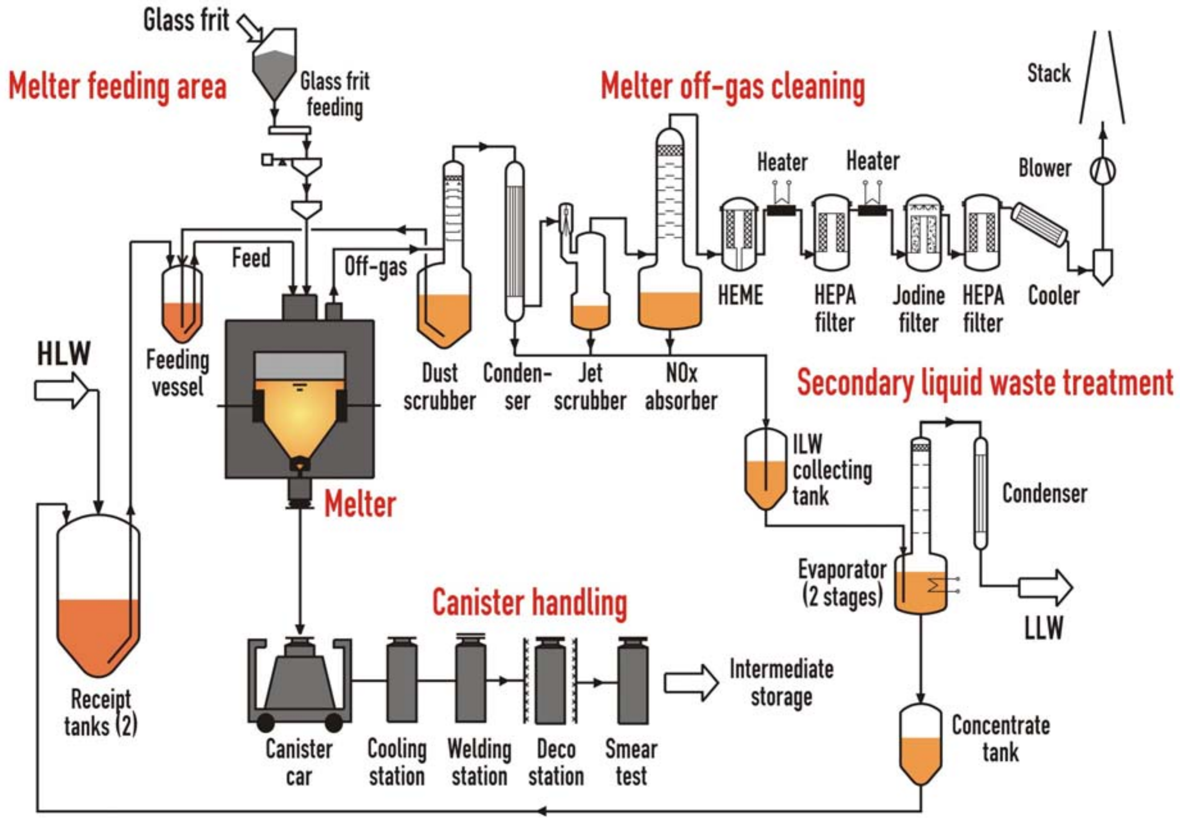

FIG. 5. Simplified flow sheet of the vitrification process.

The melter will contain about 400 kg of glass which will be filled every 15 hours (in four steps of 100 kg) into stainless steel canisters. The canisters are numbered and they will be weighed prior, during and after filling. The canisters will be transferred to the canister handling cell and there cooled down for several days. Then a lid is welded remotely on the canister and the canister is decontaminated. Prior to buffer storage neutron as well as beta and gamma dose rates are measured in the next cell. The buffer storage consists of seven pipes each storing up to six canisters. The centre pipe will only be used for empty canisters so that 36 full canisters can be stored. Each 28 canisters will be loaded to a CASTOR cask. In total 130 canisters will be produced for which five CASTOR casks will be prepared and transported to an intermediate storage facility. In Fig. 4 the way of the full canisters is shown by green arrows, the way of the empty canisters by blue arrows. There are only three differences in handling: empty canisters will not arrive in a CASTOR cask, they will not be measured for activity prior to buffer storage and they will not be treated in the canister handling cell.

4. Safeguards Approaches

Concerning the safeguards relevance of the process and the glass product, the IAEA may accept that a credible diversion path does not exist for the nuclear material. The vitrification process itself has no possibilities of recovering U and Pu and, furthermore, on the sites of FZK and WAK reprocessing capabilities do not exist anymore. The nuclear material content of the HLLW is well known to the IAEA and may be verified once more prior to vitrification. As a consequence, the design verification of VEK, sealing of back transfer lines from VEK to LAVA (which are necessary for safety reasons), and the authentication of the operator measurements on the product canisters should be sufficient for a statement of compliance with the safeguards rules. To reach this goal, a quarterly inspection may be necessary. These safeguards measures have been agreed in 1999.

Keeping the IAEA informed on the VEK project by the WAK annual activity program, we are suddenly faced with the request for additional safeguards measures in early 2006. IAEA asked for the installation of a video camera and a neutron monitor close to the canister buffer store including a recording unit outside the hot cell. This would enable IAEA to monitor the movement of canisters and to distinguish between empty and full canisters. Considering the type of material (conditioned waste) and its safeguards relevance we do not believe that this effort is necessary for the IAEA to meet their internal safeguards citeria. In fact, this late request causes technical as well as licensing problems. Due to the safety requirements, all devices in hot cells of VEK have to be fixed at anchor plates in such a way, that an earthquake or airplane crash does not lead to a damage. The compliance with these safety requirements has to be checked by a control organization and licensed by the relevant authorities. The necessary paper work is time consuming and expensive. As the Additional Protocol is in force in the European Union, we do believe that the IAEA has sufficient qualitative information to verify the absence of clandestine reprocessing in Germany.

In addition to this high effort, we have to consider the failure of camera, neutron monitor or data recording unit. If the data are relevant for a safeguards conclusion we have to look for alternatives. A possible back-up solution could be that the IAEA will get the neutron dose rates of the 28 canisters to be loaded into the next CASTOR cask. Then the IAEA may ask (on a random base) for the observation of a 10% re-measurement during loading. This procedure only needs an authentication of the operators instrumentation and the presence of an inspector. As IAEA inspectors are quite often and regulary in the area and only five CASTOR loadings will occur, the timing of such an inspection should be possible without difficulties. But if this procedure is an alternative, why does the IAEA intend to make such a high effort for safeguarding a waste conditioning plant? Of course, WAK is open for any other safeguards measure that does not modify the licensed status of VEK or will be intrusive during the 1.5 years period of facility operation. Table I shows the different safeguards approaches already discussed.

Table I. Alternatives of safeguarding VEK.

Method	Agreed concept 1999	IAEA request 2006	Back-up solution 2006
DI verification	yes	yes	yes
Containment measures	Blocking and sealing of HLLW flow back from VEK to LAVA	Blocking and sealing of HLLW flow back from VEK to LAVA	Blocking and sealing of HLLW flow back from VEK to LAVA
HLLW analytical input verification	Once in LAVA, at low frequency in VEK	Once in LAVA, at low frequency in VEK	Once in LAVA, at low frequency in VEK
Glass product verification	Receipt of operators' data on dose rates, possibly observation during measurement	Installation of IAEA camera and neutron monitor near canister buffer storage	Receipt of operators' data on dose rates, 10% re-measurement during CASTOR loading
Number of inspections	6 (1 per 3 months)	18 (1/M for data check)	6-9 (In- and output)
Additional costs for IAEA (estimated)	**0**	**> 150000 Euro**	**0**

5. Conclusion

The vitrification of the HLLW resulting from former nuclear fuel reprocessing is an essential step for the complete dismantling and demolition of WAK. The vitrification process has no capability of recovering fissile material. Regarding the dismantling status of the process building, the IAEA should accept that WAK is no longer a reprocessing plant. Furthermore, all research and development activities related to reprocessing have been definitely stopped at the FZK site. As a result, a credible diversion path for the fissile material in the HLLW no longer exists. In the recent request of IAEA to install a camera and neutron monitor these facts were not taken into account and seems therefore not to be appropriate. With the Additional Protocol in force the IAEA gets sufficient information to verify the compliance with the safeguards criteria. As a consequence, the IAEA would be able to spend their limited resources on other safeguards challenges.

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Rokkasho Reprocessing Plant
Moving from safeguards project to safeguards operations

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Abstract. The Rokkasho Reprocessing Plant (RRP), which is currently in the process of active commissioning, is scheduled to begin commercial operation in the fall of 2007. The planned operating throughput of the plant is 800t U/y in irradiated light water reactor (LWR) fuel. This corresponds to approximately 8t Pu/y. The plant is designed to separate the LWR fuel and produce plutonium-uranium mixed oxide (MOX) powder for future processing to reactor fuel, uranium oxide and vitrified waste. The RRP has been an active IAEA task since the early 1990s and is currently transitioning from a safeguards project to a safeguarded facility. The development and implementation of safeguards at the RRP represents one of the largest safeguards endeavours not only in terms of nuclear material under safeguards but also in terms of costs for equipment development, procurement, installation and human resource requirements. The IAEA has been conducting routine inspections at the RRP since the first spent fuel assemblies were received by the facility in 1998. However, when the first LWR assemblies were introduced into the process in March 2006 the IAEA implemented a regime of continuous inspections during operations. This inspector presence ensures that adequate safeguards are applied to the material, provides support for the installation, testing and validation of safeguards equipment, and facilitates development/modification of the procedures necessary for day-to-day operations.

The start-up of the RRP facility has been a stepwise process. Each time the nuclear material reaches a new stage in the process the IAEA is confronted with new challenges. Currently, the plutonium and uranium separation and purification systems are being tested and most of the nuclear material remains in solution. The next phase will be to test the MOX production processes and associated safeguards equipment. This tasks of implementing safeguards and installing, adjusting and evaluating the safeguards equipment and computer support systems have proven to be challenging for the IAEA, the State authorities and the operator. To overcome the obstacles, it is essential that there be a continuous dialogue, meticulous planning of activities and teamwork of all the effected divisions within the IAEA and with the Japanese counterparts. To date, the IAEA has been able to meet these challenges by developing mutually agreeable solutions to problems encountered. This has fostered better understanding of the complex situation and better preparations for the future. To meet current and future challenges as the RRP moves to full operation, all of the parties involved need to be committed to implementing three key components: communication, coordination and cooperation.

1. Introduction

The IAEA has been applying safeguards to reprocessing plants for more than 30 years. However, it has never faced the challenge of implementing a credible safeguards approach for a large commercial scale facility. This challenge is being realized today with the start of active tests at the Rokkasho Reprocessing Plant (RRP) in northern Japan. The plant has an operating throughput of 800 tons per year of uranium; this far exceeds any previous IAEA experience. To meet this challenge the IAEA Department of Safeguards established the JNFL project office in the early 1990s, within Safeguards Operations Division A1. The office has been the focal point for the development and implementation of safeguards for the RRP, which is the largest safeguards endeavour in the IAEA’s history. The responsibilities of the JNFL project office now rest with the Safeguards Operation Division A2 as the RRP facility shifts from a project to routine operations.

2. Facility description

The RRP has an operating annual throughput of 800 tons of uranium in irradiated light water reactor (LWR) fuel. This translates to an output of 8 tons of plutonium per year in the form of plutonium-uranium mixed oxide (MOX) powder. The facility comprises a spent fuel receipt and storage area with a capacity of 12 000 spent fuel assemblies, a head-end, a main process which includes uranium oxide conversion, a U/Pu co-denitration conversion process, MOX and uranium oxide storages, and waste treatment and storage areas. The main process employs a PUREX-type separation process for the removal of fission products and the partitioning and purification of uranium and plutonium. Uranyl nitrate and plutonium nitrate are transferred to the conversion process where they undergo a co-denitration process and are converted into MOX powder. The MOX powder is then transferred to the product storage area. Excess uranium not introduced into MOX is converted into UO_3 powder and transferred to the product storage area. All solid and liquid radioactive wastes are treated and stored in the waste treatment and storage areas. The waste processing area includes the vitrification process for the high active liquid waste.

The facility is divided into five material balance areas (MBAs):

(a) JR1C Cask receipt and storage, spent fuel unloading and storage, and head end process;

(b) JR2C Main process;

(c) JR3C Waste treatment and storage;

(d) JR4C MOX conversion; and

(e) JR5C MOX and UO3 product storage.

3. Transition from a project

The IAEA established the JNFL project office within the Safeguards Operations Division in the mid-1990s to provide a focal point for the planning, coordination and development of safeguards at the RRP. The objectives, which are described in the project plan “SGOA-01 Safeguards Systems for Rokkasho Reprocessing Plant (RRP)”, were to:

(a) Test, refine and obtain approval of a safeguards approach for the transition to routine inspection that would provide effective and adequate safeguards measures and the efficient use of resources.

(b) Test and refine for the transition to routine inspection all inventory and inventory change verification methods and additional assurance measures that would meet the stated safeguards goals of the IAEA. Complete and test all inspection procedures.

(c) Finalize installation and testing and authorize for inspection use all measurement, monitoring and containment and surveillance (C/S) systems so as to assure high quality, independent and reliable results. Complete the transition for all systems to operations as they become authorized.

(d) Finalize the development, installation and testing for the transition to routine inspection all control and collect software associated with the measurement, monitoring and C/S systems so as to assure high quality, independent and reliable results. Complete the process for authorization of all software for inspection use.

(e) Complete the design, development and testing of an integrated data collection and evaluation system which would interrogate all measurement, monitoring and C/S data; calculate as necessary; cross-correlate as specified; carry out diagnostics; and provide results and evaluations to inspectors in a clear, concise format. Complete the process for authorization of the system for inspection use. Where needed, provide 'stand-alone' evaluation software for interim use or as back-up to the integrated sub-systems.

(f) Complete the installation and testing of a sample authentication system which would provide high assurance of sample integrity from the time of sample selection, sample taking and through to analysis. Complete the process for authorization of the system for inspection use and transition to operations for routine use.

(g) Complete the installation and testing of all equipment for the joint operation of the On Site Laboratory. Finalize and test the operating procedures and analytical methods so as to assure that they provide the highest sensitivity while maintaining assurance of independent results. Complete the process for authorization of all equipment for inspection use.

(h) Complete the initial design information examination (DIE)/design information verification (DIV) activities prior to commercial operations. Finalize the DIE/DIV Plan, complete the "DIE/DIV Working Paper for the Operations Phase" and draft the DIE/DIV procedures for the lifetime of the facility so as to assure that the facility continues to be constructed and function as declared and that the safeguards approach continues to be appropriate.

(i) Establish a continuing training programme for IAEA inspectors and support staff on the use and maintenance of equipment, software and specialized methodology and in reprocessing technology. Complete the drafting of an RRP Training Manual.

The number of persons associated with the JNFL project office has varied annually depending on the number of tasks. However, by the end of 2005 the office was staffed with ten professionals and one general service staff member, and additional resources were available to the project through a matrix management system within the Department of Safeguards. The project office was disbanded in early 2006, as the RRP facility approached the active test phase of commissioning and the spent fuel was to be introduced into the process. At that point responsibilities, as outlined above, were transferred to the Safeguards Operations Division A2 which began the task of fully implementing safeguards at the RRP.

4. Facility status

Active commissioning began in March 2006 and is currently in progress. Spent pressurized water reactor (PWR) fuel has been sheared and the primary constituents (Pu,U and fission products) have been separated. The majority of the processed material remains in solution. The next major phase of active testing will be to move the plutonium nitrate solution to the denitration process and convert the material to MOX powder. The processing during the active tests has been slower than anticipated, as the plant works through its normal run-in of the equipment. This extended schedule has given the IAEA additional opportunities to fine tune the safeguards systems and troubleshoot equipment, as necessary.

All of the facility processes, from the receipt of spent fuel through the shearing and separation of purified plutonium, are operating. The IAEA has the capability to remotely monitor all of the major flows from the spent fuel pond to the output accountability tank. However, the full implementation of the automatic processing and analysis of the data have not yet been achieved and the task of retrieving and evaluating the data is handled manually by the shift inspectors. The state-of-health of the data collection systems are monitored daily to ensure that the data flow is continuous. The final safeguards systems are being installed, tested and calibrated in preparation for the receipt of material.

Currently, the application of safeguards in the facility is focused on the first two MBAs where the bulk of the material is located. The IAEA has been working to ensure that the safeguards equipment is functioning reliably and as intended. Work has progressed in the area of streamlining the data processing and ensuring that unattended systems have the appropriate output. Although all of the systems were tested with uranium solutions, this has been the first opportunity to utilize the equipment and systems under near normal operating conditions. Sampling procedures are being perfected and the range of analysis equipment and capabilities demonstrated.

5. Resource requirements

Since the start of the active tests at the RRP in late March 2006, the IAEA has maintained a continuous presence during operations. The shift regime is covered by four inspectors on three eight-hour shifts, as follows:

- Shift 1 01:00 – 09:00
- Shift 2 09:00 – 17:00
- Shift 3 17:00 – 01:00

During shift 2 (day shift) two inspectors are present. One is designated the coordinating inspector and is responsible for formal communication with the operator and State authority, the overall coordination of activities and assignment of responsibilities to the three shift inspectors. The shift schedule allows for adequate turnover and requires the coordinating inspector to personally interact with the shift inspectors.

Execution of inspection related activities at the RRP has resulted in more than 650 person days of inspection (PDI). Additionally the effort dedicated to the On Site Laboratory has resulted in the accumulation of more than 300 PDI to install, test and calibrate the equipment, and perform sample analysis.

The tremendous effort of the Safeguards Divisions of Information Technology and of Technical Services have been key to the installation, calibration, authentication and troubleshooting of the installed safeguards equipment and systems The resources to accomplish this have resulted in over 300 PDI at the facility.

6. Communication, coordination and cooperation

The number of organizations and individuals involved in the development and application of safeguards at the RRP has resulted in a need for a clear framework under which all the activities can be managed. There are two competing objectives in this case: the implementation of an effective safeguards system, and the commercial start-up of the RRP. In order to meet both of these objectives, all of the organizations involved have had to rely on communication, coordination and cooperation.

Within the IAEA, this has called for clear internal communication links to ensure that all of the necessary activities are accomplished in a timely and efficient manner. Additionally, communications with the operator and State authority have been crucial to ensure that the installation of the equipment and computer systems, as well as the development of the necessary procedures, are compatible with the needs of all of the parties involved.

Coordination of the installation and refinement of the equipment and systems have placed a burden on the scheduling activities in order to make sure that the necessary resources are available and that the objectives are accomplished with minimal impact on the other parties. Coordination of the effort within the IAEA and with the Japanese counterparts has been key to accomplishing the required tasks. Full cooperation, which relies heavily on communication and coordination with all of the parties involved, has been instrumental in meeting the overall objectives of the project.

7. Summary

Effective safeguards at the RRP rely on a sophisticated and in many cases a unique set of equipment and measurement devices which has been developed for the facility. This has incorporated reliable, high sensitivity, measurement systems that provide for independent and/or authenticated verification results. The effectiveness is further increased with the use of surveillance, solution and radiation monitoring systems, and continuing DIE/DIV activities, which also provide added assurance that the facility is operating as declared. Verification methods and procedures have been developed for all flow and inventory points within the facility. To the extent possible, unattended measurement and monitoring systems have been developed and installed. Whenever possible these systems are independently controlled by the IAEA. Those measurements, which rely on shared signals from operator-controlled systems, have had authentication measures introduced so as to assure the correctness of the data. Unattended sampling of solutions is accomplished for the majority of the vessels using an automatic sample authentication system (ASAS).

The implementation of safeguards at the RRP has been a tremendous undertaking. Much of the initial design and development work was accomplished by the JNFL project office but as the routine application of safeguards was initiated in the facility the responsibility to complete the tasks moved to the Safeguards Operations Division A2. Successful completion of this important task requires that all of the responsible divisions within the IAEA work together, with clear lines of communication, planned coordination and full cooperation. Moreover, this interaction must encompass all of the other parties involved — namely, the State authority and the operator of the facility. Each of the organizations has its set of priorities and these priorities must be brought together to ensure that all are successfully accomplished.

Safeguards approach for spent fuel transfers to dry storage

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Abstract. The IAEA has recently developed a new safeguards policy and approach for spent fuel transfers to dry storage. Transfers of spent fuel to dry storage, where the fuel is not easily accessible for verification, have become a common occurrence at many nuclear reactors. The IAEA safeguards on such transfers currently consume a considerable amount of inspection effort and are forecast to increase further. Under traditional safeguards, spent fuel transfers have been inspected by 100% inspector presence or by unattended instruments. This paper introduces a new safeguards policy, which was developed for States under integrated safeguards, and describes a safeguards approach for transfers of spent fuel to dry storage at common reactors. The new policy provides for the use of unannounced inspections to confirm operator notifications of spent fuel transfer activities. During transfers to placement in dry storage under permanent containment and surveillance (C/S), continuity of knowledge will be maintained by an unannounced inspection programme or some combination with temporary C/S measures. With the new policy, it will no longer be necessary for IAEA inspectors to be physically present during all spent fuel transfers.

1. Introduction

Transfers of spent fuel to dry storage, where the fuel is not accessible for verification, have become a common occurrence at many nuclear reactors. Traditionally, safeguards on such transfers have absorbed a substantial proportion of inspection effort and are forecast to increase further. An assessment was recently made of how the efficiency of safeguards on these transfers could be increased while maintaining effectiveness.

This approach is based on a new safeguards policy that was developed for transfers of spent fuel to dry storage for States under integrated safeguards.[1] This document introduces the new safeguards policy and describes a model integrated safeguards approach for transfers of spent fuel to dry storage at common reactor types based on unannounced inspections.

2. Traditional safeguards practice, assessment and Member State Support Programmes

2.1. Traditional safeguards practice

When spent fuel is discharged from a reactor to a spent fuel pond, it is verified by IAEA inspectors or by unattended instruments. Under traditional safeguards, permanent surveillance measures are applied to the spent fuel pond area at power reactors to maintain continuity of knowledge of the verified spent fuel. The surveillance is evaluated quarterly to meet the three-month timeliness goal for spent fuel, which is categorized as 'irradiated direct use material' [1]. When the spent fuel is transferred to dry storage, where it becomes difficult to access, the inspectors verify the spent fuel and maintain continuity of knowledge by observing the transfer activities or applying temporary containment and

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[1] Integrated safeguards are implemented in a State with a comprehensive safeguards agreement and an additional protocol in force and for which the IAEA has found no indication of diversion of nuclear material placed under safeguards and no indication of undeclared nuclear material or activities for the State as a whole.

surveillance (C/S) measures until permanent C/S can be applied. In most cases, the spent fuel is then designated as 'difficult-to-access' and dual C/S is applied.[2]

The defect test applied when the fuel is verified depends on whether the fuel assemblies can be dismantled in the storage pond. For fuel assembly types that cannot be easily dismantled (e.g. CANDU bundles and welded light water reactor (LWR) assemblies), a gross defect test is sufficient.[3] For fuel assembly types that can be easily dismantled (e.g. non-welded LWR assemblies), a partial defect test should be applied to protect against pin diversion, using the best available method approved for inspection use.[4] Between the time of final verification and placement in dry storage under C/S, continuity of knowledge should be maintained by inspector presence or C/S measures.

2.2. Assessment and Member State Support Programmes

An assessment was made of how the current safeguards practices on these transfers could be revised while maintaining effectiveness. The Standing Advisory Group on Safeguards Implementation (SAGSI), at the request of the IAEA Director General, reviewed the traditional safeguards requirements for final verification and continuity of knowledge on spent fuel prior to placement in dry storage, and advised that a more flexible approach should be adopted. Specifically, SAGSI advised that where spent fuel has previously been verified and remains under successful C/S, another verification prior to placement into storage should not be required. SAGSI also advised that continuity of knowledge during transfers to dry storage does not necessarily require continuous surveillance (i.e., inspector presence or temporary C/S measures) and can be achieved by applying safeguards measures that provide adequate assurance that diversion does not take place during the transfer, in particular, by performing unannounced inspections.

Several Member State Support Programmes (MSSPs) to IAEA Safeguards were jointly involved in assisting the IAEA in development and testing of the revised safeguards approach for transfers of spent fuel to dry storage at single and multi-unit CANDU reactors.[5] Field trials were successfully performed to test the operator's mailbox arrangements for the use of unannounced inspections, (unattended) non-destructive assay (NDA) measurements systems and implementation of unannounced inspections.[6]

3. New policy on integrated safeguards for spent fuel transfers to dry storage

The new IAEA policy is now for spent fuel previously verified and remaining under IAEA C/S not to be re-verified prior to placement in dry storage; continuity of knowledge can be maintained by an unannounced inspection programme or by some combination of C/S measures with an unannounced inspection programme.

When the spent fuel has been previously verified (the verification requirements would be the same as those specified in traditional requirements) and has remained under successful C/S measures, verification of spent fuel to be loaded into dry storage containers is not required before that spent fuel is transferred to dry storage.

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[2] In a dual C/S system, each plausible diversion path is covered by two C/S devices that are functionally independent and are not subject to a common tampering or failure mode, e.g. two different types of seal, or seals plus surveillance. Dual C/S is normally applied where the verification of nuclear material is difficult to perform, in order to increase confidence in the C/S results and reduce the requirements for periodic reverification.

[3] Gross defect refers to an item or a batch that has been falsified to the maximum extent possible so that all or most of the declared material is missing.

[4] Partial defect refers to an item or a batch that has been falsified to such an extent that some fraction of the declared amount of material is actually present.

[5] Canada, Republic of Korea and the USA jointly participated in the development effort through their individual MSSP.

[6] The mailbox system was designed for State/operator's timely notification to the IAEA on operational activities related to the spent fuel transfers to dry storage, which supports the conduct of unannounced inspections.

Between the time of final verification and placement in dry storage under permanent C/S, continuity of knowledge should be maintained by continuous inspector presence, temporary C/S measures, an unannounced inspection programme, or some combination of those measures.

Unannounced inspections could be implemented at facilities meeting the following conditions:

— The facility operator should provide timely notifications on operational activities related to the spent fuel transfers. The notifications should be submitted using a mailbox system. The notifications should include the information necessary for effective confirmation of the spent fuel transfer operations, e.g. describing spent fuel already transferred and spent fuel to be transferred, with their storage locations and the time sequence of the operational steps involved (including the availability of the spent fuel for the IAEA verification). During unannounced inspections, the facility information notified to the IAEA should be confirmed.

— The inspector access time for unannounced inspections should be shorter than the concealment time for any plausible diversion. The frequency of unannounced inspections should be sufficient to provide the required level of assurance of non-diversion during transfers. The main factors taken into account in establishing inspection conditions and frequency should be the time required for transfer operations, the amount of nuclear material per transfer, information declared by operators and access time to the strategic points.

At facilities that cannot meet the conditions for implementing unannounced inspections, the spent fuel transfer activities should be continuously observed by the IAEA inspectors or by unattended instruments.

4. Safeguards approach for spent fuel transfers to dry storage

4.1. Description of spent fuel transfers to dry storage at common reactors

The major safeguards relevant areas of spent fuel transfers to dry storage at common reactors are wet spent fuel storage, conditioning/processing areas for drying and welding and dry storage.

The wet spent fuel storage area normally consists of several interconnected ponds, which are the discharge pond, reception and defective fuel storage pond, decontamination pond, container loading pond and main storage pond. Spent fuel is transferred between ponds by conveyors or handled manually by fuel handling cranes. In general, the design storage capacity of the main storage pond is to cover about ten years of power operation. After about five to seven years of cooling, spent fuel stored at most reactors is transferred to another storage (e.g. wet or dry storage). For dry storage, the container loaded with spent fuel is welded to form a leak-tight containment for long-term storage. The facilities for the weld conditioning process are usually back-fitted into the existing reactors and vary from facility to facility. The spent fuel transfer occurs normally for a specific transfer campaign period.[7] The frequency of dry storage transfers is dependent upon the annual fuel consumption of the reactor and the storage capacity of the spent fuel pond.

4.2. Potential diversion pathways and safeguards detection measures

For spent fuel transfers to dry storage, potential diversion pathways and safeguards detection measures are listed in Table 1.

[7] For multi-unit CANDU stations, the transfers occur normally over the whole year except during the maintenance period of the facility.

Table 1. Potential diversion pathways and safeguards detection measures.

Potential Diversion Pathways	Safeguards Detection Measures
Spent fuel is removed through the surface of the spent fuel pond, conditioning/processing areas for drying and welding, not associated with any declared fuel movement, using an appropriate container.	Surveillance Unannounced inspection
Spent fuel is removed in dry storage not under the IAEA dual C/S, not associated with any declared fuel movement, using an appropriate container	NDA measurements of the container Unannounced inspection
During a declared transfer, the truck and container are driven to another location and the spent fuel is diverted. A substitute truck and container are driven to the dry storage, where the substitute container is loaded into the dry storage. Some fraction of the spent fuel in the container is removed. This is assumed to take at least several hours.	NDA measurements of the container Seal tamper evaluation Unannounced inspection
Extra spent fuel is loaded into the container and removed from the container. This diversion would require a re-engineered container.	DIV of the container Unannounced inspection
Some of the spent fuel is not loaded into the container. The extra spent fuel is removed from the spent fuel pond after the transfer campaign period.	Surveillance Unannounced inspection
Spent fuel is removed from the container under the IAEA dual C/S application	Seal tamper evaluation NDA measurements of the container Unannounced inspection

4.3 Arrangements for State/operator's timely notifications to support the conduct of unannounced inspections

The State/operator should notify in a timely fashion the operational activities related to the spent fuel transfers. The notifications should include the information necessary for effective confirmation of the spent fuel transfer operations, describing spent fuel already transferred and spent fuel to be transferred, with their storage locations and the time sequence of the operational steps involved. (See Figure 1 for an example of State/operator's notification.) Information that could usefully be notified would include a long-term plan for dry storage transfer campaign, 7-day advance notice of dry storage transfer campaign, 24-hour post notice of dry storage transfer campaign, if required (e.g. when State/operators apply the IAEA electronic sealing system to the containers) and 7-day post notice of dry storage transfer campaign.

Section A: 7-day advance notice of DSC loading

Planned Loading Date: Tue. Jan. 27, 2004
Loading information

Module ID	Number # of Bundles	Bay Grid Location	Position in DSC
XXXX	00	LXX-X	Level 1 (Bottom)
XXXX	00	LXX-X	Level 2
XXXX	00	MXX-X	Level 3
XXXX	00	MXX-X	Level 4 (Top)

Submitter: ________________________ Date of Submission: Tue. Jan. 20, 2004

FIG. 1. An example of State/operator's 7-day advance information notification.

4.4. Arrangements for unannounced inspections

During unannounced inspections, the facility information notified to the IAEA is confirmed. The access time is defined as the time between the arrival of inspectors at a facility and the actual start of inspection activities. The inspector access time for unannounced inspections should be shorter than the concealment time for any plausible diversion from the gate of a facility to the safeguards strategic points (e.g. spent fuel storage building, conditioning/processing areas, dry storage area, transfer route from the spent fuel storage building to the dry storage areas). If the access time is too long, it may be possible for evidence of a diversion to be concealed. Based on the objectives of the unannounced inspections, the IAEA must determine the maximum access time for the inspection to be effective. If the access time achievable in a particular facility is too long, unannounced inspections should not be used.

4.5. Possible options of C/S and NDA systems as complementary measures with unannounced inspections

4.5.1. Surveillance system in the spent fuel pond and conditioning/processing areas

In order to maintain continuity of knowledge after verification of spent fuel in the spent fuel pond, a temporary surveillance system may be installed in the spent fuel pond and conditioning/processing areas at a facility that does not maintain a permanent surveillance system (e.g. LWR spent fuel ponds).

For on-load reactors, the IAEA's ability to re-verify nuclear material in the spent fuel pond is very limited and a surveillance system is maintained in the spent fuel pond in order to maintain continuity of knowledge after continuous monitoring of internal flows from the reactor to the spent fuel pond.

4.5.2. Electronic sealing system applied to the container

Before the container is transferred to dry storage, the State/operator may attach the IAEA electronic sealing system to the container. The State/operator's activities of attaching the electronic sealing system to the container are monitored by the surveillance system (temporarily) installed in the spent fuel pond. Within one working day after the system is attached to the container, the detailed electronic information on the attached sealing system is provided to the IAEA and confirmed on a random basis during unannounced inspections.

4.5.3. NDA measurements on a random basis or with unattended routine use mode

The radiation profile measurements along the length of the loaded container may be performed on a random basis during unannounced inspections to confirm missing nuclear material from the containers. An NDA system with unattended routine use mode may also be used to monitor nuclear material loading and to detect removal of nuclear material from the containers.

4.5.4. Inspection activities during unannounced inspections

The State/operator's operational activities notified to the IAEA will be confirmed on a random basis during unannounced inspections. Major inspection activities to be performed at each safeguards strategic point are shown in Table 2.

Table 2. Unannounced inspection activities at safeguards strategic points.

Safeguards strategic points	Unannounced inspection activities
Spent fuel pond and conditioning/processing areas	Verification of spent fuel (to be) loaded into a container NDA measurements of the container, if any Evaluation of C/S and NDA records Visual check of the spent fuel pond, conditioning areas and containers Review of source documents (e.g. SSAC/operator's spent fuel loading records)
Dry storage	NDA measurements of the container, if any Evaluation of C/S and NDA records Visual check of dry storage areas and containers Application of C/S Review of source documents (e.g. SSAC/operator's spent fuel loading records)
Route from the spent fuel pond to dry storage	Evaluation of C/S and NDA records Visual check of containers

5. Conclusion

The changes in the safeguards approach for spent fuel transfers to dry storage will no longer require IAEA inspectors to be physically present during all spent fuel transfers. The new policy and safeguards approach with unpredictability of unannounced inspections will considerably improve effectiveness and efficiency of the current IAEA safeguards system. The IAEA has estimated more than a 50% decrease of inspection effort for spent fuel transfers to dry storage through the use of unannounced inspections.

The new policy and safeguards approach are applied for States under integrated safeguards with the added safeguards assurance provided by measures of the additional protocol [2]. The broadening of available verification measures would require greater adaptability at the implementation level. More options would be available to IAEA inspectors and there would be less emphasis on routine inspection activities. The new approach has been already incorporated into the approved State-level integrated safeguards approach for Canada, soon to be implemented.

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Implementation of an integrated safeguards approach for transfers of spent fuel to dry storage at multi-unit CANDU generating stations

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Abstract. The Canadian Nuclear Safety Commission (CNSC) in 2002 undertook an IAEA Support Program task to develop and test an Integrated Safeguards (IS) approach for transfers of spent fuel to dry storage at multi-unit CANDU stations. The IS approach that Canada proposed in 2003 was successfully field tested in Canada by the IAEA in 2004. Since then the IAEA has drawn a positive conclusion for Canada and is preparing to implement IS in Canada based on the State-level Integrated Safeguards Approach (SLA) for Canada that was concluded in November 2005. Implementation of an IS approach for transfers to dry storage, similar to the one proposed by Canada, will be part of the initial phase of the implementation of the SLA. This paper briefly reviews the IS approach for transfers proposed by Canada and explains how the approach may be modified for implementation as part of the SLA to take into account lessons learned from the field test, operator and IAEA concerns, and developments in technology.

1. Introduction

Significant Agency inspector effort is used to safeguard transfers of spent reactor fuel to dry storage. Much of that effort is expended in Canada particularly at the multi-unit CANDU reactor stations. In recent years such transfers have consumed a substantial portion of the Agency inspection effort and will absorb much more going forward as the number of transfers increase.

As part of a Member State Safeguards Support Programme task to help the Agency find an effective and more efficient approach, Canada, in 2003, proposed a conceptual IS approach for transfers to dry storage that would, if implemented, result in substantial inspection resource savings for the IAEA at the multi-unit stations without undermining safeguards effectiveness. In 2004 the IAEA, Ontario Power Generation (OPG) and the Canadian Nuclear Safety Commission (CNSC) field tested the proposed IS approach at the Pickering Nuclear Generating Station and the Pickering Used Fuel Dry Storage Facility.

In 2005 the IAEA reached, for the first time, the broader safeguards conclusion on the non-diversion of declared nuclear material and the absence of undeclared nuclear material and activities in Canada and also concluded the “State-level Integrated Safeguards Approach for Canada” (SLA).

The IAEA is poised to begin phased implemention of IS in Canada based on the SLA. An essential step before implementation of the SLA is the preparation of procedures for short notice random inspections (SNRI) and, as appropriate, unannounced inspections (UI) for each of the sectors of the Canadian nuclear fuel cycle. Due to the large size and complexity of the Canadian fuel cycle, it was agreed with the IAEA that the implementation of the SLA would be phased in by sector. The IS approach for transfers of spent fuel to dry storage at the multi-unit CANDU stations will be implemented in the initial phase.

Canada’s approach to safeguards is to maintain an open, transparent and cooperative process with the IAEA and the Canadian industry, on all important safeguards matters. From experience we have seen the benefits of early and frequent consultation with the Agency and the industry on safeguards implementation matters. At times we have found it useful to form dedicated trilateral (CNSC, operator and IAEA) working groups to develop and carry out important aspects of safeguards such as field trials.

The same co-operative trilateral (CNSC, operator and IAEA) approach that was so successful in carrying out the field trial is being now being used to refine the IS approach for transfers and to develop the site/facility specific arrangements and procedures necessary for implementation.

This paper briefly describes the transfer process at the multi-unit stations, the current safeguards approach, and the conceptual Canadian proposal. The steps currently being taken by the IAEA and Canada to implement a refined version of the Canadian conceptual proposal as reflected in the procedure for "Unannounced Inspections for Spent Fuel Transfers at Multi-unit CANDU Stations in Canada under Integrated Safeguards" are also discussed.

2. The transfer process at the multi-unit stations in Canada

In the spent fuel bay (SFB) four modules, each containing 96 CANDU spent fuel bundles are loaded underwater into a concrete Dry Storage Container (DSC). After the DSC is filled with the 384 spent fuel bundles it is transferred from the SFB to the Processing Area (PA) of the on-site Dry Storage Facility (DSF). In the PA the lid of the DSC is welded to the body before the DSC is put into the Storage Area (SA) of the DSF for interim storage.

3. The current safeguards approach

Agency installed safeguards equipment verifies spent fuel as it moves from the reactor core and to the SFB. The entire SFB is under continuous IAEA camera surveillance.

The IAEA inspector re-verifies the spent fuel in the modules before it is loaded into a DSC and escorts the loaded DSC to the PA of the DSF. Soon after the DSC arrives in the PA, the inspector takes two passive gamma radiation signatures (gamma profiles) known as "fingerprints" of the DSC. After the operator welds the lid on the DSC, dual containment and surveillance (Dual C/S) is applied to the DSC by the application of two types of IAEA seals in the PA before the DSC is put into interim storage in the SA.

Continuity of knowledge (CoK) of the contents of the filled DSC is maintained by IAEA camera surveillance in the SFB area, by human surveillance during its transfer to the DSF, by camera surveillance in the PA, and by dual C/S in the SA which is not under IAEA camera surveillance.

Fingerprints taken by the IAEA of the DSCs soon after their arrival in the PA establishes two unique signatures (gamma profiles) for each DSC. Subsequent confirmation fingerprints can provide assurance that modules loaded with spent fuel bundles have not been removed from the DSC in the event of Dual C/S system failure.

The current approach requires considerable IAEA inspector presence. An IAEA inspector must monitor each transfer and each major stage of the process. Each transfer requires 4 PDIs of IAEA inspector effort. For every transfer of a DSC the IAEA inspector has to:

(1) Verify the spent fuel before it is loaded into the DSC
(1) Witness the loading of the spent fuel into the DSC
(2) Escort the transfer of the loaded DSC from the SFB area to the DSF

These three activities take 3 person days of inspection (PDIs) per DSC transfer. Another PDI is used for application, verification and replacement of seals and other inspection activities.

4. Brief overview of the conceptual integrated safeguards approach for transfers to dry storage proposed by Canada

The proposed conceptual IS approach is based on the provision of information on operational activities to support an unannounced inspection regime. The Canadian proposal is designed to reduce the requirement for IAEA inspector presence for every DSC transfer to presence only during random unannounced inspections. The number of times that an IAEA inspector is required to be present at the station is reduced from 100% of the DSC transfers to the frequency of random UI.

According to the Canadian proposal, DSC transfers would be randomly selected for unannounced inspection based on the continuously updated postings by the operator of transfer activities to a safeguards mailbox. Every DSC would be eligible for selection but only some would be randomly selected for UI verification. For each randomly selected transfer, UI activities would be performed at one or more of the three access points between the spent fuel bay where modules may be verified prior to loading, through the transfer of the DSC to the processing area of the dry storage facility.

The proposed approach takes into account certain characteristics of the safeguards program already in place at the multi-unit stations such as the existing installed IAEA safeguards equipment and State-level factors, particularly the presence of the IAEA Regional Office in Toronto. This latter consideration facilitates UIs in Canada.

The major elements of the proposed approach are:

(1) **A Safeguards Mailbox** consisting of provision of information by the operator of operational activities. The advance schedule of upcoming DSC loadings and transfers to the DSF are provided by the operator to the mailbox. The information is updated whenever the plans change;
(2) **Operator activities** which include the taking of two fingerprints of each loaded DSC in the SFB area under IAEA camera surveillance before the DSC is transferred to the DSF; and
(3) **Unannounced inspections** by the IAEA at the three strategic access points of the transfer process (the SFB, the transfer route, and the DSF). Various random UI activities may be undertaken in order to confirm the operator’s information of activities and to ensure the absence of undeclared nuclear material or activities at the facilities.

Some of the other important aspects of the proposed approach are:

— During random UI of the SFB, the IAEA inspector could, as part of the list of UI activities at that access point, item count (to confirm the number of bundles) and take spot-check NDA measurements of bundles in the modules (to confirm for spent fuel) waiting to be loaded into the DSC. However, the verification would no longer have to done for each DSC transfer; and
— The inspector does not escort the loaded DSC to the DSF. However, an inspector may monitor the transfer during random UIs, as part of the list of UI activities at that access point.

The current safeguards approach for transfers requires substantial IAEA inspector effort because every transfer and every strategic point of the transfer process requires the presence of an IAEA inspector. The proposed IS approach with its use of random UIs introduces the powerful element of unpredictability into the inspection process and the possibility of significant savings in IAEA inspector effort. The IAEA can perform verification/inspection activities on a randomly selected sample of the transfers and at randomly selected access points of the transfer process. As long as the operator cannot predict the arrival of inspectors, the access point or points selected and even the exact activity, effort can be significantly reduced without undermining effectiveness.

With the proposed approach, three out of the four PDIs currently expended would be eliminated by not needing an inspector to 1) verify the four modules before loading them into the DSC 2) witness the loading of the DSC and 3) escort the loaded DSC to the DSF. Further reductions in effort may be achieved with the development of remote monitoring, enhanced fingerprinting techniques, and new technologies such as next-generation seals.

The taking of fingerprints of the DSC by the operator early in the process would give the IAEA the capability to take confirmation fingerprints of randomly selected DSCs to re-establish CoK of the contents at any time later, making it unnecessary for the IAEA inspectors to escort the DSC to the DSF. With the removal of scheduled inspections, the operator would have greater flexibility in scheduling transfers and transfer operations because inspectors would only interrupt the operation during a few random UIs.

All of these activities could be performed during random UIs that provide 100% coverage with less effort for the IAEA and less interruption to the facilities' operation than the current approach. Another advantage of the proposal is that it does not depend upon the development, installation, and on-going maintenance of new equipment which would involve a considerable amount of time and cost for the IAEA, the State, and the operator.

5. The field trial

To determine the feasibility of the Canadian conceptual proposal and to uncover any implementation issues, it was decided to conduct a field test of the Canadian proposal at a multi-unit station in Canada. The CNSC felt that the early and continued involvement of the multi-unit operators and the IAEA in the trial was essential. The IAEA and the multi-unit operators agreed. A trilateral IAEA, CNSC and operator working group prepared for the field trial that was carried out in the spring 2004.

In the lead-up to the field trial, operating procedures were drafted for each major element of the proposal approach.

For the mailbox, it was necessary to:

— define the information to be posted by the operator;
— determine the timing of those postings;
— determine the security classification of the Mailbox information; and
— establish and test the mailbox system.

For fingerprinting, it was necessary to teach OPG operators how to take fingerprints of a DSC in the SFB area prior to its transfer using IAEA equipment.

For UIs, it was necessary to:

— define maximum access times to the three strategic access points;
— develop a menu of UI activities that could be performed by the IAEA at each strategic access point during normal- and after-working hours; and
— decide on the measures that could be taken to freeze the situation in the event that the inspector could not reach the strategic point within the allotted time or if certain UI activities could not be performed due to unavailability of equipment or lack of operator support etc. (Force Majeure).

In April and May 2004, the field trial was successfully carried out. It demonstrated the feasibility of the IS approach and identified certain implementation issues. For example, the operator was concerned with the regulatory risk associated with taking fingerprints on behalf of the IAEA in the event that fingerprints were not taken correctly or the operator damaged the Agency's equipment.

6. Preparing to implement the IS approach for transfer to dry storage at the multi-unit Stations

After the successful completion of the field trial, in September 2005, the IAEA reached a broader safeguards conclusion for Canada. In November 2005, the IAEA finalized the IS state-level approach for Canada. Furthermore, in 2006, the IAEA developed a policy for spent fuel transfers to dry storage. The policy anulled the requirement that spent fuel be re-verified before going into difficult to access storage and affirmed that continuity of knowledge of spent fuel during transfers to dry storage could be maintained by an unannounced inspection programme. As a consequence, the basis for implementing an IS approach for transfers of spent fuel to dry storage at multi-unit CANDU stations in Canada has been established.

It was agreed with the IAEA that the implementation of the SLA would be phased in by industry sector and the implementation of an IS approach for transfers to dry storage at the multi-unit stations, similar to the one proposed by Canada, would be part of the initial phase of the implementation of the SLA.

Canada is aware that early and sustained involvement of the industry is important for successful development and implementation of safeguards. For that reason within two months of the conclusion of the SLA, in February 2006, the CNSC met with the OPG and Bruce Power (BP) multi-unit station operators to inform them that an IS approach for spent fuel transfers would be part of the first phase of IS implementation. The operators and the CNSC felt that the IAEA/CNSC/operator trilateral approach that proved so successful for the field trial should also be used to collaborate on implementation matters, such as the development of site/facility specific procedures for unannounced inspections, the work plan, and the refinement of the IS approach.

In March 2006, a trilateral working group was struck to co-operatively refine the details of the IS approach and develop the procedure for “Unannounced Inspections for Spent Fuel Transfers at Multi-unit CANDU Stations in Canada under Integrated Safeguards” that would be included in an Annex of the SLA for Canada. The procedure would prescribe the main elements of the IS approach for transfers of spent fuel to dry storage at the CANDU multi-unit stations.

In June 2006, at the first meetings of the trilateral working group, the core membership was agreed. It was further agreed that sub-groups should focus on selective aspects of the implementation such as the provision of information, security, and the development of a work plan. The group also agreed that there should be trial runs to test access and establish realistic access times for UIs. In August 2006, the first of four access trials for IS implementation took place at OPG’s, Pickering facilities.

Although the field trial demonstrated the feasibility of the 2003 Canadian conceptual proposal, the IAEA determined that certain changes to the approach should be made. The most significant is with regard to operator activity. In the Canadian proposal the operator would fingerprint each DSC prior to its transfer to the DSF. In the current IAEA proposal the operator would not fingerprint DSCs. Instead, the IAEA could fingerprint DSCs on a random basis during UIs. Other filled DSCs could be fingerprinted and sealed by the IAEA in batches during announced sealing campaigns. Relieving the operator of this activity eliminates the operator’s concern with the regulatory risk associated with doing NDA measurements on behalf of the Agency.

However, the fingerprinting and sealing of DSCs in batches by the IAEA, instead of one at a time by the operator, could create implementation problems for the operator. For example, the PA’s of the DSFs have only enough room to store a few fully processed DSCs. It may be necessary, in order to prevent congestion in the PA that would interfere with the safe and efficient operation of the PA, to create a dedicated lay-down area in the SA where many DSCs, waiting to be fingerprinted and sealed during the next IAEA sealing campaign, could be kept under IAEA camera surveillance.

At this writing, the IAEA procedure for “Unannounced Inspections for Spent Fuel Transfers at Multi-unit CANDU Stations in Canada under Integrated Safeguards” is still being developed and the trilateral working group is continuing to meet on a regular basis to review it and discuss implementation. The IAEA's procedure may undergo further modification before it is finally approved at the Agency.

Each new version of the procedure or new ideas suggested by the IAEA at the trilateral meetings will be carefully considered by the CNSC and the industry members of the working group. Industry representatives will continue to provide the IAEA and CNSC with practical input regarding the feasibility and the effectiveness, and the practical difficulties of proposed measures. In the end, the trilateral process will improve the IS approach, as it did with the field trial project, and tailor it for effective and efficient implementation at each multi-unit CANDU station in Canada.

It is recognized that that the IS approach is evolutionary and could change as implementation experience is gained and in accordance with developments in safeguards. Developments in technology such as improved spent fuel verification techniques or instruments could affect the approach. With improved fingerprinting equipment and next-generation remotely readable seals and other technologies, there may be ways to further reduce the inspection effort without reducing safeguards effectiveness.

7. Conclusion

Under traditional requirements for safeguards, transfers of spent fuel to dry storage require substantial IAEA inspection effort. An unannounced inspections regime under integrated safeguards can drastically reduce the resources required to verify such transfers without undermining effectiveness.

Such reductions will occur in Canada when the procedure for "Unannounced Inspections for Spent Fuel Transfers at Multi-unit CANDU Stations in Canada under Integrated Safeguards" as part of the State-level Integrated Safeguards Approach for Canada is concluded and an unannounced inspection regime for transfers at multi-unit stations is implemented.

The procedures that will lead to reductions in effort are being developed by the IAEA in a context of open and transparent trilateral discussions between the IAEA, CNSC and the operators to ensure that the Agency has the information and co-operation it needs to make the procedure effective, efficient and timely.

Safeguards for spent nuclear fuel in transfer from wet storage to dry storage in on-site interim storage facilities

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Abstract. Germany initially planned to store spent nuclear fuel in the two away-from-reactor interim storage facilities built at Ahaus and Gorleben. The current approach for spent fuel management is on-site interim storage in transport and storage casks as part of a political agreement between the German government and the nuclear operators on the future use of nuclear energy. A reason for this is to avoid near term transportation of spent fuel determined for direct final disposal via public road or rail systems to away-from-reactor storage facilities. Recent legislation has triggered the construction of 12 on-site dry storage facilities at nuclear power plants. Currently, such facilities are being taken into operation on a step-by-step basis. There is a strong need to develop acceptable safeguards concepts for both transfer and dry storage of spent fuel, ideally, a standard safeguards concept that would match all German nuclear power plant sites without ignoring technical and organisational differences. The paper will address the relevant issues and give an overview of the status of safeguards implementation.

1. INTRODUCTION

Germany initially planned to store spent nuclear fuel in the two away-from-reactor interim storage facilities built at Ahaus and Gorleben. The current approach for spent fuel management is on-site interim storage in transport and storage casks as part of a political agreement between the German government and the nuclear operators on the future use of nuclear energy. A reason for this is to avoid near term transportation of spent fuel determined for direct final disposal via public road or rail systems to away-from-reactor storage facilities. Recent legislation has triggered the construction of on-site dry storage facilities at 12 nuclear power plants. Currently, such facilities are being taken into operation on a step-by-step basis (December 2002 and between February 2006 and April 2007).

In view of the considerable number of new facilities there is a strong need to develop acceptable safeguards concepts for both transfer and dry storage of spent fuel, ideally, a standard safeguards concept that would match all German nuclear power plant sites without ignoring technical and organisational differences.

The paper will address the relevant issues and give an overview of the status of safeguards implementation.

Figure 1. On-site interim dry storage facility at Emsland NPP, Lingen/Germany [1].

Figure 2. Emsland NPP with interim storage facility [2].

2. TECHNICAL CONCEPT OF ON-SITE INTERIM STORAGE

In principle, an on-site dry storage facility consists of three parts: reception area, maintenance area, and storage area. Altogether, 12 licenses were issued for the construction of on-site interim storage facilities at nuclear power plant sites. The first license was received by the Emsland NPP located at Lingen in Lower Saxony, Northern Germany (see Figures 1 and 2). This interim storage facility will be briefly described as an example. It was taken into operation in December 2002 and has the following features.

The facility consists of two buildings: storage building with storage area and reception area for spent fuel casks; control building in which plant operation is controlled. The permitted storage period is limited to 40 years beginning with the emplacement of the first spent fuel cask in the storage building. There are 130 cask positions, five of which being reserved for empty casks only. The Lingen interim storage facility has a length of about 110m, a width of about 30m, and a height of about 20m. The wall thickness is about 1.2m, while the monolithic roof is about 1.3m thick [3]. The floor is made of concrete armoured with steel.

In the reactor containment, spent fuel will be loaded into shielding casks, e.g., of the CASTOR®-type, and transferred out of the reactor building into the associated on-site dry storage facility (see Figure 3).

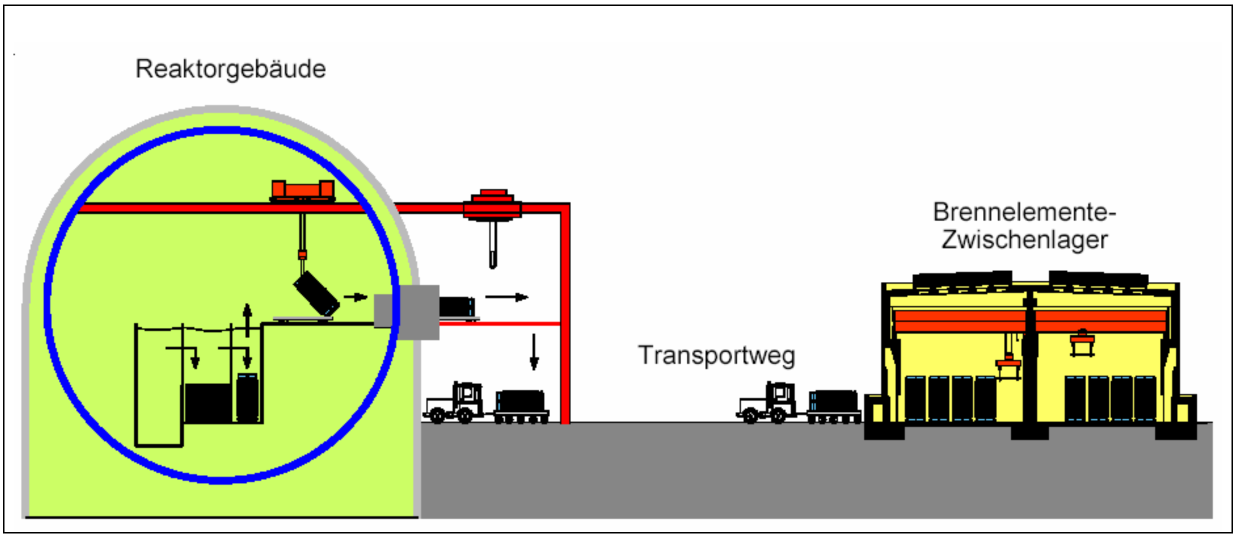

Figure 3. Transfer of spent fuel casks from the reactor building to the on-site interim storage facility [4].

3. SAFEGUARDS ASPECTS AND REQUIREMENTS

International Atomic Energy Agency (IAEA), Euratom, and German plant operators will be interested in keeping the safeguards concepts as simple as possible, while there should be consistency between all the on-site interim storage facilities. Although it is desirable to harmonise and standardise the safeguards concepts for all on-site interim storage facilities, there may be technical and organisational differences in the individual facilities that will have to be taken into account. From the State's point of view the safeguards concepts have to comply with requirements related to operational safety, radiation protection, and physical protection. Furthermore, they have to take into account the political and technical boundary conditions as well as the time schedule for spent fuel transfers coordinated between all the nuclear power plant operators in Germany.

During the first three years, the number of casks will be quite high that will be transferred out of reactor containments into on-site storage facilities. According to the current schedule, this will be a total of 49 CASTOR®-type transport and storage casks in 2006; in 2007 the number will be 64, and in 2008 it will be 71. Subsequently, the number of transfers per year will reach an equilibrium of up to 45 casks per year, so that the storage facilities will acquire basically a character with very infrequent operators' transfer activities. At each power plant site the frequency is estimated to amount to 6 casks/two years. The plant operators will issue reliable time schedules on a semi-annual basis and provide the confirmed information to Euratom.

For reasons of keeping persons' exposure to radiation as low as reasonably achievable (ALARA principle) the storage area will not be suited for frequent access. This should be taken into account when designing an adequate safeguards concept for a 40-years spent fuel storage. In the storage facilities, the inspection effort should be as low as possible, in order to minimise accidental risk and radiation exposure of both the plant operators' staff and the inspectors.

4. SAFEGUARDS APPROACH

A safeguards approach for the transfer of spent fuel out of the containments of nuclear power plant reactors and its dry storage in on-site facilities has not yet been negotiated between the IAEA, Euratom, and the plant operators. This type of transport is different from the transport of spent fuel to the away-from-reactor interim storage facilities, since there are no public transport roads involved. Therefore, the transport package of the spent fuel casks is different, and this may also affect the application of seals. The intention is to develop acceptable safeguards concepts, ideally, a standard safeguards concept that would match all German nuclear power plant sites without ignoring individual technical and organisational differences.

An on-site dry storage facility will be a material balance area (MBA) of its own, so that, in the future, each power plant site will have at least two MBAs. For each MBA the rules of nuclear material accounting apply.

The basic technical characteristics (design information) of the layout and operating characteristics of the on-site storage facilities have to be reported to the Euratom Safeguards Office which in turn will re-transfer this information to the IAEA. Furthermore, plant operators have to announce to Euratom, in advance, dates and times of the CASTOR® loadings.

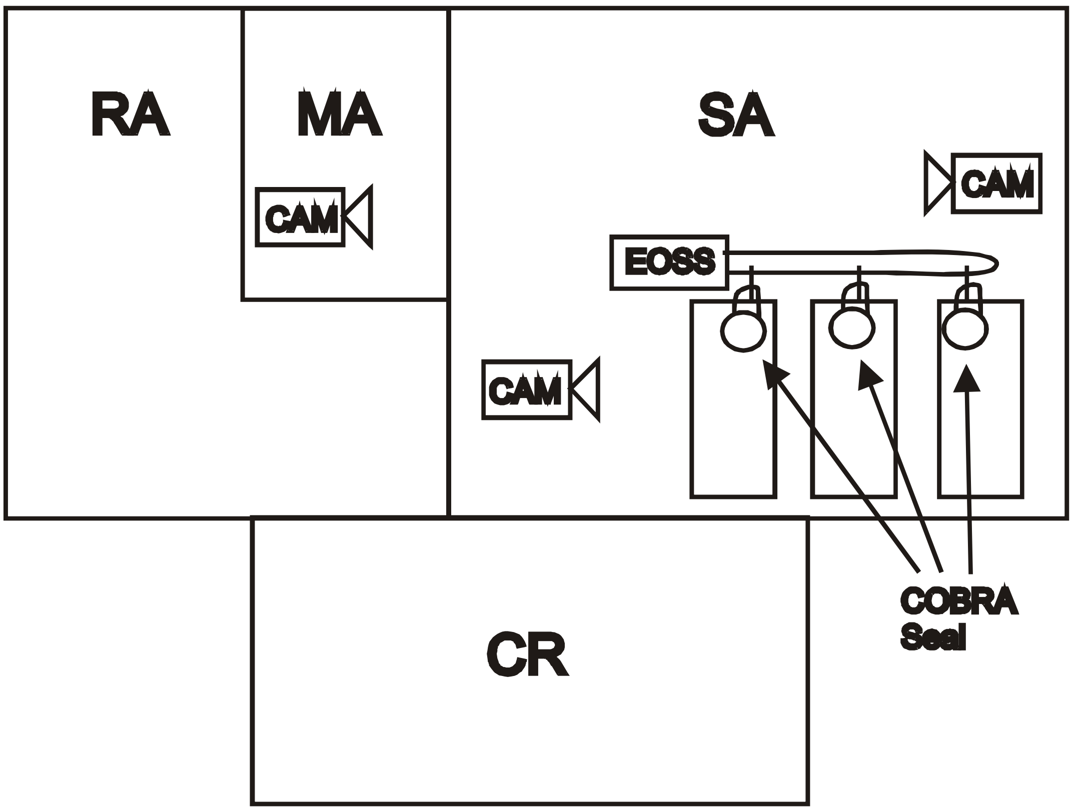

Figure 4. On-site Storage Facility
RA reception area; MA maintenance area; SA storage area; CR control room.

For the potential safeguards concepts sealing, optical surveillance, and radiation monitoring are discussed. Possible types of seal are metal cap seal, COBRA fibre optic seal, and EOSS electronic seal. Possible digital image surveillance systems are the DCM14-based ALIS and DSOS systems. The feasibility of their application should be proven in view of a facility specific safeguards instrumentation concept. Furthermore, installations of cables may become necessary, and the installed safeguards instrumentation has to comply with national safety requirements. In order to keep the inspection effort as low as possible, the potential for reduction should be analysed. For instance, interfacing of electronic sealing and digital image surveillance offers the possibility for the plant operator to apply or remove a seal under surveillance in the absence of the safeguards inspector. Finally, the aspect of remote data transmission (RDT) out of on-site interim storage facilities should be taken into account. Due to its construction concept, this type of facility may lend itself very much for RDT. Therefore, the potential of RDT should be studied on a case-by-case basis. The German Administration has to define the boundary conditions for remote transmission of safeguards data out of Germany.

In the storage area one or two cameras of the DSOS-type could be applied. In the maintenance area one camera could be operated, e.g., ALIS in overwrite mode. In the reception area it does not appear to be necessary to use a camera. Also, there should be no need for neutron monitoring. In the storage area, in addition to digital image surveillance, there could be applied electronic seals of the EOSS-type with remote interrogation capability (i.e., interrogation of the seal from outside the storage hall via RS485 interface). One such seal could be used to secure a group of casks, in order to counter the scenario of undeclared removal of a cask and to limit the number of seals. For reasons of redundancy, each cask could also be sealed with a COBRA-type fibre optic seal. As a backup measure there could be a metal cap seal on each cask. However, the current problem with the metal cap seal is that the IAEA attributes an applicability of three years, after which the seal has to be replaced by the inspector.

Another issue is the meaningfulness of measuring the spent fuel with an Ion Fork device upon its being loaded into the casks. In addition, such measurements put a burden on the plant operator regarding coordination and escorting of inspectors. Current IAEA safeguards criteria require that, prior to being transferred to dry storage, a partial defect test should be applied to LWR fuel that can be easily dismantled. The use of the Fork Detector is approved for that purpose. For spent fuel assembly types that cannot be easily dismantled a gross defect test is sufficient.

Regarding the circumstances to be taken into account here, there is no justification for these different requirements. The exchange of fuel pins in a fuel assembly is quite a rare event in German power reactors and requires complex handling procedures that can be observed by the according surveillance camera [5]. The same argument holds for the removal of diverted irradiated fuel pins from the pond. The removal of a couple of pins in one action would require a heavy shielding cask which is also easy to detect by surveillance. This scenario is, in fact, not different from the scenario of removing a whole spent fuel assembly. Removing single pins may require less bulky shielding than for spent fuel assemblies, but this action had to be repeated many times, in order to acquire a noteworthy amount of nuclear material and, thereby, increase the risk of detection. Cutting a pin into pieces to better conceal the removal is not regarded a plausible diversion scenario, as there are no hot cells associated with the reactors.

The IAEA requirement is to apply the best available method to verify these assemblies, before they become difficult-to-access in the dry storage facilities. Studies performed during the last years under Member State Support Programmes established the performance and the limits of the Fork Detector devices used for this purpose. First of all, this method cannot be called an "independent" verification, as the results heavily depend on data to be provided by the operator such as burn-up and cooling time. Falsified data will lead to incorrect measurement results. As stated in ref. [6], the detection limit of a partial defect is about 20% of pins missing for BWR assemblies with a burn-up of 18 MWd/kg or higher. Only if a large portion of the pins is removed from an assembly, this could be revealed by the use of a Fork Detector. Since pin exchange is an operation that is used only in exceptional cases in German reactors, the extraction or exchange of such a large number of pins and the operations necessary to remove them from the fuel pond would unavoidably show a lot of unusual operations in the surveillance records. Considering these circumstances the use of a Fork Detector is considered inappropriate compared to the use of other gross defect test devices such as the improved or digital Cerenkov Viewing Device.

The technical concept of the on-site interim storage is static, i.e., with very infrequent operator's activities. Therefore, the safeguards concept should be adequate and compatible with the requirements of radiation protection and operational safety. To the end of keeping radiation exposure of operator's staff and safeguards inspectors as low as possible, safeguards should require infrequent access to the casks. This could be achieved by the application of remote data transmission. Furthermore, failure of a seal should not require a re-measurement or even opening of a cask but should be backed up by redundancy measures.

A viable option could be for the plant operator to apply an electronic seal to the protecting lid of the cask in the absence of the inspector, provided the safeguards system is able to indicate correct functioning of the seal. The system could be based on the DCM14-surveillance system, the EOSS electronic seal, and the Euratom data acquisition software system RADAR.

5. SAFEGUARDS EXPERIENCE AND INTEGRATED SAFEGUARDS

During the loading phase of casks prior to transfer into on-site storage facilities IAEA inspectors have shown up three times, in order to perform the following safeguards activities: (1) Ion Fork measurement, (2) loading control, and (3) seal attachment. This seems to be an inspection effort that could be reduced. In one reactor facility two additional cameras were installed. A metal cap seal was attached to the secondary lid of the cask, while the protective lid was sealed with both a metal cap seal and a COBRA seal.

A pre-condition for the implementation of the IAEA's Integrated Safeguards is a state level evaluation of the country under consideration that allows the IAEA to draw the broader conclusion on the absence of undeclared nuclear materials and activities. The assurance gained through the Additional Protocol measures, i.e., that there exists no clandestine reprocessing facility in the state, should allow re-assessing the meaningfulness of the pin removal scenario. The model Integrated Safeguards approaches also foresee the use of unannounced or short notice random inspections. These are measures with a high deterrence value for a state considering a diversion of spent fuel in any form. In our view, there is no justified need for a different treatment of spent fuel, if the design allows the exchange of fuel pins, but this action is rather an exception than a normal procedure.

6. SUMMARY AND CONCLUSIONS

From the German point of view it is desirable to develop a standard safeguards concept that would match all 12 German nuclear power plant sites, where currently on-site dry storage facilities are being taken into operation on a step-by-step basis until April 2007. While individual technical and organisational differences have to be taken into account, the safeguards concept should be compatible with the requirements of radiation protection, operational safety, and physical protection.

Taking into account the fact that the exchange of fuel pins is rather an exception than a normal procedure in German reactors, the use of a Fork Detector is considered inappropriate to verify spent fuel prior to being transferred to dry storage. The use of other gross defect test devices such as the improved or digital Cerenkov Viewing Device should be sufficient to meet the verification requirements.

An on-site dry storage facility will be a material balance area (MBA) of its own, so that each power plant site will have at least two MBAs, for which the rules of nuclear material accounting apply.

In the maintenance and storage areas DCM14-type digital image surveillance systems could be implemented, while each transport and storage cask will be sealed in the reactor building, i.e., before its transfer from the wet storage pond in the reactor building to the dry storage hall. A COBRA fibre optic seal could be applied to the protective lid of the cask. During storage an EOSS electronic seal could be applied to a group of casks with the possibility to interrogate these seals from outside the storage hall. The reasoning behind this would be to minimize the need for personal access to the storage hall and, thus, to minimize persons' exposure to radiation. For the same reason, the potential of remote data transmission should be analysed. The DCM14-systems have approved remote data transmission capabilities. Finally, inspection effort could be further reduced, if the plant operator would handle electronic seals under camera surveillance in the absence of an inspector. The EOSS seal and the DCM14-technique offer this possibility.

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A multi-technique approach to environmental particle analysis for nuclear safeguards

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Abstract. The International Atomic Energy Agency uses Environmental Sampling for Safeguards (ESS) to give additional assurance of the absence of undeclared nuclear materials and activities in States covered by comprehensive safeguards agreements. The analysis of individual micrometer-sized particles (so-called "particle analysis") has been used since the beginning of the ESS programme because of its high inherent sensitivity (pg levels) and the wide range of isotopic information which can be obtained using mass spectrometric analysis techniques. This paper examines the application of a range of other analytical techniques to providing complementary information on individual particle morphology, structure and composition. The techniques applied are secondary electron microscopy (SEM) with energy-dispersive X-ray fluorescence microanalysis (EDX), transmission electron microscopy (TEM), micro-Raman spectroscopy, and ion-microprobe secondary ion mass spectrometry (IM-SIMS) used in full spectral mode. We start by applying these techniques to a range of U test compounds – uranium oxides, fluorides (UF_4 and UO_2F_2) and di-uranate salts – to establish the fingerprint information available (including, in the case of UO_2F_2, a preliminary assessment of environmental stability). The diagnostic value of the multi-technique approach is then trialled by application to the differentiation of a set of 6 "blind" UO_2 samples of identical isotopic composition. This was successfully achieved. Our overall conclusion is that application of the expanded range of particle analysis techniques has much to offer in the analysis of nuclear environmental samples, providing information on, for example, the process history of particles.

1. Introduction

Environmental sampling has proven to be an invaluable technical verification measure available to the International Atomic Energy Agency (IAEA) in support of international nuclear safeguards [1]. Residues are collected by IAEA inspectors using cotton swipes, and these swipes distributed to the IAEA's worldwide network of analytical laboratories (NWAL) for extraction and isotopic analysis using either fission-track thermal ionisation mass spectrometry (FT-TIMS) or secondary ion mass spectrometry (SIMS).

Isotopic composition, albeit of crucial importance for the identification of nuclear processes, is but one part of a spectrum of information – including particle morphology [2], stoichiometry, purity, structure and bonding - potentially available from a large range of techniques applicable to particle analysis. The present paper investigates the diagnostic information available from a sub-set of these – secondary electron microscopy (SEM) with energy-dispersive X-ray fluroescence microanalysis (EDX), transmission electron microscopy (TEM), Raman spectroscopy, and use of ion-microprobe (IM) SIMS in full spectral mode. We first catalogue the results from a range of model U compounds, including UF_4, different U oxides, UO_2F_2 (freshly-prepared at IRMM Geel, Belgium [3]), and ammonium diuranate, before going on to explore their application in field studies. The example field study reported here is the analysis of a set of 6 blind" UO_2 samples which could not be differentiated by isotopic analysis.

2. Experimental

2.1. Samples

The U test compounds analysed in this work are listed in Table 1. Materials 1-10 were bulk powder samples of declared composition from the nuclear fuel cycle. These were prepared by suspension of small quantities in ethanol, deposition onto substrates and allowing to dry, so as to appear similar in form to extracts from environmental swipes prepared for SIMS analysis. Different substrate types were used as appropriate for the different techniques - Si wafer pieces for SEM-EDX and Raman (also SIMS in some cases), carbon planchets for SIMS, and holey-carbon grids for TEM. Sample 11 comprised U_3O_8 reference particles prepared by aerosol formation from uranyl nitrate solution [4], and supplied deposited onto blank environmental swipes. These were extracted and deposited onto a carbon planchet following the standard procedure used for SIMS analysis.

Table 1. Description of U test compounds used in this study.

Sample ID	Compound	Bulk Colour	Theoretical U concn. (wt %)	Found U concn.[a] (wt %)	Techniques applied
1. 7049-01	UF_4	Dark green	75.8	76.2	SEM, TEM, Raman, SIMS
2. 7555-03	UF_4	Light green	75.8	60.1	SEM, TEM, Raman, SIMS
3. 7049-02	UO_2	Black	88.1	86.5	SEM, SIMS
4. 7041-05	UO_2	Brown	88.1	87.9	SEM, TEM, Raman, SIMS
5. 7069-02	UO_2	Green/brown	88.1	87.7	SEM, SIMS
6. 9055-01	U_3O_8	Black	84.8	83.5	SEM, TEM, Raman, SIMS
7. 9064-01	ADU[d]	Yellow	76.3	74.7	SEM, Raman, SIMS
8. 9081-01	UO_4[b]	Yellow	78.8	68.7	SEM, Raman, SIMS
9. 9042	UO_3	Yellow	83.2	78	SEM, Raman, SIMS
10. 7714	Yellowcake	Yellow	NA[c]	NA	SEM, TEM, SIMS
11. 8132	U_3O_8	NA	NA	NA	SEM, SIMS
12. IRMM [3]	UO_2F_2	NA	NA	NA	SEM, Raman, SIMS

[a] Bulk U assay by NBL-modified Davis and Gray titration at SAL.

[b] This sample was declared as "UO_4" which is not a recognized oxide of uranium. The material was a result of precipitation from solution by treatment with H_2O_2.

[c] Not available

[d] Ammonium diuranate, $(NH_4)_2U_2O_7$

In the case of UO_2F_2, where there was specific interest over the environmental longevity of the material, particle samples were freshly prepared for investigation by controlled vapour phase hydrolysis of UF_6 at IRMM Geel, Belgium, as detailed in reference [3].

2.2. Anaytical techniques

SEM-EDX. Samples were examined at low-to-medium resolution using a Jeol 6100 SEM (W-filament source), and/or at high-resolution using a Jeol 6400 field-emission SEM. EDX analysis was carried out in the Jeol 6100 SEM using an Oxford ISIS detector with ultrathin window. Spectra and imaging were carried out using 5-20keV electron beam energies. For EDX spectra from particles, 10keV was found to be optimal in most cases to minimise substrate signals whilst retaining sensitivity to U and other elemental constituents of the particle.

Raman spectroscopy. A Renishaw micro-Raman spectrometer was used with Ar-ion laser incident radiation (514nm) focussed to a ca.1µm diameter spot onto selected specimen particles. The laser was used at its lowest power settings, sometimes with further attenuation using neutral density filters to minimize the risk of damage to specimen particles. Spectra were collected over the 200-4000cm^{-1} range using typically 10-60 minute acquisition times. Samples deposited onto Si wafer substrate pieces were mostly used, as their high surface quality and reflectivity simplified the optical selection and positioning of particles for analysis.

TEM. Samples for TEM were prepared by solvent deposition onto holey-carbon electron-transparent grids. Bright-field TEM imaging was carried out using a Jeol 4000EX microscope operating at 400keV. Image calibrations are relative to standard TEM calibration samples. Selected area diffraction (SAD) patterns and EDX spectra were also recorded.

Ion-microprobe (IM) SIMS. A Cameca 4f spectrometer was used with 8.5keV O_2^+ primary ion bombardment at a current of 1-2nA in a focused spot (estimated diameter 5-10µm). Mass spectra spanning either 0-300amu (full mass range) or 180-300amu (high mass range) were then collected, scanned in 0.1amu steps with 0.1s dwell time. Samples were analysed at several positions. In each case a 150-250µm diameter resistive anode encoder (RAE) field was imaged and a prominent U particle or agglomerate selected for the analysis. Whole-field spectra, obtained using a defocused spot, were also recorded.

3. Results from analysis of U Test Compounds

In the folowing discussion, analytical results from the samples in Table 1 are presented and discussed by technique, to allow ready comparison of the diagnostic features.

3.1.1. SEM-EDX

Figure 1 shows examples of the particle morphologies encountered. A common feature was that larger particles appeared to comprise a sub-structure of smaller grains or nuclei. This often caused larger particles to appear porous or flaky, with the grains defining the smallest indiviudal particles seen. In some cases distinct large particle forms were observed, as exemplified by the U oxide samples (11) (spherical) and (3) (tendency to appear facetted), although some porosity is evident in both cases. Irregularly-shaped large particles were frequently associated with a more fragile sub-structure in which the individual grains were more clearly apparent. U_3O_8 sample (6) particles comprise a network of interconnected rounded grains in the 50-500nm size range, which was a common morphology amongst the sample set investigated. UF_4 sample (2) particles showed distinctly platelet-like or lamellar grains, and the large particles seemed correspondingly more fragile, with some evidence of facetting, as a result.

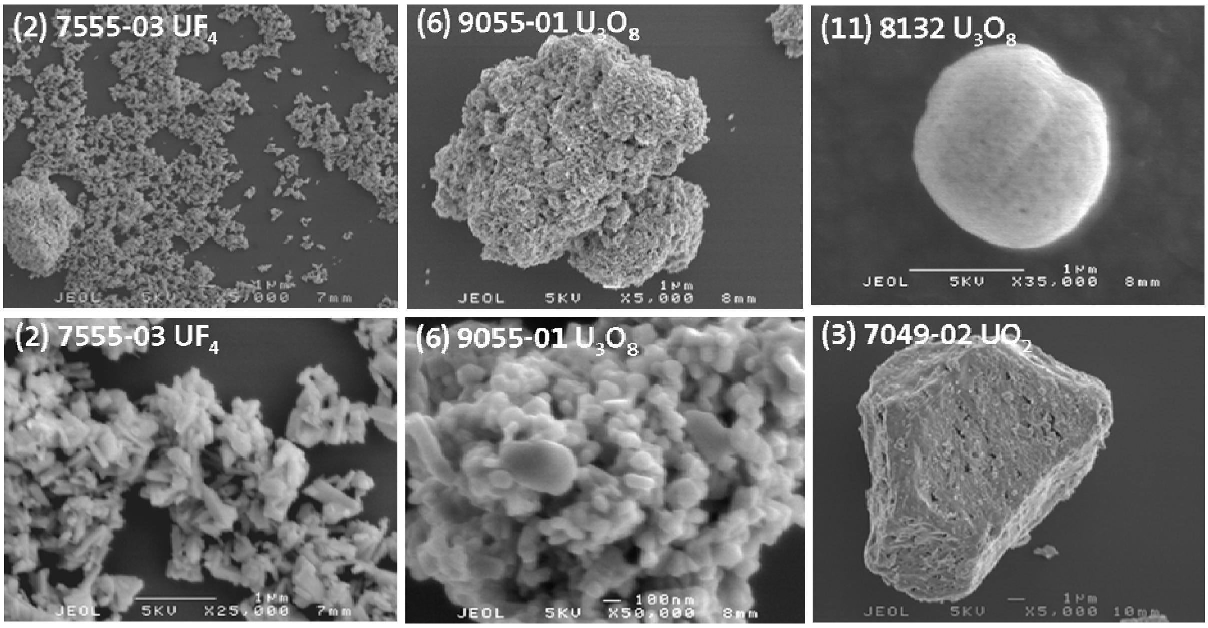

FIG. 1. Example SEM images from U test compounds showing range of particle morphologies.

3.1.2. TEM

A sub-set of five U compounds (1,2,4,6 & 10) were investigated using TEM. TEM gave more detailed information on the grain structure, complementing the SEM analysis. The example TEM image in figure 2 from a U_3O_8 sample (6) particle shows it to be a cluster of agglomerated and conjoined grains in the 50-200nm size range with a variety of shapes (rounded and facetted). The polycrystalline SAD pattern from the particle is consistent with each grain being a separately-oriented crystallite. In fact, SAD patterns from all samples investigated showed individual grains to be crystalline with either a hexagonal close-packed or face-centred cubic structure. Detailed TEM images from many grains showed diffraction contrast due to internal strain or defects. Some, especially platelet-like particles, contained pinholes or voids.

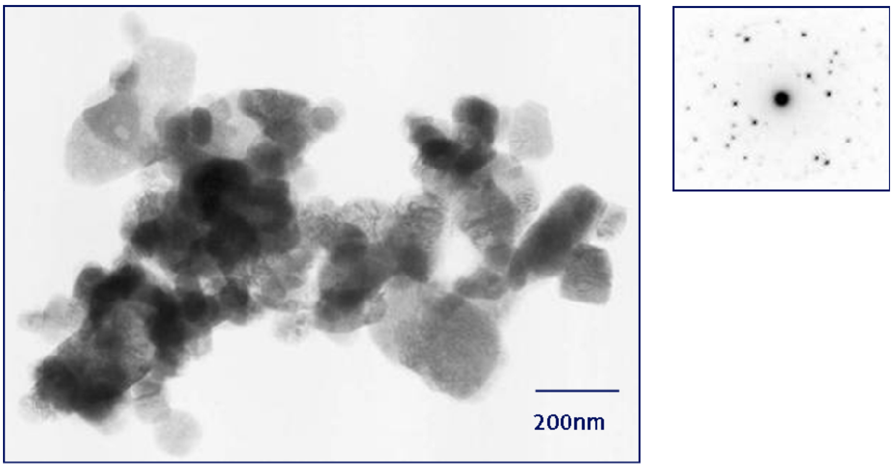

FIG. 2. TEM image and SAD pattern from U_3O_8 sample (6) 9055-01 particle.

3.1.3. Micro-Raman spectroscopy

In general, very few or weak spectral features were seen outside the range 400-1150cm^{-1}, and only four of the samples exhibited sharp peaks which appeared valuable for diagnostic purposes. These were (1) UF_4 (917cm^{-1}), (8) UO_3 (830 & 870cm^{-1}), (9) UO_4 (840cm^{-1}), and (10) UO_2F_2 (866cm^{-1}).

Peaks in the latter case were extremely weak, though this may reflect the small particle size. The absence of a peak at 917cm^{-1} in UF_4 sample (2) may be associated with a reduced level of crystallinity evident from TEM images (possibly related to a very small grain size). Spectra from the UO_2 and U_3O_8 samples investigated showed several relatively weak and/or broad peaks in the 400-1150cm^{-1} range. However, significant particle-to-particle variations were also observed. In general in Raman it is necessary to position the particle exactly at the focal point of the beam, where it experiences the maximum laser power density, or else signal strengths may be significantly reduced. Some materials may damage under such conditions, in which case reliable spectra can only be obtained at much reduced laser power. This was the case with ADU, where noticeable darkening was observed after exposure, unless the beam was attenuated. Overall these results are similar to those reported recently by Baude [5].

3.1.4. Ion-microprobe SIMS spectra

As expected, U-containing molecular ions in the mass spectral range m/z 230-280 have proven most diagnostic of the test compounds investigated. In this region the Si substrate contributes only very weak clusters with maxima at at m/z 236 ($Si_5O_6^+$), 252 ($Si_5O_7^+$) and 268 ($Si_5O_8^+$) amu, plus USi^+ at m/z 266. In lower mass regions, spectra from the Si substrate dominated, though additional impurity signals were also observed.

Figure 3 shows example spectra from U oxide and fluoride particles of natural U isotopic abundance. All U oxide spectra were dominated by the U^+, UO^+, and UO_2^+ ion series, giving ^{238}U-series peaks at m/z 238, 254 & 270 amu, and with corresponding ^{235}U (depending on isotopic enrichment) and UH ion series also present. It is immediately apparent from figure 3 that F in UF_4 contributes very strong and diagnostic ions at m/z 257 ($^{238}UF^+$), 273 ($^{238}UFO^+$) and 276 ($^{238}UF_2^+$).

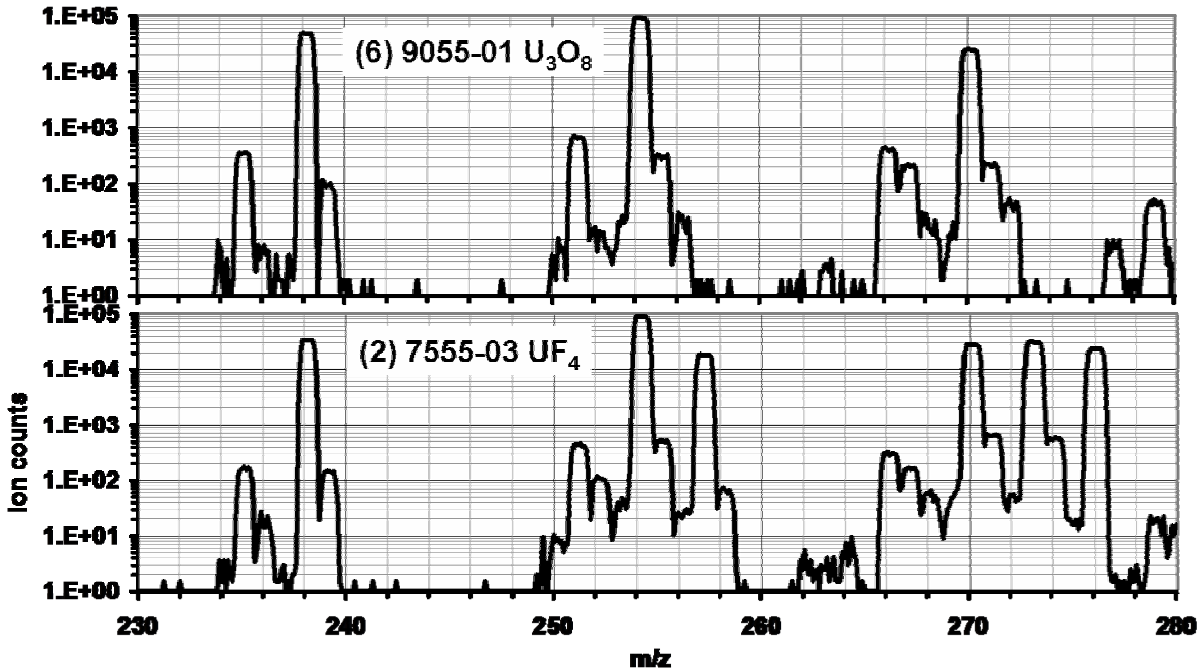

FIG. 3. IM-SIMS spectra from example U fluoride and oxide test compound particles.

Apart from the UF ion series, only the relative increases of UO_2^+ and UH-series signals at higher oxidation state (ie UO_3 or “UO_4”) appeared (weakly) diagnostic. In the case of ADU, no evidence of the presence of N, either at mass 14 or UN^+ at 252, was apparent. Quantitatively, and allowing for particle-particle variation, UF^+/U^+ peak ratios from UF_4 were consistently in the range 0.1-1, and UO^+/U^+ ratios in UO_2 & U_3O_8 were 0.1-1, whereas ratios >1 appeared symptomatic of the higher U oxides.

3.1.5. *IM-SIMS spectra from UO_2F_2 particles*

UO_2F_2 particles are an expected consequence of industrial UF_6 release, and a preliminary study of their diagnosis and environmental stability was carried out in collaboration with R.Kips and R.Wellum at IRMM. This group have developed a process for particle deposition directly onto substrates by the controlled vapour phase hydrolysis of UF_6, followed by vacuum desiccator stabilisation [3]. The main aims of the present work were to characterize the UO_2F_2 analytical signature (a) fresh after preparation, (b) after open storage in air, and (c) after heat treatments of the type used for regular environmental swipe preparation.

Figure 4 compares spectra from UO_2F_2 particles (in which the $^{235}U/^{238}U$ ratio is ~20%) after 4 months storage and a 450°C heat treatment (as is used in the standard swipe preparation procedure for SIMS analysis). Spectra from freshly-prepared particles showed strong UF ion series (with the UF^+/U^+ peak ratio ~0.1, at the lower range of the values in UF_4 spectra) and these showed only a slight decrease after 4 months ageing in laboratory air. Comparison of UO_2F_2 and UF_4 spectra in figures 3 & 4 show that the compounds are readily distinguishable from the relative patterns of UO_2/UOF/UF_2 ion series.

Heating resulted in a major reduction of the UF ion signals to a UF^+/U^+ ratio of ~0.001 at 250°C and above. This residual level of the peak at m/z 257 appeared to be relatively stable, and was detectable, at least for the case of larger particles, owing to the high SIMS sensitivity and dynamic range. This suggests that evidence of UO_2F_2 could be expected to survive field sampling and NWAL preparation.

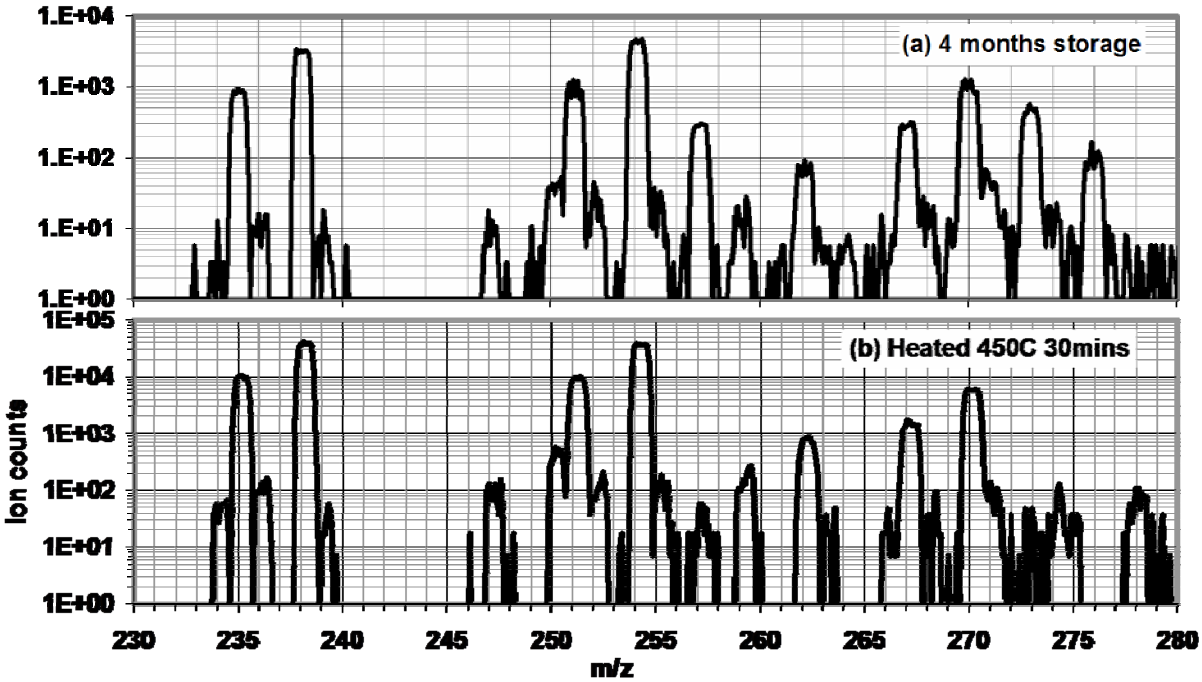

FIG. 4. IM-SIMS spectra from treated synthesised UO_2F_2 particles [3].

4. Results from “blind” UO_2 samples

Investigation of a set of six “blind” UO_2 samples was performed, deploying the expanded range of particle analysis techniques. These samples were all declared to be UO_2 of nuclear-grade purity (with U assay to confirm this), and were previously determined to be indistinguishable by isotopic analysis (i.e. all were natural U). They had been obtained at different times from 3 different facilities, and there was interest in determining whether any differences in process history could be distinguished. The main methods applied were IM-SIMS, Raman, SEM and TEM. SEM-EDX and IM-SIMS were used to test for any differences in elemental impurities present that could act as a “signature”. In fact no impurities were detected in EDX spectra from the particles, and IM-SIMS spectra from the largest particles probed on each sample were similarly inconclusive. That no significant differences in the elemental particle analysis were found is testament to the high purity of the samples. Raman spectra were also inconclusive, showing no clear systematic sample-to-sample differences. This is consistent to some extent with the results from the U test compounds showing relatively weak and broad spectral features from UO_2, hence of limited diagnostic value in this particular problem.

SEM images from numerous particles were collected from each sample. These showed a range of characteristics, which, as for the test U compounds, where most readily categorised at both the single particle level and at the smaller grain level. The observations are summarised in Table 2, and by the representative images shown in figure 5, from which it is straightforward to group the set of samples into 3 pairs on the basis of their particle morphologies.

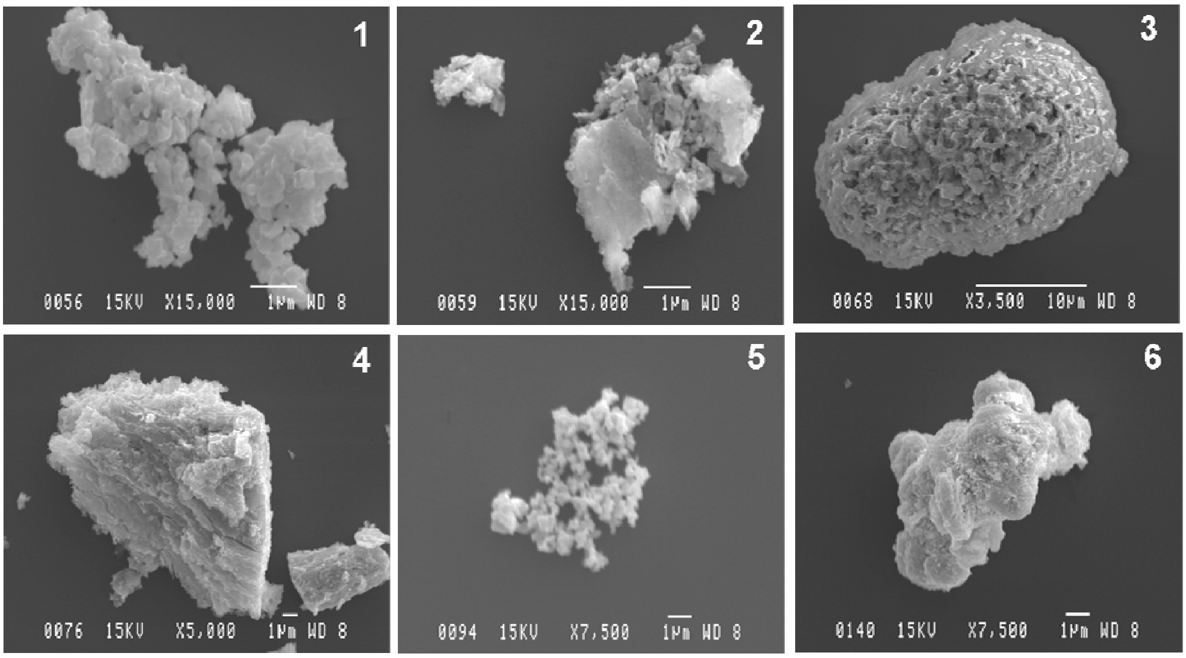

FIG. 5. Representative SEM images from the 6 "blind" UO_2 sample particles.

Table 2. Results of morphological examination of "blind" UO_2 field samples.

Sample	SEM	TEM
1	Irregular particles appear to be agglomerates of fused sub-particles	100-500nm crystalline sub-grains frequently fused at interfaces
2	Irregular particles showing a distinct flake-like (lamellar) sub-structure	Confirms flake and needle-like fragments present, some with hexagonal voids or pits
3	Intact rounded larger particles with porous, net-like substructure	Melded rounded crystalline sub-grains of 50-200nm containing voids or pits
4	Irregular particles showing distinct flake-like sub-structure	Smaller particles consistent with 50-250nm-sized flakes with many voids and pits
5	Irregular particles comprising agglomerated sub-micron sub-particles	20-500nm crystalline sub-grains frequently fused at interfaces
6	Intact larger particles having fused rounded shapes with a porous internal structure	Melded crystalline sub-grains in the 30-100nm range

Particles from samples 2 and 4 were characterized by their very irregular form, with a lamellar structure and a strong tendency to flake. Particles from samples 3 and 6 were recognizable by the presence of intact rounded particles based on spheres or agglomerated spheres. These have a porous, or cage-like, structure, comprising a fused array of grains in the 100nm-size range. Particles from samples 1 and 5 comprise similarly sized sub-grains, though the particles are much more irregular in shape. The grain networks may be more open and/or less strongly interconnected than samples 2 and 4, resulting in greater fragmentation. TEM images gave more information at the grain level, fully supporting the SEM findings. Particles from samples 3 and 6 appeared as dense networks of smaller particles (50-200nm sized grains or nuclei) appearing rounded and in many cases necked or melted together without clear boundaries. In the case of particles from samples 1 and 5, irregular fragments comprising a few grains were more common, with clear grain boundaries in many cases. Particle images from samples 2 and 4 were consistent with clusters of overlying flake-like grains.

This grouping of the 6 UO_2 samples, carried out “blind”, based on classification of their particle morphologies as described above, was fully consistent with the known origins of the samples.

5. Conclusions

The techniques applied in this study are capable of supplying two classes of particle information that may be of value in environmental nuclear safeguards monitoring – composition (bulk and impurities, from IM-SIMS, EDX and Raman) on the one hand and morphology (with some structural information also, from SEM, TEM and Raman) on the other. Bulk compositional information should, in combination with isotopic analysis, enable the particle origin in the nuclear fuel cycle to be pinpointed. An impurity fingerprint may be additionally specific to a particular process or equipment. Classification of particle morphology is expected to inform on process history [2], at the simplest level for example whether vapour- or liquid- phase. The techniques may be applied in a purely fingerprinting manner, in order to answer specific questions related to comparing or matching different samples. Indeed, as the successful case study presented here have shown, they are already deployable for this purpose. However, greater value may ultimately be derived from the ability to make interpretations of the process origins of particles founded on understanding. In this respect the methods are still at an early stage and it is necessary to accumulate a wider base of analytical data and experience. The results presented here suggest that this would be a valuable and worthwhile exercise.

ACKNOWLEDGEMENTS

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Development of safeguards environmental sample analysis techniques at JAEA as a Network Laboratory of IAEA

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Abstract. JAEA has been developing analytical techniques for ultra-trace amount of nuclear materials in the environmental samples in order to contribute to the strengthened safeguards system. Development of essential techniques for bulk and particle analysis of the environmental swipe samples has been established as ultra-trace analytical methods of uranium and plutonium. In January 2003, JAEA was qualified as a member of the IAEA network analytical laboratories for environmental samples. Since then, JAEA has conducted the analysis of domestic and the IAEA samples and obtained reliable data. In addition, JAEA is developing the analytical techniques to improve the analytical ability for the safeguards environmental samples. For bulk analysis, study is focused on the improvement of reliability of isotope ratio measurements by ICP-MS. Alternative chemical separation schemes are under development and a desolvation module is introduced to reduce the polyatomic interferences. In particle analysis, the sample preparation procedure for SIMS method is modified to measure the $^{234}U/^{238}U$ and $^{236}U/^{238}U$ ratios for individual particles. We are also developing fission track-TIMS method to detect uranium particles more effectively from the viewpoint of preliminary estimation of uranium enrichment. A screening instrument of X-ray fluorescent analysis is equipped to measure elemental distribution on a swipe surface, which provides information about impurities to be considered in chemical purification of uranium and plutonium. This paper deals with the track of research and development activities at JAEA as well as progress in development of advanced techniques.

INTRODUCTION

Nuclear nonproliferation is one of the most important issues for peaceful utilization of atomic power. In current international situation where undeclared nuclear activities become serious threats to the nonproliferation regime, environmental sample analysis plays more important role in the strengthened safeguards system of the IAEA. The objects of the analysis at present are swipe samples taken from nuclear facilities. In order to detect undeclared nuclear activities, it is necessary that isotope ratios of nuclear materials of ultra trace amount in the samples are accurately analyzed.

Japan Atomic Energy Agency (JAEA) has been developing techniques for the environmental sample analysis since 1996 under the auspices of the Ministry of Education, Culture, Sports, Science and Technology of Japan (MEXT). A clean-room facility, Clean Laboratory for Environmental Analysis and Research (CLEAR), was constructed in 2001. By the end of 2002, fundamental techniques for ultra-trace analysis of uranium and plutonium were established with sufficient sensitivity and accuracy [1-4]. This resulted in the qualification of JAEA, in 2003, as a member of the IAEA Network of Analytical Laboratories (NWAL) with regard to both particle analysis and bulk analysis, the latter of which covered hot-cell swipe samples as well. Thereafter JAEA has contributed to the international safeguards as well as to the domestic safeguards, by carrying out the analysis of safeguards environmental samples.

Since 2003, the second phase of the project was started to develop advanced techniques, with which JAEA aims at technical support to the enhancement of safeguards system. This paper deals with the track of research and development activities at JAEA as well as progress in development of advanced techniques.

OUTLINE OF ANALYTICAL TECHNIQUES

Figure 1 shows the analytical techniques for the safeguards environmental samples in JAEA. The non-radioactive samples are analyzed at CLEAR. The radioactive samples and the samples from hot cell are treated at NUCEF. Before bringing swipe samples into the clean area, α, β and γ rays of the swipe samples are measured with a survey meter and then with a high resolution γ-ray detector to estimate the radioactive materials such as uranium, plutonium, americium, fission products, and so on.

For bulk analysis, after high temperature ashing, the swipe samples are digested with nitric acid, hydrofluoric acid and perchloric acid. Uranium and plutonium in the obtained solution are separated by the anion-exchange method in hydrochloric acid media. The isolated uranium and plutonium are measured by ICP-MS at CLEAR and by TIMS at NUCEF. The isotope dilution method (IDM) is adopted to determine the amounts of uranium and plutonium. Total-reflection X-ray fluorescence spectrometry (TXRF) is used to determine the amount of spike to be added to the solution [4].

For the particle analysis, secondary ion mass spectrometry (SIMS) and fission track-thermal ionization mass spectrometry (FT-TIMS) are adopted. The particle recovery method with impact collector was developed. After screening for the presence of uranium by TXRF, the isotope ratio of uranium in individual particles is measured with SIMS [5,6]. In the FT-TIMS method, a fission track detector is prepared to find particles containing uranium and the isotope ratio of uranium is measured with TIMS [7].

RESEARCH AND DEVELOPMENT ON ADVANCED TECHNIQUES

Bulk analysis

The swipe cotton material, TexWipe 304® currently used in environmental sample analysis of the IAEA, contains a few nano-gram of natural uranium as impurity, which primarily limits the detection limit of uranium isotope analysis even though the analysis is carried out in clean room environment. Recently, a new synthetic fiber swipe material, ExlanWipe® was manufactured on trial by JAEA and EXLAN Co. The main ingredient of the new swipe is acrylonitrile and free from insoluble impurities, such as TiO_2. It was found that the uranium content was 43 pg/wipe (see Fig.2), which was about 1/100 of TexWipe 304®. The examination showed that the swipe had favorable ashing and acid digestion properties as well as almost same swiping performance [8].

It is well known that $^{238}U^{1}H^{+}$ interferes with the measurement of ^{239}Pu. In addition, if a certain amount of lead presents in the sample, the polyatomic ions of $^{207}Pb^{16}O_2^{+}$ and $^{208}Pb^{16}O_2^{+}$ interfere with the measurement of ^{239}Pu and ^{240}Pu, respectively. Spectroscopic interference by polyatomic ions is the serious problem for the isotope ratio measurement of ultra-trace amounts of nuclear materials by ICP-MS. The most of polyatomic ions are composed of metallic atoms as impurities and non-metallic atoms from solvent. We try to reduce the intensity of polyatomic ion from the standpoint of the chemical separation and the mass spectroscopy.

Although the technique for the bulk analysis shown in Fig.1 has been applied to the real samples successfully, it is necessary to improve the chemical separation ability in case of samples containing large amounts of heavy metals such as lead. We developed further separation method using anion-exchange resin disk (Empore® Anion-SR, 47 mmϕ). In this method, a centrifugal machine was used to increase the flow rate of solutions through the resin disk, which reduced the processing time. Figure 3 shows the Teflon tube for the separation. The resin disk was set on the disk holder (Fig.3(a)). After assembling the parts as shown in Fig.3(b), the Teflon tube (Fig.3(c)) was set into the centrifugal machine and used in the same manner as a centrifuging tube. The advantage of this method was that diluted nitric acid could be used to elute plutonium from the resin disk and the eluted solution could be directly measured by ICP-MS. Since major source of uranium blank in the bulk analysis is uranium elution from the surface of Teflon vessels during the dry-up process for the matrix matching [3,4], the separation using the resin disk free from dry-up process has a potential for low blank chemistry.

Another challenge to reduce polyatomic interference is to make a desolvation module practical as sample introduction systems of ICP-MS. The desolvation module reduces the intensity of solvent-derived polyatomic ions.

Particle Analysis

SIMS

For particle analysis, we developed an effective method by the combination of TXRF and SIMS [9,10]. In this method, the particles in the swipe samples are recovered onto a Si carrier with vacuum suction – impaction collector (see Fig.4). The carrier surface is coated with grease to avoid bounce-off effect of the particles. Prior to the SIMS measurement, the carrier with the collected particles is screened for uranium by TXRF. The combination method is effective and routinely used in the $^{235}U/^{238}U$ ratio analysis for individual particles with micrometer diameter.

We improved this method to measure the $^{234}U/^{238}U$ and $^{236}U/^{238}U$ ratios in individual particles. In this case, reducing background signals is important because the signal intensities of $^{234}U^+$ and $^{236}U^+$ are very weak. For this purpose, the Si carrier is not preferable because it produces polyatomic ion peaks at masses 236, 237 and 238. Because the glassy-carbon carrier without grease gives no interferences on the uranium ion peaks, the carrier is replaced by the glassy carbon carrier. However, background signals significantly increase by using the grease on the carrier surface. Figure 5(a) shows the spectrum of the glassy carbon carrier coated with grease. In this spectrum, background signals were observed at the whole mass range. These signals make us impossible to measure the $^{234}U/^{238}U$ and $^{236}U/^{238}U$ ratios accurately. We found that these signals could be reduced to negligible levels (see Fig. 5(b)) by heating the carrier at 340 ºC and evaporate the grease. Furthermore, a new technique was developed. Individual particles were picked up, with a gold-coated fine glass needle, from a Si carrier under scanning electron microscope (SEM) and transferred to another carrier, which were followed by SIMS analysis [11]. This technique has an advantage when uranium particles coexsist in the carrier with large amounts of metallic impurities.

FT-TIMS

For the SIMS method, it is difficult to analyze sub-micrometer particles due to the limitation of SIMS sensitivity. On the other hand, the FT-TIMS method, in which the particles of interest are confined in a fission track detector, has the merits such as high detection efficiency, simplicity of sample preparation and the possibility to classify uranium particles according to their enrichment by controlling the etching time [12]. However, it was found that a part of uranium particle contained in the detector may dissolve during the etching process of the detector.

In order to overcome the problem, we are developing a sample preparing method in which the detector of fission track and the layer containing particles are separated. In this case, the detection process of the particle with fission tracks is time-consuming and complicated due to the discrepancy between the position of the particles and the fission tracks. We newly developed the method to solve the problem by fixing a part of a detector and a particle layer [13]. In this method, it was confirmed that the discrepancy was negligible even though the etching process was repeated. To find the particle containing uranium, the fission track is firstly sought under a microscope by focusing on the detector. Once it is found, and the particle can be easily detected to move the focus from the detector to the particle layer. Figure 6 represents the example of uranium particle detection in which uranium particles and dusts are clearly distinguished. The block spots in solid circle indicate the uranium particles and the corresponding fission tracks can be seen in the left-hand side of the particle. On the other hand, the block spots in broken circle indicate the dusts since the corresponding fission tracks are not observable. The detected particle is cut out and then the isotope ratios are measured with TIMS.

ICP-TOFMS

For particle analysis, the detection probability of the undeclared nuclear activities increases with increasing the number of particles measured. The particle analysis by SIMS normally takes two or three days to measure about hundred particles. The FT-TIMS method is also time consuming. We are developing an instrument to measure the isotope ratios of particles automatically, which may have the ability to measure a few thousand particles within one day. The instrument consists of an inductively coupled plasma (ICP) ion source and a time of flight mass spectrometer (TOFMS). Individual particles in the swipe sample are directly transferred to the ICP ion source by an argon gas flow. Produced ions are analyzed with TOFMS. In the TOFMS, ions are separated according to the mass to charge ratio, which enables us to record a wide range mass spectrum rapidly. Optimization of the instrumental set up and the analytical parameter will be performed. And then the instrument will be applied to the safeguards environmental samples.

Screening

The estimation of uranium amount in swipe samples is important in order to know the optimal amount of the spike to be added for uranium quantitative analysis using IDM. γ-ray measurement is often applied but it is inefficient due to the poor detection limit of uranium (about 1μg per swipe sample) because the uranium isotopes in interest have long half life and the emission probability of γ rays are low. On the other hand, heavy metals like lead interfere with the measurement of the isotope ratios of uranium and plutonium as mentioned in "Bulk analysis" section. A suitable chemical separation scheme should be selected to obtain a sample solution with acceptable level of heavy-metals. It is, therefore, important to estimate uranium and heavy metals in the samples before analysis. Such information is also very useful for particle analysis because it enables us to carrry out effective particle recovery and to foresee possible interference in SIMS analysis.

Semi-quantitative measurement instrument was developed using X-ray fluorescent spectrometer as shown in Fig 7. This instrument can determine elemental distribution on a swipe surface: target elements are from potassium to uranium were measured by each area of 7 x 5 mm on a swipe sample. To avoid cross contamination, the instrument was designed to measure the swipe sample in a plastic bag. Performance evaluation tests are in progress. Until now, it was detemined with simulated swipe samples containing various amounts of lead that one sample was measured within 6 hours with sufficient sensitivity, as well as linearity between concentration and signal intensity at the lead amount of 30 – 1,000 ng/spot.

CONCLUSIONS

JAEA joined the IAEA network analytical laboratories for environmental samples. Since then, we conducted the analysis of domestic and the IAEA samples. In parallel, we are developing new techniques and introduced new technology to improve the analytical ability of the environmental samples for safeguards.

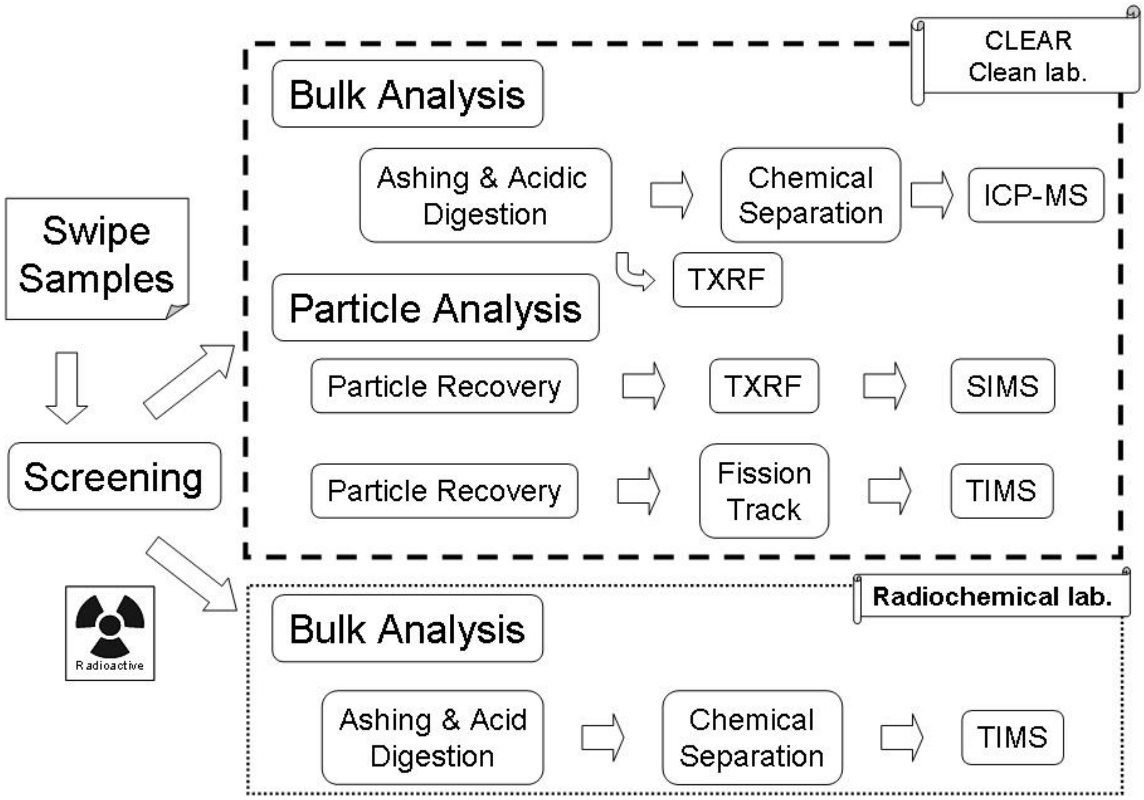

Figure 1. Flow diagram of environmental sample analysis at JAEA.

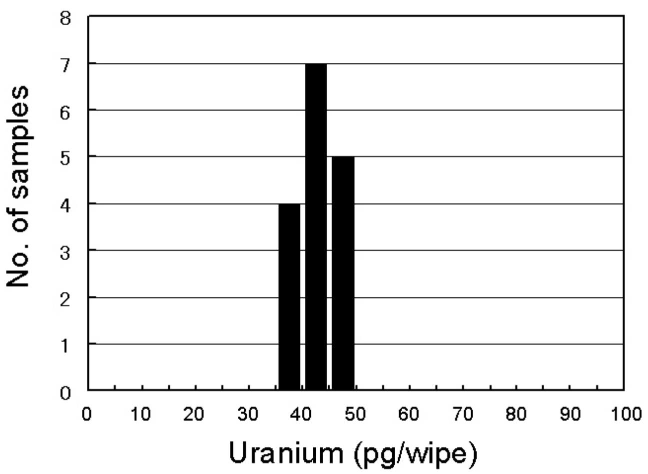

Figure 2. Uranium content in new swipe material (ExlanWipe®). Less uranium (43 ± 3 pg/wipe) than TexWipe 304® (2,000-5,000 pg/wipe).

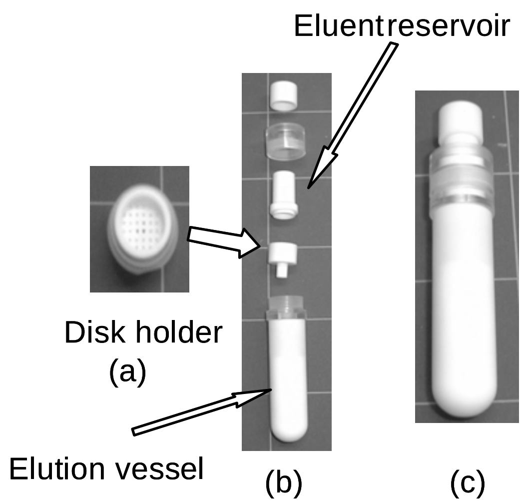

Figure 3. Teflon tube for centrifugal ion exchange.

Figure 4. vacuum suction – impaction collector.

Particles in swipe sample are removed by vacuum pumping and collected onto carrier by impaction.

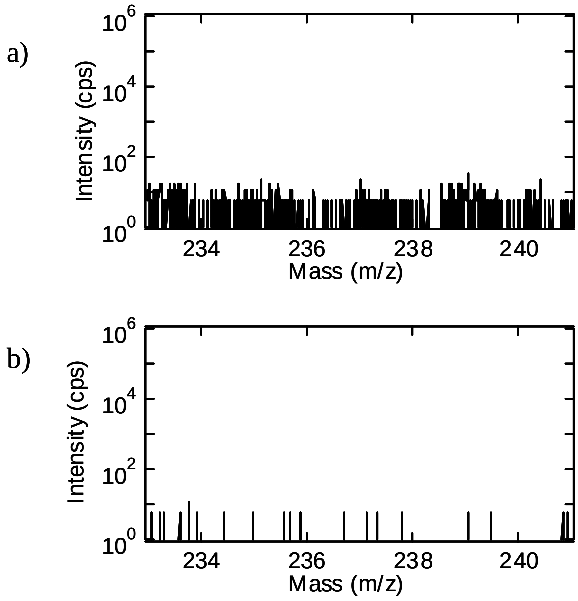

Figure 5. a) Mass spectra of the glassy-carbon carrier coated with grease; b) Mass spectra of the carrier after heating at 340°C. The temperature was measured at the carrier surface.

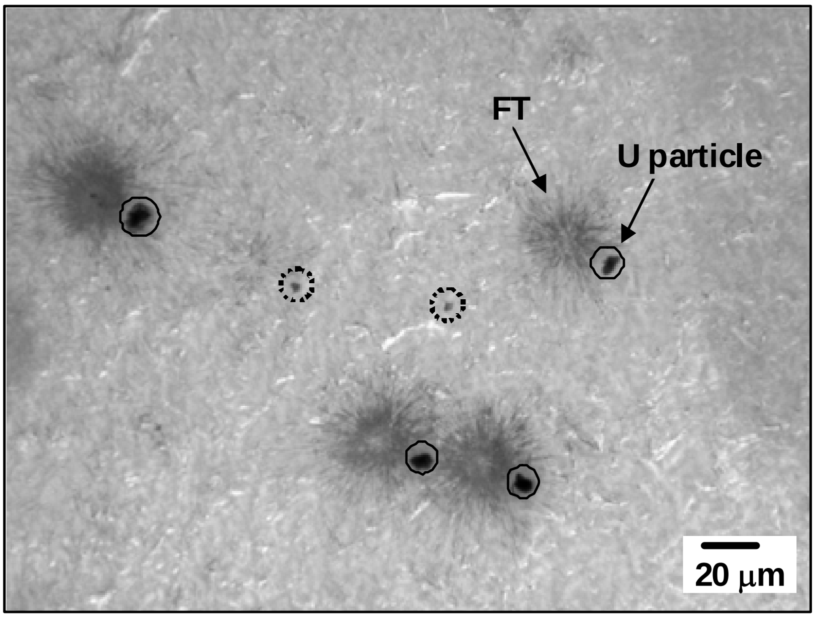

Figure 6. Microscopic image of the fissionn tracks and uranium particles with natural isotopic composition observed on the ‘same screen’ (solid circles: uranium particles, broken circles: dust).

Figure 7. A screening instrument to measure elemental distribution on swipe surface.

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Detection and characterization of neutron activation activities using gamma spectrometric analysis

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Abstract. Neutron activation analysis (NAA) of environmental samples is normally performed using research reactors. IAEA inspectors collect swipe samples at such reactors to verify that undeclared irradiation activities have not been carried out. High resolution gamma spectrometry (HRGS) analytical results of the collected swipes may identify signatures which indicate irradiation activities have been performed at the reactor. The modelling of typical irradiation scenarios used for NAA of material with known elemental composition, followed by the comparison of the calculated activities with the HRGS measurement results of the swipe, allows for the identification of the approximate irradiation date and the type of irradiated material present on the swipe. This approach has been applied for identifying signatures of NAA of soil samples.

1. Introduction

Research reactors are used for a variety of activities including neutron activation analysis (NAA) of various environmental samples, such as soils, sediment, vegetation, rocks and ores. IAEA inspectors collect swipe samples at such reactors to verify that undeclared irradiation activities have not been carried out. The samples are then measured by high resolution gamma spectrometry (HRGS) to determine the activities of gamma emitters present on the sample. This information may be used to identify signatures of various irradiation activities at a reactor, such as NAA or medical isotopes production.

This paper presents the results of the modelling of a typical irradiation process used for NAA. The results can be used to identify the approximate date when irradiation took place and the type of irradiated material present on samples (e.g., soil, vegetation and ore).

2. Example scenario

We consider the typical HRGS measurement results of a swipe sample taken in the hot cell of a research reactor loaded with low enriched uranium (LEU) fuel. The reactor is used for:

(a) Irradiation of Te, Sm, Mo or Ir targets for radioisotope production;

(b) Neutron cross-section studies and fission track analysis;

(c) Irradiation of ^{197}Au for neutron flux monitoring;

(d) NAA of biological samples, electrolytes, films, soil, sediment, vegetation, rock and ore; and

(e) Medical isotope production, such as ^{99}Mo and ^{131}I.

The analysis results are given in Table 1.

Table 1. HRGS measurement results for the swipe sample.

Isotope	Half-life, days	Activity, Bq/swipe
^{46}Sc	83.79	15.4±.0.5
^{51}Cr	27.702	86.9±3.1
^{54}Mn	312.12	29.6±0.9
^{58}Co	70.82	7.1±0.6
^{59}Fe	44.503	47.4±1.2
^{60}Co	1925.4	238.6±5.0
^{65}Zn	244.26	30.1±1.1
^{95}Nb	34.975	1.4±0.3
^{95}Zr	64.02	0.6±0.2
^{124}Sb	60.20	39.0±1.5
^{181}Hf	42.39	0.8±0.1
^{182}Ta	114.43	17.1±0.4
^{233}Pa	26.967	9.8±0.4

These results suggest that NAA has been performed, rather than the other activities mentioned above, given the absence of the key isotopes (Sm, Te, Mo, Ir, Au, I) that would be present if the other activities had occurred.

Of particular interest is the presence of ^{233}Pa on the swipe, which may indicate a recent irradiation of Th containing material. Taking into account that electrolytes or films are unlikely to contain Th and that the Th content in biological samples is relatively low (0.1-10 µg/kg), one may assume that NAA has been carried out for soil or sediment samples where the Th content is relatively high (5-50 mg/kg).

3. Modelling NAA of environmental samples

In order to confirm this assumption, computer modelling of the NAA for a soil sample with known elemental composition was performed using the irradiation code 'Nuclear Analysis' [1]. The IAEA reference material SOIL-7 was chosen for the NAA modelling. Table 2 lists the element concentrations, parent stable isotopes, their natural abundance and the indicator radionuclides which would result from the irradiation of the SOIL-7 material. The material contains almost all elements which, when irradiated, would generate radionuclides that can be detected on the swipe. (Compare data in Tables 1 and 2.)

The source data for the model calculations include irradiation parameters and initial inventory. The irradiation parameters used are those of a typical thermal neutron NAA scenario and are as follows [2]:

(a) One group thermal model for neutron flux;

(b) Neutron flux of 3e13 n/cm^2/s at 2200 m/s;

(c) Maximum irradiation time of 5 days; and

(d) Initial inventory (taken from Table 2) assuming a sample mass of 1 g.

Note that (n,2n) and (n,p) reactions were not considered. Therefore the reactions with Mn (^{55}Mn (n,2n)^{54}Mn) and Ni (^{58}Ni (n,p)^{58}Co) were not included in the calculations.

The calculation results (the indicator radionuclide activities) were obtained for various irradiation and cooling times (T_{irr} and T_{cool}), which range from 2 hours to 5 days and from 1 day to 500 days, respectively.

Table 2. Data for NAA analysis of the SOIL-7 reference material.

Element	Concentration, mg/kg	Parent isotope	Natural abundance, fraction	Indicator radionuclide
Sc	8.3	^{45}Sc	1	^{46}Sc
Cr	60	^{50}Cr	0.043	^{51}Cr
Mn	631	^{55}Mn	1	^{54}Mn
Ni	26	^{58}Ni	0.68	^{58}Co
Fe	25700	^{58}Fe	0.028	^{59}Fe
Co	8.9	^{59}Co	1	^{60}Co
Zn	104	^{64}Zn	0.49	^{65}Zn
Zr	185	^{94}Zr	0.17	^{95}Zr
		^{95}Nb		^{95}Nb
Sb	1.7	^{123}Sb	0.43	^{124}Sb
Hf	5.1	^{180}Hf	0.35	^{181}Hf
Ta	0.8	^{181}Ta	1	^{182}Ta
Th	8.2	^{232}Th	1	^{233}Pa
U	2.6			

4. Analysis of relative activities

A direct comparison of the calculated activities for the SOIL-7 sample (A^i_{calc}) with the measured activities for the swipe (A^i_{meas}) is difficult, because they are expressed in different units: A^i_{calc} in Bq/g and A^i_{meas} in Bq/swipe. Therefore the following relative activities are used:

$$A^i_{calc}(rel) = A^i_{calc} / \sum_{i=110}^{N} A^i_{calc}\,, \tag{1}$$

$$A^i_{meas}(rel) = A^i_{meas} / \sum_{i=1}^{N} A^i_{meas}, \tag{2}$$

where N is the number of indicator radionuclides (see Tables 1 and 2). For further consideration the index *rel* is omitted.

A relative deviation Δ_i between A^i_{calc} and A^i_{meas} is calculated from the equation:

$$\Delta_i = (A^i_{calc} - A^i_{meas})/[(A^i_{calc} + A^i_{meas})/2]. \tag{3}$$

One advantage of this definition is that such a relative deviation cannot be larger than 2.0 and smaller than –2.0. This definition is convenient for the analysis of environmental objects where differences between calculated and measured activities can reach several orders of magnitude. A typical definition of relative difference, such as (A^i_{calc}-A^i_{meas})/A^i_{meas}), suggests an asymmetric behaviour: it has the minimum value of –1.0 for the calculated activities close to zero and increases infinitely if the calculated activities are significantly larger than the measured ones.

The average square relative deviation between relative activities for the swipe and the SOIL-7 sample is a Chisquare (χ^2), defined by the equation:

$$\chi^2 = \sum_{i=1}^{N}[\Delta_i]^2/N. \qquad (4)$$

The calculations performed for various irradiation and cooling times showed that the irradiation time does not significantly affect the χ^2 value. Therefore all calculations were carried out for T_{irr} = 12 hours, which is a typical irradiation time for NAA.

The estimated cooling time (the period of time between the end of irradiation and the HRGS measurement) indicates the date of irradiation. The cooling time was estimated by minimizing χ^2 as a function of T_{cool}. Figure 1 shows calculated Δ_i and χ^2 versus T_{cool}, which ranges from 0 to 500 days.

One can see that Δ_i for different radionuclides cross the zero horizontal line where $A^i_{calc} = A^i_{meas}$ at the different cooling times. In particular, for ^{51}Cr the cooling time (approximately 20 days) is significantly lower than that for ^{233}Pa (approximately 90 days) and that for ^{59}Fe and for ^{65}Zn (approximately 160 days). The latter is significantly lower than the cooling time for ^{95}Zr, ^{181}Hf and ^{60}Co (330-440 days). For several radionuclides (^{124}Sb, ^{46}Sc, ^{182}Ta) the Δ_i does not cross the zero line at all. Such a wide scattering of observed values might indicate that the elemental composition of the SOIL-7 sample differs significantly from the material on the real swipe.

One should note that the details of this effect, i.e. how this difference affects the cooling time, is still not clear and requires additional investigation and modelling. A wide scattering of observed values for the estimated cooling times might also occur if the real swipe has traces of NAA performed for different materials and during different time periods.

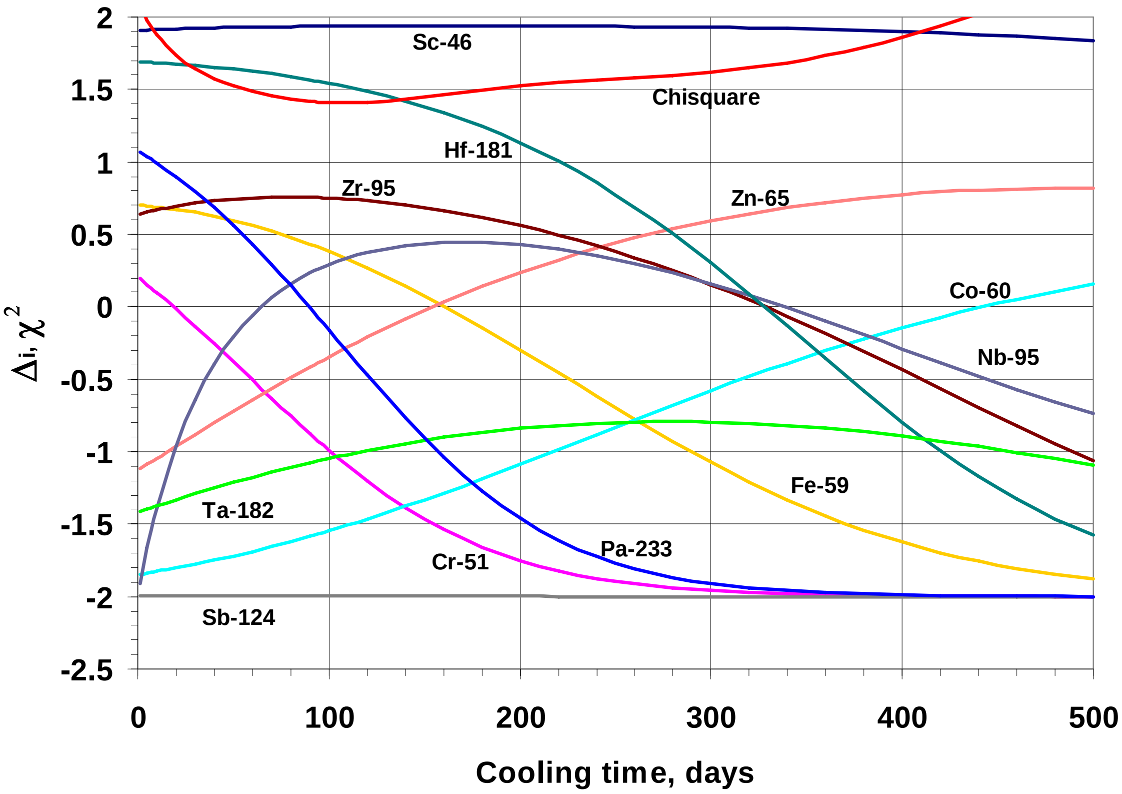

FIG. 1. Relative deviations of the calculated and measured activities for the example scenario as functions of cooling time.

The χ^2 has a broad minimum at T_{cool} = 120 days. The χ^2 as a function of T_{cool} is quite sharp for low values of T_{cool} and "flat" for large values of T_{cool}. One can conclude that T_{cool} is more than 40 days and less than 400 days.

5. Analysis of activity ratios

As previously mentioned, a wide scattering of the cooling time estimated (from 40 to 400 days) might be due to significant difference between the elemental composition of the measured sample and the SOIL-7 material. In order to reduce this difference one can use the activity ratios for some elements whose concentrations are correlated.

Correlations between element concentrations are used in geology and geochemistry for identification of different types of the earth's crust [3] and for creating fingerprints or geochemical signatures of shale [4]. These correlations are observed due to the basic processes of the earth's evolution. Sediments, soils and other environmental objects created in the evolution of the earth's crust material (mainly rocks) maintain original element ratios for certain elements.

The $^{95}Zr/^{181}Hf$ activity ratio is one of the best candidates for estimation of T_{cool} of the environmental samples analysed by NAA. Zr and Hf behave like geochemical twins because they have similar ionic radii and the same valence. The Zr/Hf mass ratio is approximately 34 for the earth's crust samples and this value varies from 27 to 45 for a wide variety of samples [5]. The Zr/Hf mass ratio is approximately 36 for the SOIL-7 sample.

The relation between Zr/Hf element ratio m(Zr)/m(Hf) and $^{95}Zr/^{181}Hf$ activity ratio $A(^{95}Zr)/A(^{181}Hf)$ is as follows:

$$A(^{95}Zr)/A(^{181}Hf) = m(Zr)/m(Hf)*k*\exp[(\lambda_{Hf}-\lambda_{Zr})*T_{cool}], \qquad (5)$$

$$k = Ab(^{94}Zr)/Ab(^{180}Hf)*\sigma_c(^{94}Zr)/\sigma_c(^{180}Hf)*M_0(^{180}Hf)/M_0(^{94}Zr), \qquad (6)$$

where λ_{Hf} and λ_{Zr} are decay constants of ^{181}Hf and ^{95}Zr respectively, $Ab(^{94}Zr)$, $\sigma_c(^{94}Zr)$, $M_0(^{94}Zr)$ are natural abundance, neutron capture cross section and atomic mass of ^{94}Zr. Similar notations are used for the ^{180}Hf parameters.

In contrast with the mass ratio, the activity ratio is time dependent. In particular, the $^{95}Zr/^{181}Hf$ activity ratio decreases by a factor of 2 in 126 days. However, if this ratio can be measured, the cooling time can be obtained from equation 5 above.

The $^{181}Hf/^{182}Ta$ activity ratio can be another indicator of NAA of environmental samples (the Hf/Ta mass ratio is about 6 for the SOIL-7 material). All three elements -- Zr, Ta and Hf -- are considered to occur in constant ratios in the planetary body [5].

The cooling time calculated from equation 5, using the Zr/Hf and Hf/Ta mass ratios from Table 2 and the $^{95}Zr/^{181}Hf$ and $^{181}Hf/^{182}Ta$ activity ratios measured on the real sample, is about 290 and 315 days correspondingly.

Another estimation of the cooling time was performed using the calculated $^{95}Zr/^{181}Hf$ and $^{181}Hf/^{182}Ta$ activity ratios for the SOIL-7 sample by the Nuclear Analysis code. Figure 2 shows relative deviations of the calculated activity ratios for SOIL-7 from the measured activity ratios on the swipe versus the cooling time. One can see that $^{95}Zr/^{181}Hf$ activity ratio gives T_{cool} of approximately 330 days. The $^{181}Hf/^{182}Ta$ activity ratio gives T_{cool} of approximately 410 days. Both estimates are generally consistent with the estimates obtained from equation 5 above.

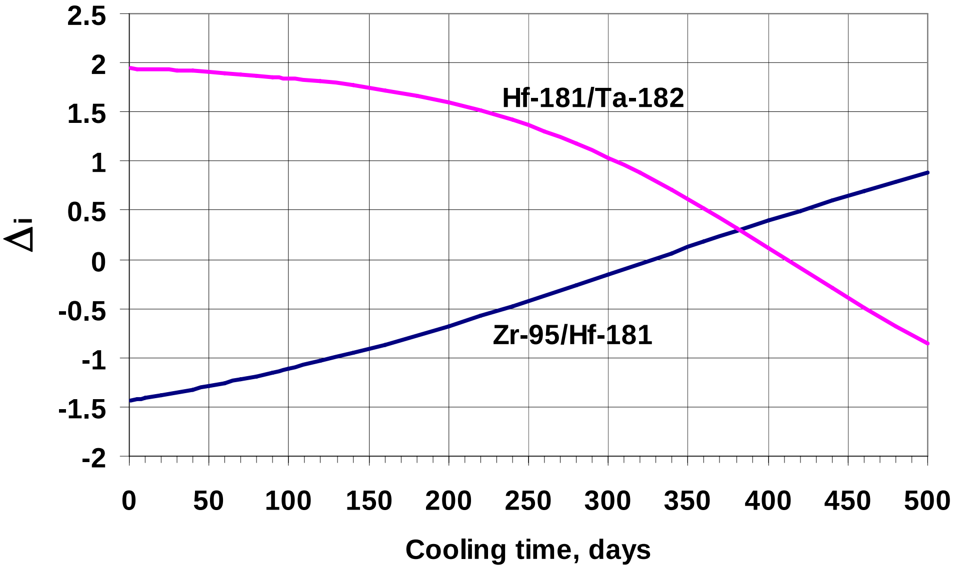

FIG. 2. Relative deviations for the calculated and measured activity ratios.

6. Analysis of uranium isotopic composition

The Th/U ratio also can be used as an indicator of NAA of environmental samples, which originate from the earth's crust (rocks, sediments, soils, etc). The irradiation of such samples generates U isotopes, particularly, ^{233}U and ^{236}U as follows:

$$^{232}Th + n \Rightarrow {}^{233}Th(0.37h) \Rightarrow {}^{233}Pa(27d) \Rightarrow {}^{233}U \text{ (long-lived)} \qquad (7)$$

$$^{235}U + n \Rightarrow {}^{236}U \text{ (long-lived)} \qquad (8)$$

Assuming that all ^{233}Pa has decayed into ^{233}U (4-5 months after irradiation) the $^{233}U/^{236}U$ ratio in a linear approximation is proportional to the Th/U ratio in the sample:

$$^{233}U/^{236}U = 10.5 * Th/U. \qquad (9)$$

The proportionality coefficient in (9) does not depend on the neutron flux and irradiation time. Apparently it is a ratio of the neutron capture cross sections of ^{232}Th and ^{235}U divided by the ^{235}U enrichment (0.0072). The Th/U mass ratio for samples of the earth's crust is about 3.3 [6]. It gives an estimate of the $^{233}U/^{236}U$ ratio of approximately 35. If the measured $^{233}U/^{236}U$ mass ratio is significantly higher than 35, then irradiation of Th enriched targets may have taken place. A ratio lower than 35 could indicate a relatively short cooling time. If the $^{233}U/^{236}U$ ratio is lower than 10, one can conclude that the irradiated sample does not originate from the earth's crust.

One should note that the $^{233}U/^{236}U$ ratio cannot be measured by HRGS and requires another analytical method, e.g. mass spectrometry.

7. Conclusion

The modelling of typical irradiation scenarios used for NAA of material with known elemental composition, followed by the comparison of the calculated activities with HRGS measurement results of real swipe samples, allow for the estimation of the irradiation date and type of irradiated material present on these samples.

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Cluster analysis for TIMS particle data analysis

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Abstract. Environmental samples taken for IAEA safeguards are analyzed in laboratories to determine characteristics of uranium and plutonium bearing particles. Analytical methods include fissionable particle detection by the fission track method and subsequent analysis of selected particles with a thermal ionization mass spectrometer (TIMS). Typically, laboratory analysis of each particle provides information on four uranium isotopes (^{234}U, ^{235}U, ^{236}U, ^{238}U) and, if detected, on four plutonium isotopes (^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu). These attributes form multi-dimensional patterns that are characteristic of particular nuclear processes, such as burn-up of nuclear fuel, enrichment of uranium or mixing of two or more nuclear materials. One way to see the main characteristics of the particle data is to determine groups (i.e. clusters) of similar particles. Cluster analysis is also used to reduce the data sets and to compare different data sets. Thus, one can determine if the particles in different samples have similar patterns, which may suggest the same origin of nuclear materials in the samples. New analysis software for cluster determination was developed to assist the needs of environmental sampling data evaluation.

1. Introduction

For cluster analysis, two alternate methods were selected for different needs. To reduce the data set one can try to join together in a cluster as many particles as possible. The method of quality threshold (QT) fits this purpose well. In this method two particles are joined if their individual uncertainties are overlapping. The new cluster is then regarded as a new particle and the process is run iteratively until no changes occur in the data set of single particles and clusters. To find the characteristic values in the data set a method of 'overlapping clusters' was chosen. In this method all the particles are allowed to group together within their uncertainties and in the second pass the particle is assigned only to a cluster where it best belongs.

Sometimes the nuclear process and mixing of the materials are so complicated that the user may have more information than just what the numbers may suggest. Therefore, one additional feature was added to the software: the user may specify clusters interactively.

Other important needs for the evaluation are to be able to show the results graphically, to compare the results to reactor burn-up data and to uranium enrichment calculations and to perform decay correction to isotopic composition of particles. Another important feature in the developed software is to calculate the age of plutonium due to ^{241}Pu decay.

2. Reactor burn-up and cluster analysis

When uranium and plutonium are burned in a reactor their isotopic composition changes. Clusters of particles found on spent fuel samples may indicate specific burn-up of a fuel element. The verification process includes analysis of consistency between the operator declared fuel characteristics and detected particles. Reactor codes like Origen[1] are used to model the behavior of actinides as a function of burn-up (see Figure 1).

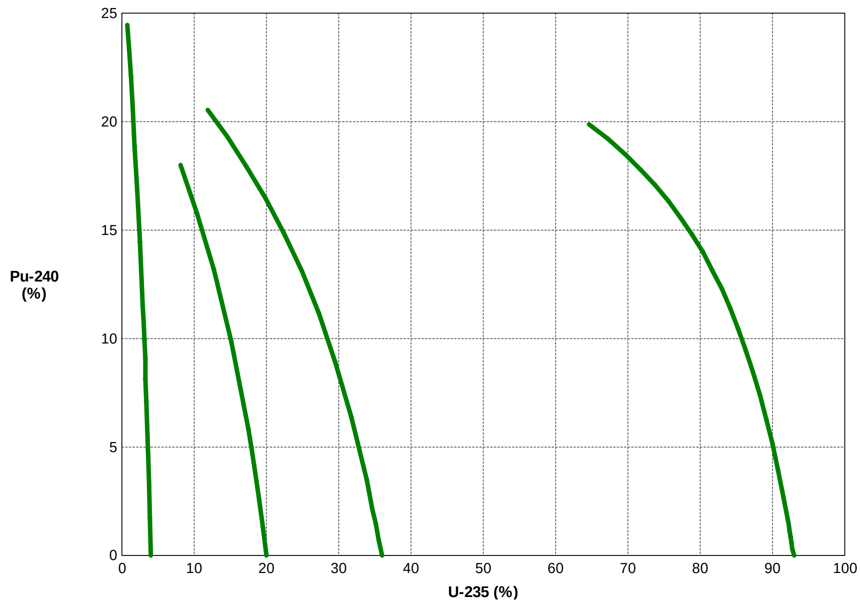

FIG. 1. Model of ^{240}Pu and ^{235}U behaviour (atom%) for four different fuel types (93% HEU, 36% HEU, 20% LEU and 4% LEU). In the ideal case all the particles are located close to the corresponding burn-up curve and clusters of particles represent specific burn-up of irradiated fuel.

A sample taken from spent nuclear fuel may contain substantial amounts of all uranium and plutonium isotopes that are modeled using the cluster analysis software. The method of overlapping clusters fits well the need for analysis of these kinds of results, since the purpose is to find the characteristic clusters that can explain the irradiation history of the material. The isotopic composition of a detected cluster can be used to verify if the identified particle characteristics are in line with the declared fuel irradiation history. The verification process includes also internal consistency testing: all the plutonium and uranium isotopes inside a cluster should be consistent with a single burn-up value of nuclear material.

The age of the plutonium is an essential attribute for spent fuel and reprocessing verification. Due to its 14 years half-life, ^{241}Pu provides a nuclear clock for cooling time estimation. Once the plutonium isotopic composition is calculated with reactor codes, the decay of ^{241}Pu can be calculated with the cluster analysis software to see how old the detected plutonium is.

3. Enrichment of uranium and cluster analysis

In a normal centrifuge cascade, samples may be taken from feed, product or tail stream. These are the locations where the processed materials may be dispersed to the environment. Therefore, sets of clusters found in the swipe samples taken at the enrichment plant may represent feeding materials, products of the enrichment and depleted tail. If the intention is to produce high enriched uranium (HEU), usually a set of cascades is connected in series, so that the higher cascade is using the product of the lower cascade as a feed. During the ^{235}U enrichment, abundances of the minor isotopes (^{234}U and ^{236}U) are also elevated. Therefore, in addition to ^{235}U, the minor isotopes play an important role in the characterization of the process. Computer codes such as MSTAR[2] are used to model the enrichment process (see Figure 2.).

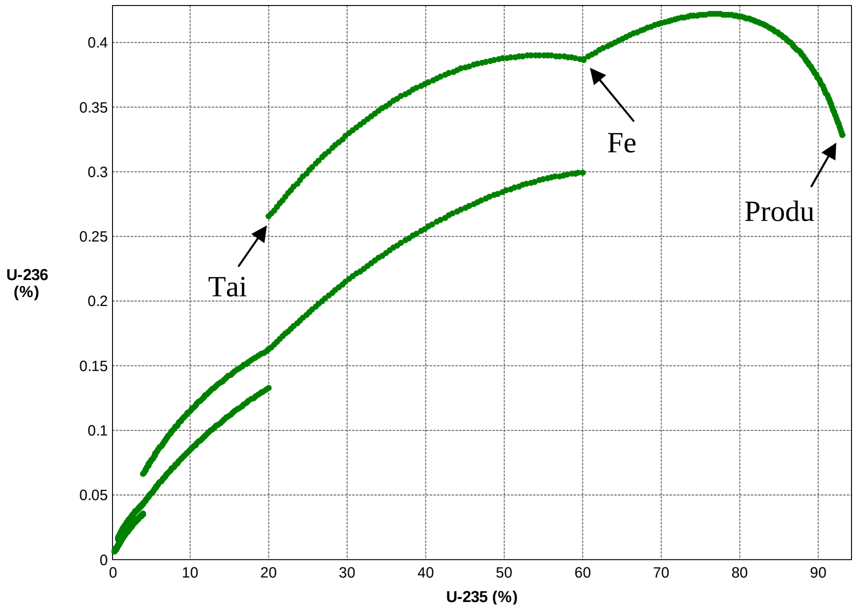

FIG. 2. Model of four uranium enrichment cascades connected in series. If recycled uranium is used as a feed material, ^{236}U will accumulate together with ^{235}U in the product. Feed, product and tails of the top cascade are highlighted in the figure. Clusters of found particles with isotopic composition close to these points may confirm the design of this cascade.

A sample taken from enrichment facility can be very rich in uranium content. Particle analysis may reveal thousands of uranium containing particles with individual isotopic composition. In this case, the first step of data interpretation is to find the most frequent combinations of isotopic abundances. The method of "overlapping clusters" fits well this purpose.

4. Hot cell samples and cluster analysis

Nuclear and other radioactive materials are processed in hot cells. These materials usually leave traces inside the cell. The mixing of various plutonium- and uranium-containing materials is possible on the surfaces outside and inside the hot cells where swipe samples are taken. Therefore, evaluation of the analysis results on a swipe sample taken from hot a cell has to consider also the mixing of detected materials.

The particles found in the environmental samples often consist of mixtures of two or more materials. A simulated example of an evaluation case is presented at Figure 3. Seven different kinds of materials were simulated to show typical particles consisting of, among other things, ^{236}U and ^{235}U. In this model case an explanation of depleted uranium (DU) particles with 0.20% and 0.60% ^{235}U was required. After cluster analysis, the 0.20% DU particle fits nicely inside the cluster of 0.2% DU. The 0.6% DU particle seems to fit the mixing lines of various clusters.

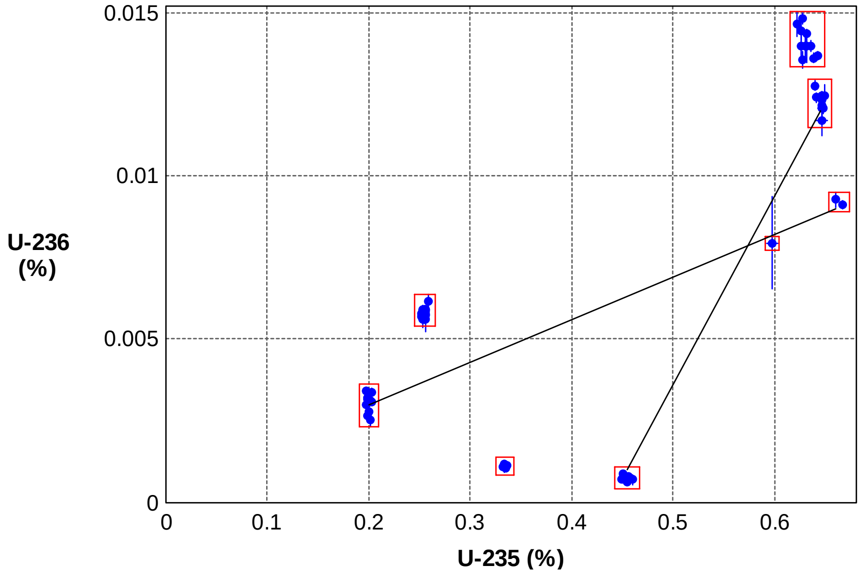

FIG. 3. An example of evaluation with cluster analysis. The task was to find an explanation for DU particles with 0.20% and 0.60% ^{235}U. 0.20% DU particle seems to fit the existing cluster whereas the 0.60% DU particle appears to fit mixing lines of various clusters.

The purpose of cluster analysis in this case is to find materials originally used in the hot cell. QT-clustering gave a direct answer to the question. After the clusters were identified, it was easy to see that the materials in question fit the known materials or that they were the result of mixing of two or more materials.

5. Conclusion

Cluster analysis can be used to characterize nuclear processes relevant to the nuclear material safeguards purposes. The automated or user guided cluster analysis makes the method of finding characteristic sets of particles easier and provides tools to reduce the data for analysis purposes. The graphical presentation of clustering is essential for verifying the correctness of the automated process. The cluster analysis software has now been tested on safeguards samples and the process gives consistent results. Sample sets including thousands of particles have been used for testing.

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Particle analysis for uranium isotopics on swipe samples using new generation Cameca IMS 7f SIMS supported by SEM automated uranium detection

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Abstract. This study examines the detection and the analysis for uranium isotopic composition in micrometer size particles extracted from swipes using scanning electron microscopy (SEM) and secondary ion mass spectrometry (SIMS) techniques. The use of low sputtering rates and high objective magnification allows sensitivity improvement of the SIMS analysis. Accurate 234/238 and 235/238 uranium isotopic ratios are determined in 1 µm diameter uranium oxide particles with relative precision of ± 6% and ± 0.7% (1σ), respectively.

1. Introduction

Recent safeguard issues revealed a need for increasing swipe samples analysis capability within the Agency's Network of Analytical Laboratories (NWAL). Our laboratory is qualified since 2001 to analyze uranium isotopics in particles using the Fission Tracks / Thermo-Ionization Mass Spectrometry method (FT-TIMS). In addition, we recently acquired a Secondary Ion Mass Spectrometer (SIMS) in order to broaden particle analysis capabilities in the laboratory. The SIMS technique enables to lower the response time for urgent analysis and to maintain a swipe sample analysis capacity even when our reactor for neutron irradiation is not available.

This presentation focuses on the results that have been obtained on particle analysis for uranium isotopes using our new generation Cameca IMS 7f. Particles are extracted from swipe samples. In this study, all sample processing steps have been examined with the view to improve the reliability of the results and to shorten analytical response time.

2. Particle extraction and sample mounting

All sample preparations are conducted in a class 10 clean room according to the method used by most laboratories that participate in the NWAL [1]. Particles are transferred from swipe to ethanol suspension by ultrasonification. The suspension was then evaporated to ~ 1.5 mL. Half volume was deposited onto a heated (60°C), diameter 25 mm carbon disk by means of a 100 µL micro-syringe allowing the output of 10-20 µL droplets. Finally, preparations are baked at 400°C for 2 hours to volatilize organic compounds from sample surface.

3. Uranium-bearing particle detection

The main difficulty in particle detection arises because sample screening software which are commonly used for SIMS automated uranium-bearing particle search [1] still have to be updated to a version compatible with the IMS 7f software (e.g. *P-search* by Evans Analytical).

3.1. SEM capability for automated detection of uranium-bearing particles

In this study, the automated detection of uranium-bearing particles is performed using a FEI XL 30 environmental SEM fitted with an EDAX system. An adaptation of the *Gun Shot Residue* forensic software allows the automatic search for uranium-containing particles using back-scattered electron image analysis and qualitative micro-analysis of major elemental composition by energy dispersed X-ray spectrometry. In addition, secondary electron images of uranium-containing particles can be acquired in order to characterize their morphology. An overnight GSR run may investigate a ~ 1 cm^2 deposition area, detecting with a high probability all uranium-bearing particles with diameter > 1 µm. The GSR program provides a listing of uranium-bearing particle coordinates relative to the SEM sample stage.

3.2. SEM to SIMS relocation of particles

Sample mountings are equipped with an internal reference consisting of 2 aluminum foil triangle pieces. This internal reference enables the determination of parameters in the transformation of coordinates relative to the SEM stage, to coordinates relative to the SIMS sample stage according to triangulation method [2] with a precision better than 50 µm. Then, sample surface is rastered with a 70 nA O_2^+ primary ion beam over a 400 µm×400 µm area centered on the calculated position of the particle of interest. In all the preparation analyzed, it appears that 100 % of the uranium-bearing particles previously detected by SEM could be pointed out on ion images acquired at mass 238 using microprobe mode.

Compared to manual SIMS detection, this method presents some advantages: the SEM/EDX detection is non-destructive (the whole particle is available for the IR measurement), non-susceptible to isobaric interferences, more efficient, faster, and provides some additional relevant information on individual particles (e.g. volume, morphology, and major elemental composition).

4. SIMS analysis of uranium isotopic ratios

4.1. Sensitivity improvement

Sensitivity is critical in particle analysis, considering the very small amounts of uranium to be analyzed (~ a few pg). Analyses of particles for isotopic ratios using monocollector small radius magnetic sector SIMS such as Cameca IMS7f instrument are commonly performed using high sensitivity instrumental configuration (i.e. low mass resolution, maximal energy pass band, large contrast diaphragm and field aperture) [1]. In this study, we investigate the influence of primary and secondary optical settings sensitivity during uranium isotopic analysis in synthetic particles of natural uranium oxide with diameter ~ 3 µm.

4.1.1. Primary ion beam parameters

It is observed that sputtering for 1 hour by static primary ion beams (2, 5, and 20 nA intensity) does allow reaching complete consumption of particles. According to recommendations in [4], one has thus to consider "practical sensitivity" as the appropriate measure of sensitivity (i.e. for uranium, the ratio between the number of detected secondary ions of uranium and the number of primary ions hitting the sample).

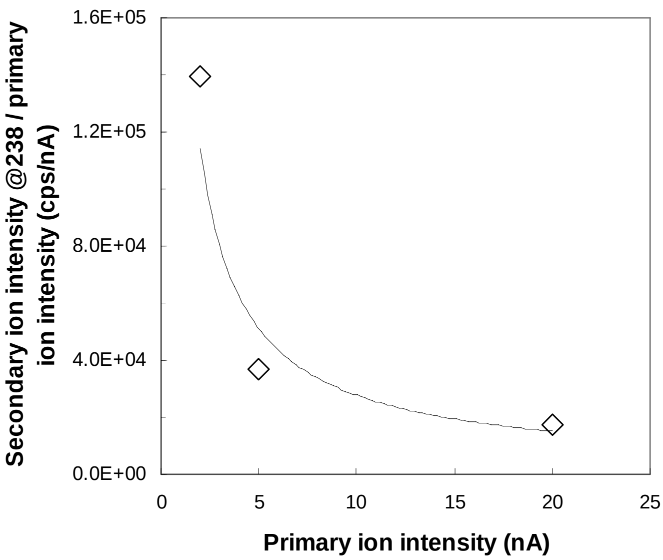

FIG 1. Practical sensitivity as a function of primary ion intensity for 1 hour sputtering on 3 µm diameter uranium oxide particles. The value given for secondary ion intensity corresponds to the average over the experiment.

It appears that practical sensitivity is one order magnitude higher at low sputtering rate (primary ion intensity of 2 nA vs 20 nA). This result may indicate that low sputtering rate enhances secondary ionization rate of particle uranium. It is observed that sputtering for 1 hour by static primary ion beams (2, 5, and 20 nA intensity) does allow reaching complete consumption of particles. According to recommendations in [4], one has thus to consider "practical sensitivity" as the appropriate measure of sensitivity (i.e. for uranium, the ratio between the number of detected secondary ions of uranium and the number of primary ions hitting the sample). It appears that practical sensitivity is one order magnitude higher at low sputtering rate (primary ion intensity of 2 nA vs 20 nA). This result may indicate that low sputtering rate enhances secondary ionization rate of particle uranium.

4.1.2. Transfer optics configuration

Besides, sensitivity improvement is obtained by improving transmission characteristics (i.e. the ratio between the number of ions detected and the number of ions formed). In this study, the configuration of transfer optics has been optimized according to [5].

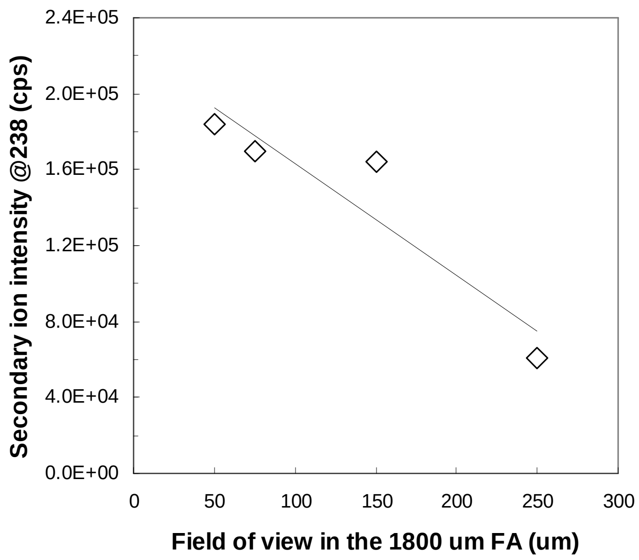

FIG. 2. Secondary 238-uranium ion intensity measured on 3 µm diameter uranium oxide particles as a function of transfer optics configuration. The values given on X-axis axis correspond to the diameter of the field of view fitting the 1800 µm field aperture (FA). Actually, this value is inversely proportional to ion image magnification.

The 2 transfer electrostatic lenses are set such as to maximize the magnitude of the ion image at the plane of the field aperture. This has the effect of reducing ion beam crossover diameter and thus decreasing the part of secondary ions crosscut at the entrance slit and contrast diaphragm plane.

4.1.3. Spectrometer settings and optimization of mass peak shape

Besides, this study points out that analyzer setting is a key point for high sensitivity analysis. The trajectories that do not cross the field aperture and the entrance slit at their respective center will not be focused at the exit slit plane. The use of large field aperture (1800 µm) and large energy slit (125 eV) during high sensitivity analyses increase the blurring of the entrance slit image at the exit slit plane due to second order aberrations. When the coupling between the spectrometer lens and the electrostatic sector is not optimized, we commonly observed that the transmission to the analyzer of ions with > 100 eV energy range leads to strong asymmetry of aberrations at the exit slit plane. This aberration feature appears to be correlated with dissymmetric mass peak geometry which is not suitable for the determination of precise isotopic ratios.

4.2. Repeatability, reproducibility, and accuracy of measurements

In order to evaluate the reliability of the measurements, we acquired synthetic particle samples with diameter calibrated to ~1 µm that were prepared at ITU from NIST certified reference materials (CRMs): U-010, U-020, U-030, and U-500 [3].

4.2.1. Repeatability and reproducibility

We use primary ion beam intensity of 3 nA and transfer optics of 50 µm, corresponding to higher practical sensitivity. Typical secondary uranium ion intensities are for instance in U-010 sample: 3×10^5 cps, 3×10^3 cps, 20 cps at mass 238, 235, and 234, respectively. Counting times are set at 0.5 s, 1 s, and 2 s, respectively. An analysis run is divided in 12 cycles. Under this conditions, a repeatability (i.e. the relative standard deviation of the mean over the isotopic ratios measured during an analysis run) ranging between 0.5% and 1.5 % and between 2% and 8% is obtained on 235/238 and 234/238 ratios, respectively. This value is in rather good agreement with the theorical relative standard deviation of the mean calculated from the Poisson's law (between 0.3% and 0.8% and between 3% and 7% on 235/238 and 234/238 ratios, respectively).

Point-to-point reproducibility is estimated from the standard deviation over isotopic ratios measured over different particles (Fig. 3, Table 1). The values obtained are similar to the within-run repeatability measured on the 4 samples. This indicates that the analytical conditions can be considered as reproducible from particle to particle in the synthetic samples.

Moreover, a good agreement is also observed between SIMS isotopic measurements and certified values for 234/238 and 235/238 (Table 1). Average instrumental mass fractionation factors (ie. the ratio between measured and certified isotopic ratios) over the 4 samples are 1.0083 ± 0.0052 and 1.036 ± 0.021 for 235/238 and 234/238 ratios, respectively. This means that a correction for instrumental mass fractionation has to be applied to SIMS measurements in order to obtain accurate results. It should be noted that this correction is an additional source of uncertainty of the final results. Error propagation calculations give typical relative uncertainty (1σ) on corrected SIMS measurements ranging between ± 0.5% and ± 1.5% (average: ± 0.7%) and between ± 3% and ± 13% (average: ± 6%), for 235/238 and 234/238 ratios, respectively.

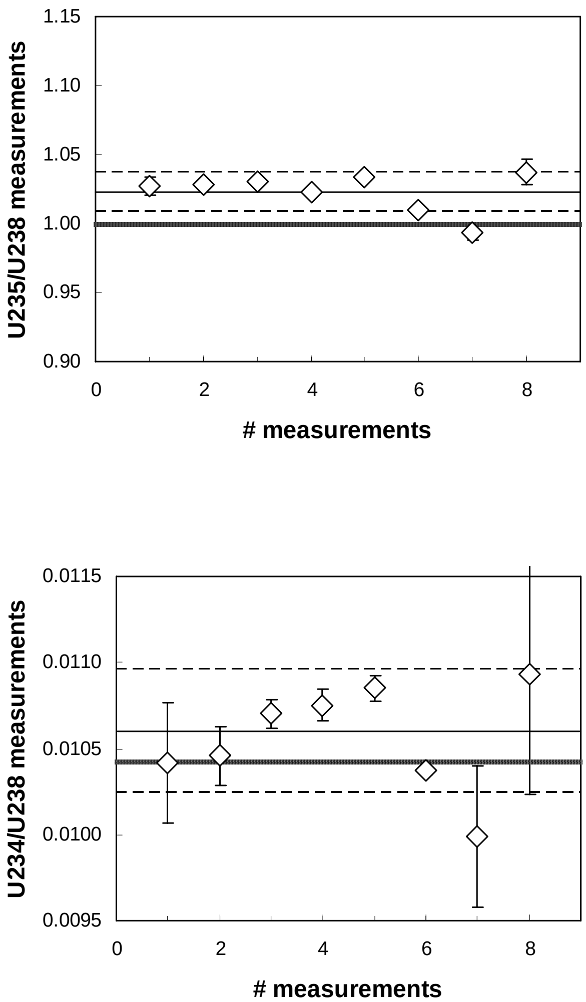

FIG. 3. Uranium isotopic ratios measured on 8 different 1 µm diameter particles of NIST certified reference materials (CRM) U-500. All the analysis are made using the same analytical settings. Error bars correspond to the within run repeatability. Solid and dashed lines indicate the mean and the standard deviation over the 8 measurements, respectively. The bold lines represent the certified values for 235/238 and 234/238 uranium isotopic ratios.

	U-010	U-020	U-030	U-500
$^{235}U/^{238}U$ measurements[a]	$(1.0145 \pm 0.0050) \times 10^{-2}$	$(2.0723 \pm 0.016) \times 10^{-2}$	$(3.167 \pm 0.049) \times 10^{-2}$	1.0230 ± 0.0037
Certified Value	$(1.01400 \pm 0.00051) \times 10^{-2}$	$(2.06872 \pm 0.00057) \times 10^{-2}$	$(3.142960 \pm 0.00059) \times 10^{-2}$	$(9.997 \pm 0.010) \times 10^{-1}$
Relative Std Error[b] (1σ, within run)	0.50%	0.76%	1.6%	0.36%
Reproducibility[c] (1σ, point-to-point)	$\pm 0.0072 \times 10^{-2}$	$\pm 0.0098 \times 10^{-2}$	$\pm 0.089 \times 10^{-2}$	± 0.014
Number of analyzed particles	n=7	n=6	n=9	n=8
Average IMF factor[d]	1.0005 ± 0.0050	1.0020 ± 0.0076	1.008 ± 0.016	1.0233 ± 0.0039

	U-010	U-020	U-030	U-500
$^{234}U/^{238}U$ measurements[a]	$(5.71 \pm 0.65) \times 10^{-5}$	$(1.93 \pm 0.18) \times 10^{-4}$	$(1.14 \pm 0.24) \times 10^{-4}$	$(1.060 \pm 0.022) \times 10^{-2}$
Certified Value (1 σ)	$(5.465 \pm 0.025) \times 10^{-5}$	$(1.7683 \pm 0.0015) \times 10^{-4}$	$(1.9605 \pm 0.0016) \times 10^{-4}$	$(1.0422 \pm 0.0013) \times 10^{-2}$
Relative Std Error[b] (1σ, within run)	± 5.5%	± 8.2%	± 13%	± 2.2%
Reproducibility[c] (1σ, point-to-point)	$\pm 0.65 \times 10^{-5}$	$\pm 0.32 \times 10^{-4}$	$\pm 0.32 \times 10^{-4}$	$\pm 0.036 \times 10^{-2}$
Number of analyzed particles	n=7	n=6	n=9	n=8
Average IMF factor[d]	1.046 ± 0.057	1.09 ± 0.10	9.92 ± 0.058	1.017 ± 0.021

[a] The uncertainty corresponds to the standard deviation of the mean over measurement cycles (i.e. within-run repeatability).
[b] Relative standard deviation of the mean over measurement cycles (i.e. within-run relative repeatability).
[c] Standard deviation over measurements on n different particles (i.e. point-to-point reproducibility).
[d] Instrumental Mass Fractionation (IMF) equals to the ratio between measured and certified isotopic ratios. The value given corresponds to the mean ratio over the n measured particles. The associated uncertainty corresponds to the standard deviation over the n measurements.

Table 1. Results obtained on 1 µm diameter particles of NIST certified reference materials (CRM) U-010, U-020, U-030, U-500.

5. Conclusions

The combination of SEM and SIMS techniques allows the detection of uranium-bearing particles overnight and the analysis for isotopic ratios of 40 particles daily. SIMS technique allows the determination of accurate 235/238 and 234/238 uranium isotopic ratios on 1µm diameter particles with a relative precision (1σ) of about ± 0.7% and 6%, respectively. The main drawback of this technique with regards to fission tracks / TIMS method is that it is not sensitive to ^{235}U-enrichment of the detected particles. As a consequence, no priority can be drawn among the particles to be analyzed for isotopic ratios.

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New developments and methodology for actinide measurements at ultra trace levels using ICP-MS

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Abstract. For safeguards purposes, there is a real need for accurate and reliable measurements of plutonium isotopes at the lowest level in environmental samples. It is of prime necessity to detect the ultra-trace levels with the best confidence in order to avoid any false positive or negative detection. To do this, an analytical methodology devoted to plutonium measurements at femtogram levels in environmental samples has been optimised. This methodology is based on the combination of an efficient radiochemistry and of a very sensitive ICP-MS detection. This work first identifies and quantifies the polyatomic interferences that occur at m/z = 239. Heavy elements like mercury can generate $^{199}Hg^{40}Ar^{+}$at a rate ranging from 10^{-4} to 10^{-3}. These interfering elements concentrations in purified solutions have been determined at trace ($pg.ml^{-1}$) levels but their contributions need anyway to be corrected. Then, our method for determining plutonium detection limits on real samples is described. It is based on the combination of standard deviations over uranium hydride, abundance sensitivity, impurities from ^{242}Pu isotopic dilution tracer corrections, and standard deviation over count rates of selected neighbouring (241-247) masses acquired during the measurements of the samples. The specific radiochemistry, devoted to ultra-trace measurements is presented. The different sources of contamination have been quantified. The crucial step for uranium elimination from purified solution has been identified to be the rinsing of anionic chromatography column with adequate volume of 8M HNO_3. Micro-nebulisers can be used, down to $50\mu l.min^{-1}$ in operational conditions. Metrological settings of ICP-MS have to be optimised, especially dead time and mass bias correction. Finally we investigated the potentialities of the coupling of femtosecond laser ablation system and ICP-MS as an alternative to TIMS with respect to particle analysis. Preliminary results appear to be very promising because LA-ICP-MS is sensitive, rapid and easy to use.

1. INTRODUCTION

Both ultra-trace actinide measurements and accurate isotopic ratio determinations are needed for safeguards purposes, because they can deliver precious information relative to the occurrence and the nature of a process. Environmental samples generally include very low masses of the elements of interest. The challenge is to determine those ultra-traces with the best degree of confidence to avoid any false detection or missing their occurrence.

For more than 10 years, mass spectrometry techniques take advantage over radiometry techniques for the measurements of long lived radionuclides. Starting from 1996, many papers [1-8] report detection limits that are in the femtogram range and sometimes even below for some actinide isotopes, particularly ^{239}Pu and ^{240}Pu, when using ICP-MS (ICP-QMS, ICP-SFMS and MC-ICP-MS). The lowest reported [7] detection limit, as low as 0.05 femtogram of ^{239}Pu, relates to an instrumental detection limit. Theoretically, instrumental detection limits could be reached on real samples provided a perfect radiochemistry could be performed, prior to the measurement. This means that the radiochemistry should almost completely remove the matrix to avoid any losses in sensitivity, should also remove uranium at such levels that uranium hydrides UH^{+} could be negligible when measuring ^{239}Pu isotope and finally should sufficiently eliminate the isotopes like lead or mercury that could form isobars at m/z = 239 when combining with argon, chlorine or oxygen in the plasma. Generally speaking the radiochemistry has to be conducted in the cleanest conditions in order to avoid any contamination from other samples, materials, glassware, reagents, etc. This is true for uranium isotopes measurements, as well as for plutonium measurements. A very clean and efficient radiochemistry also gives the opportunity to use micro-nebulisers as introduction systems which allows a considerable reduction of the absolute detection limit on real samples. Theoretically, detection limits lower than 0.05 femtogram could be reached on real samples.

This work first identifies and quantifies the interferences that occur at m/z = 239 and alter the background noise, originating from isotopes ranging from m/z = 180 to m/z = 210. The quantities of interfering isotopes in purified solutions from real samples have been determined. Calculations of the operational detection limits, taking into account those interferences, are detailed. The radiochemistry used for purifying the samples is presented with an emphasis on the different sources of contamination and a way to overcome them. The performances of different micro-nebulisers are examined with respect to sensitivity. Furthermore, procedures used for the measurements of real samples are described, focusing on mass bias and detector dead time corrections.

Besides, as our laboratory is also involved in particle analysis as a NWAL member, using the Fission Track-TIMS technique, we also search for alternative analytical technique to TIMS. Taking into account that ICP-MS is, intrinsically, more sensitive than TIMS and, consequently, better suited for ultra-trace measurement, the potentialities of Laser Ablation as an introduction system, coupled with ICP-MS is evaluated for direct sampling on micrometer size uranium particles. Preliminary trials performed using a femto-second laser coupled with an ICP-MS are presented. Global sensitivity of the system is compared to the traditional liquid introduction system and the capacity of measuring precise isotopic ratios is discussed.

2. Experimental

2. 1. Identification and quantification of polyatomic interferences at m/z = 239

At the end of the mass range, the actinides have typically been considered far-removed from the notorious interferences that tend to plague the lighter elements. Data already reported [2], indicate that interferences from $^{207}PbOO^+$, $^{208}PbOO^+$ and $PbCl^+$ were negligible. But, given that ICP-MS detection limits in the sub-femtogram range are now attainable, all types of potential interferences must receive additional consideration. Even when sample matrices are reasonably clean and care has been taken to minimize oxides during tuning, measurements made near the instrumental detection limit are especially susceptible to over-estimation due to polyatomic interferences.

We quantified some of those interferences at m/z = 239 using a double-focusing sector field ICP-MS ("Axiom SC", VG Elemental, Winsford, Cheshire, UK), already described [6-7]. Many elements ranging from tungsten to bismuth were studied. Known quantities of standard solutions containing those elements were injected in the ICP-MS and the signal at m/z = 239 was monitored. Formation rates of polyatomics are reported in table 1. They are defined as the ratio of the count rate of the polyatomic specie on the count rate of the heavy element that gives rise to the polyatomic.

Polyatomic species	*Formation rates*
$^{207}Pb^{16}O_2^+$	$(5 \pm 2)\times10^{-10}$
$^{204}Pb^{35}Cl^+$	$(2.0 \pm 0.8)\times10^{-10}$
$^{209}Bi^{14}N^{16}O^+$	$(3.0 \pm 1.5)\times10^{-10}$
$^{191}Ir^{16}O_3^+$	$(1.5 \pm 0.8)\times10^{-6}$
$^{199}Hg^{40}Ar^+$	$(6 \pm 3)\times10^{-4}$

Table 1. Formation rates of polyatomic ions involving lead, bismuth, irridium and mercury and arising at m/z = 239 for the ICP-SFMS "Axiom". Uncertainties are given with a coverage factor of 1.

Of course, the values reported in table 1 could vary, depending on the instrumental parameters (gas flow rates, plasma power, high voltages for ionic lenses…) but they indicate that interferences originating from lead and bismuth have a low formation rate. The most awkward interference for low level plutonium measurements is $^{199}Hg^{40}Ar^+$. Its formation rate is even higher than that of $^{238}UH^+$, already reported to be close to 3×10^{-5} [6-7]. Consequently, the mass of mercury in the samples has to be carefully determined, prior to plutonium determination.

2.2. Corrections of interferences for the measurement of Pu at ultra-trace level

The analytical method developed in our laboratories for plutonium measurements takes into account those polyatomic ions formation rate. Masses starting from 180 to 210 are systematically scanned in order to determine the concentrations of interfering elements and calculate the corrections at the mass corresponding to the isotopes to be detected. A scan obtained from a purified swipe sample from IAEA is given in figure 1. For Pu measurements at picogram levels, the contribution of those interferences are negligible but for the measurements at femtogram levels, they have to be taken into account, as they could generate few cps at m/z = 239.

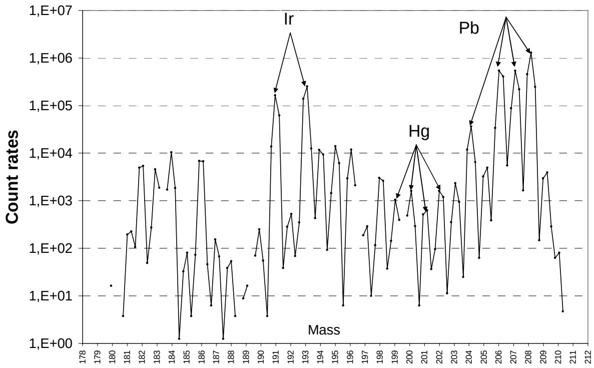

Figure 1. ICP-SF-MS VG "Axiom" count rates (cps) from mass 180 to 210 on a purified swipe sample from IAEA.

2.3. Method for determining plutonium detection limits on real samples

A potential plutonium detection limit is validated only if heavy elements counts cannot give birth to a significant signal. Even in that case, we generally observe variations of the background from one mass to the other. This leads to an important question: how can we properly define the detection limit for plutonium isotopes (^{239}Pu and ^{240}Pu)? We think that it is not liable to use background values at masses 239 and 240 from the method blank or from the rinsing solution because they generally are far less affected by interferences. Therefore, detection limits may be underestimated and this can lead to false detection of plutonium. Actually, the detection limits for plutonium isotopes is defined as three times the standard deviation over background values measured for the sample itself at neighbouring masses. The formulas of detection limits we use for ^{239}Pu is reported hereunder.

$$LoD_{239}(ppq) = 3\cdot\sqrt{\frac{\sum_{i=b_1}^{b_p} s_{C_i}^2}{p^2} + \left(\tau_{UH+AS,239}\cdot s_{C_{238}}\right)^2 + \left(C_{238}\cdot s_{\tau_{UH+AS,239}}\right)^2 + \left(\tau_{imp.239}\cdot s_{C_{242}}\right)^2 + \left(C_{242}\cdot s_{\tau_{imp.239}}\right)^2}$$

With c_i is the count rate at mass i, τ_{UH+AS} is the uranium hydride ratio + abundance sensitivity, τ_{imp} is the impurities ratio of the ^{242}Pu isotope dilution tracer and masses b_1 to b_p are background masses in the actinide mass range, measured in the sample. S_{ci}, $S_{\tau imp}$,… are the standard deviations respectively over the count rate of isotope i, over the impurities ratio of the ID tracer….

These detection limits are three times the standard deviation that is the combination of standard deviations over uranium hydride, abundance sensitivity, impurities from ^{242}Pu and isotopic dilution tracer corrections, and standard deviation over count rates of selected neighbouring (241-247) masses acquired during the measurements of the samples. This method can possibly lead to overestimation of the detection limits as the neighbouring masses are more affected by polyatomic species than plutonium isotopes. Anyway, it avoids the risk of false plutonium detection. Figure 2 reports detection limits that we determined on the real samples that we analysed in our laboratory during the years 2003 to 2005, using Axiom.

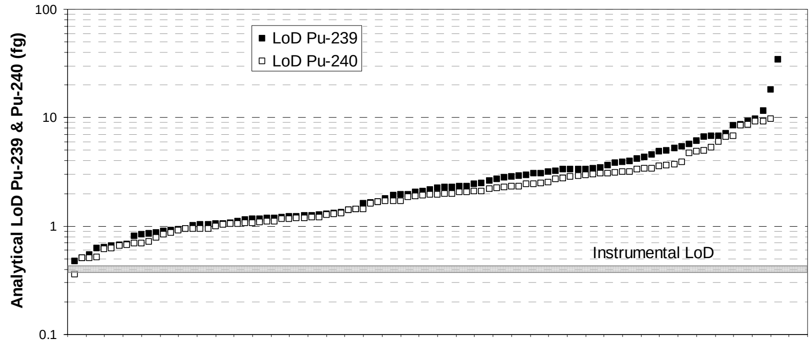

Figure 2. ICP-SFMS ^{239}Pu and ^{240}Pu analytical limits of detection for real samples analysed for years 2003 to 2005 calculated according to the 3σ criteria, using bkg values at neighbouring masses.

Figure 2 indicates that the detection limits on real samples range from 0.5 to 10 femtograms, so are generally higher than the instrumental detection limit due to the phenomena described above. The highest detection limits are mainly due to the presence of an excess of uranium in the solutions, because of a poor purification. In order to lower the detection limits, we adapted the radiochemical procedures for the purification of samples devoted to plutonium measurements. It is described below.

2. 4. Radiochemistry adapted to the purification of samples for plutonium measurements.

The objective of the separation and purification steps is to minimize the mass of uranium and other interfering elements in the eluted plutonium solution. Our "tolerated operational limit" is 50 pg of ^{238}U as it is equivalent to about 1 femtogram of ^{239}Pu, taking into account the uranium hydrides. The main steps of the separation and purification procedure for Pu are presented in figure 3.

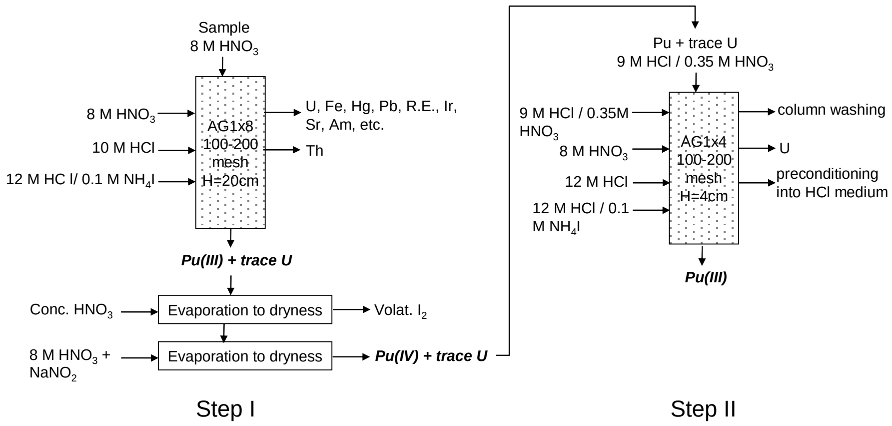

Figure 3. Radiochemical separation of plutonium from real environmental samples.

<u>Step I</u>: After complete digestion, the sample is dissolved into 8 M HNO_3 and few pellets of $NaNO_2$ are added to stabilise plutonium at the valence IV. The solution is added to an AG 1x8 anion exchange resin column (1 cm i.d. × 20 cm height pre-washed with 8 M HNO_3). The column is washed two or three times with 50-mL portions of 8 M HNO_3 to remove U and other elements. Then, Th is discarded by washing the column twice with 50-mL portions of 10 M HCl. Finally, Pu (and trace of U) is eluted by reduction into the trivalent state by iodide in concentrated hydrochloric medium (50 mL of a mixture 12 M HCl – 0.1 M NH_4I). About 5 mL of concentrated HNO_3 is added to the eluate and the mixture is evaporated to dryness (volatilization of I_2). The residue is dissolved with a mixture of 8 M HNO_3 – $NaNO_2$ (oxidation of Pu(III) into Pu(IV) by nitrite ions) and evaporated to dryness.

<u>Step II</u>: The residue is dissolved in 2 – 4 mL of a 9M HCl – 0.35 M HNO_3 mixture. The solution is added to an AG 1x4 anion exchange resin column (1 cm i.d. × 4 cm height pre-washed with 9 M HCl

– 0.35 M HNO_3). The column is washed 3 times with 3-mL portions of 9 M HCl – 0.35 M HNO_3. The column is washed with 20-mL 8 M HNO_3, 5-mL 12M HCl. Finally Pu is eluted 3 times with 3-mL portions of 12 M HCl – 0.1 M NH_4I. After volatilization of I_2 by evaporation to dryness with 0.5-mL concentrated UP HNO_3, the residue is dissolved for ICPMS analysis with 10 mL of 2 % UP HNO_3.

The influence of crucial steps and the sources of contamination coming from the atmosphere, the glassware and the resins have been studied. It appears that the atmosphere is not a major source of contamination, only from 1 to 10 pg, even if the radiochemistry is not performed in a clean room, provided the samples are protected with watch glasses during evaporation. It also appears that the glassware is a moderated source of contamination, from 1 to 10 pg of uranium if borosilicated Pyrex glassware is used and washed with diluted high purity nitric acid before use. Concerning the radiochemical procedure, the washing of the big column and also the small column with nitric acid is the crucial step and it has been optimized. Washing with large volumes, 100 to 150 ml for the 20 cm column and 20 ml for the 4 cm column, of ultra-pure 8M nitric acid almost completely eliminates the uranium from the sample. Therefore, if the elution step is performed with ultra-pure reagents, then the total quantity of uranium originating from the radiochemistry can be as low as 10pg. This way, with very strict operating conditions, detection limits for plutonium in real samples can be closer to the instrumental detection limit as uranium hydrides are considerably reduced. Those operating conditions will be systematically generalized in our laboratory, leading progressively to the lowest detection limits indicated in figure 2. This figure also indicates that the optimised washing using 8M HNO_3 contributes to the elimination of mercury, iridium and lead, that are, with U, the main interfering elements. Work is in progress in order to exactly quantify the purification factor.

2.5. The performance of micro-nebulisers with respect to sensitivity.

The purification described above allows reducing the final volume of the purified solutions down to 500 µl. With such low volumes, the use of micro-nebulisers is mandatory. Many micro-nebulisers are available and we tested them in order to determine the most suited one. Figure 4 presents the sensitivity of PFA nebulisers with flow rates ranging from 20 to 100 µl.min^{-1}, compared to a conventional Meinhard. With respect to this nebuliser, the loss in sensitivity using PFA-50 and PFA-100 is limited, whereas the loss is more important using PFA-20. Owing to the fact that PFA-20 appeared less robust and could generate clogging, PFA-50 and PFA-100 are chosen for the measurements.

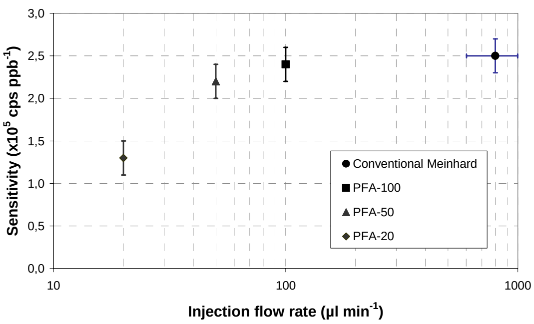

Figure 4. Sensitivity of different micro-nebulisers compared with conventional Meinhard.

2.6. Measurements on real samples

From 2003 to 2005, hundreds of low levels plutonium and uranium samples were measured in our laboratories. Most of them, coming from the former French nuclear test sites, are low level samples with concentrations ranging from 1fg to 40pg of ^{239}Pu and 1fg to 5pg of ^{240}Pu per sample. At the lowest levels, the major source of uncertainty for low level plutonium measurements generally comes from the poor counting statistics. The problem is different for uranium ratios measurements, as counting is often excellent. Therefore, the main issue is to correct for instrumental mass bias to obtain as accurate as possible results. Consequently, for uranium but also for plutonium measurements, metrological settings are optimised because of the use of uranium standards, natural or NBS005, that are included in the analytical procedure to assess the mass bias. Also, the best detector dead time has to be carefully determined in order to avoid another bias.

An example of the mass bias, measured on NBS005 uranium standard at a concentration of 1µg.L^{-1}, using a ThermoElectron "X-seriesII"(Bremen, Germany), is shown in figure 5. It shows the good stability of the mass bias that is close to 0.23% per mass unit. RSD is 0.18% and same tests conducted using a NBS500 uranium standard gave 0.04%. To obtain the best accuracy, the mass bias must remain very stable throughout the whole analysis. In our analytical procedures, we use uranium standard bracketing of the samples in order to check the stability of the mass bias and to provide the best correction. In order to determine the best detector dead time, we measure 5-10 times in a random order a series of four natural uranium standard solutions with different concentrations. With a detector dead time fixed at 37 ns, the $^{235}U/^{238}U$ ratios remain constant all over the range. This value is regularly checked and changed if necessary.

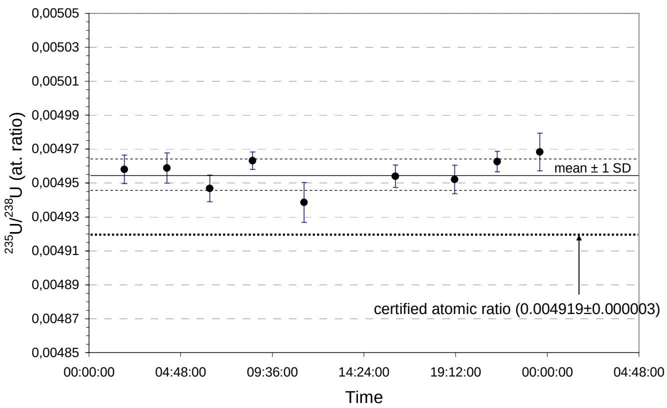

Figure 5. Mass bias measured on a 1µg.L^{-1} NBS005 uranium standard solution.

2.7. The potentialities of the femtosecond laser ablation ICP-MS.

Taking into account the intrinsic performance of ICP-MS, particularly its great sensitivity for uranium, better than what TIMS (at least with a common filament source, without specific treatment like benzene carburation that enhances sensitivity) could offer, we investigated the potentialities of the coupling of a laser ablation system and ICP-MS for uranium isotopic measurements in micrometer size particles. A femtosecond laser was chosen because it should present many advantages with respect to classical nanosecond pulsed lasers, namely : i) a potential for reduced fractionation, ii) a better yield to convert a small quantity of mass from a solid sample into vapour phase and transport that vapour efficiently to analytical source, iii) a better ability to ablate well defined crater with minimal thermal heating of the surrounding area (less melt zone, micro-cracks, shock waves and heat transfer to surrounding material). From an analytical point of view, it is expected a better signal/background, leading to a better detection limit. Those advantages have already been described [9]. The femtosecond laser we used, already described in [10] is operated by the "laboratoire de chimie analytique et bioorganique (LCABIE, Pau, France), in the frame of a collaboration with our institute.

The goal was to evaluate the potential of laser ablation-ICPMS technique for particle analysis. This technique allows a discrete sampling of a few micrometers. For these first trials, particles of natural uranium (2µm diameter) were deposited on polycarbonates disks. We used an Alfamet femtosecond laser (Novalase, France) with a wavelength of 1030 nm. The energy delivered was around 50µJ per pulse. As this laser allows a large range of repetition rate (from 1 to 10,000 Hz), we investigated the effect of this parameter on the degradation of the matrix around particles. An optimum repetition rate was determined around 100 to 300 Hz which allows a high sensitivity without affecting the matrix around the laser shot (and thus decreasing the risk to ablate particles in the vicinity). The ICP-MS used in this study is a Thermo Electron "X-series" quadrupole-based ICP-MS equipped with a single detector. As the signal is transient (50 s) and noisy, the precision of isotopic ratios is largely dependant on the integration time used during the measurement of each isotope. We typically used a dwell time of 50ms for each isotope.

Figure 6 shows the ^{238}U intensity response for a particle of 2 µm-diameter. A very high sensitivity was obtained, up to 700,000 counts at the maximum. The quality of the introduction efficiency of the ionisation in the plasma was evaluated, with respect to the liquid (2% nitric acid) media. Total counts of 8×10^6 were acquired. Taking into account the size of the particle and the sensitivity of ICP-MS, 1.5×10^5 cps.ppb^{-1}, using a 100µl micronebulizer, total counts acquired with a liquid source would have been very close to that value of 8×10^6. That means that the efficiency of the ionisation in the plasma, using laser ablation as an introduction system, is very good, close to the liquid introduction one.

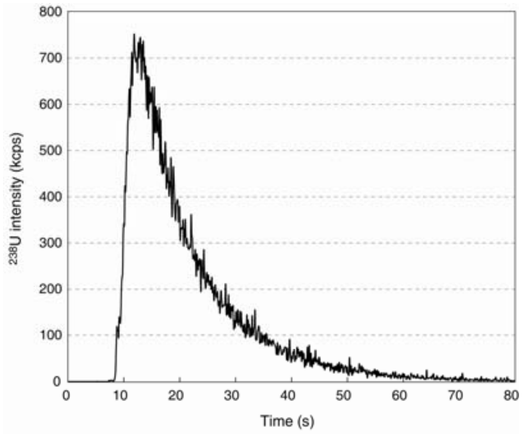

Figure 6. ^{238}U intensity response for a particle of 2 µm in diameter using LA-ICP-MS.

Isotopic ratios were also determined during the acquisition. Individual ratios are plotted in figure 7. A large dispersion of ratios is shown, mainly because we used a single collector ICP-MS. However, the average isotopic ratio for $^{235}U/^{238}U$ is 0.714×10^{-2}, just a little biased with respect to the isotopic ratio of the used standard that is $0{,}725\times10^{-2}$. No mass bias correction was applied for these results.

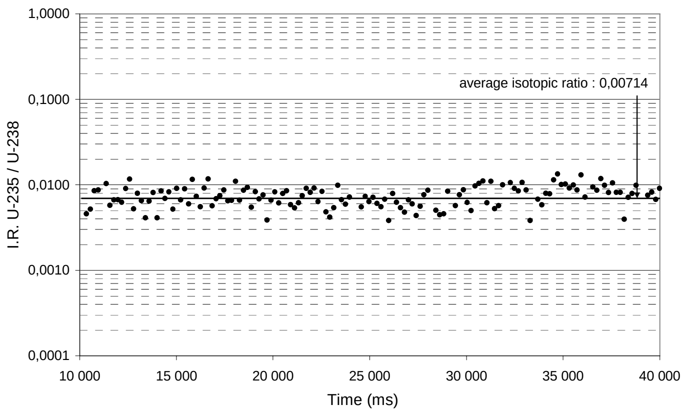

Figure 7. $^{235}U/^{238}U$ measurements of a 2µm uranium particle using LA-ICP-MS.

The first attempts are very promising. The technique is sensitive, rapid and easy to use. In comparison with classical fission tracks TIMS process, this approach is less time consuming, as there is no need to irradiate samples and to sample particles on the track etch detector. Moreover, the use of a MC-ICP-MS could provide very reliable isotopic ratios. Nevertheless, there is a lot of work to do for optimising this technique, determining limits of detection, uncertainties, budget, etc…, investigating all possible biases and traps, and validating it by analysing real particle samples and comparing these results to other techniques.

Conclusion

For the measurement of plutonium in real samples at femtogram levels, it is first necessary to identify and quantify the polyatomic interferences that occur at m/z = 239 and 240. The concentrations of the interfering elements have to be measured in the purified samples along with Pu isotopes in the same analytical batch. Plutonium detection limits in real samples need to be evaluated conservatively, in order to avoid any false positive detection. The combination of standard deviations over uranium hydride, abundance sensitivity, impurities from ^{242}Pu isotopic dilution tracer corrections, and standard deviation over count rates of selected neighbouring masses acquired during the measurements of the samples has to be taken into account. It is also necessary to optimise the radiochemistry, prior to the measurements, by eliminating most of the interfering elements identified initially. Besides, for precise and accurate uranium isotopic ratio measurements, metrological settings of ICP-MS need to be very precise, particularly detector dead time. Furthermore, bracketing the samples with certified isotopic standards is necessary to provide accurate correction for mass bias. In the second part of this study, related to traces measurements, the potential of Laser Ablation-ICP-MS has been evaluated as an alternative to TIMS for the particle analysis and the first results appear very encouraging.

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The ESARDA Working Group on Containment and Surveillance: Activities and achievements

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Abstract. One of the discipline-oriented working groups of the European Safeguards Research and Development Association (ESARDA) is the working group on containment and surveillance (C/S). Its mission is to provide the safeguards community with expert advice on C/S instruments and methods and on their performance; and act as a forum for the exchange of information on such instruments and methods, including unattended and remote monitoring systems. Currently, fifteen institutions contribute to the working group as members or observers. The institutions represent safeguards authorities including the Euratom Safeguards Directorate and the International Atomic Energy Agency (IAEA), instrument developers, plant operators, and national authorities. The working group meets twice per year, in order to address and discuss topics of interest. To this end, individual working group members volunteer to prepare discussion and working papers. The goal is to publish the results of the working group's discussions, preferably in the ESARDA Bulletin. The working group's current major discussion topic is related to the development of methods to determine the performance and assurance of C/S instruments. For the ESARDA web site, the working group is maintaining a compendium of C/S instruments and methods and drafting technical sheets on safeguards-specific topics. Together with the ESARDA working group on techniques and standards for non destructive assay (NDA) guidelines for developing unattended and remote monitoring and measuring systems were elaborated. The paper describes the working group's current terms of reference, membership, working method, topics of interest, achievements, and gives an outlook on the topics which the working group will address in the near future.

1. INTRODUCTION

The European Safeguards Research and Development Association (ESARDA) [1] provides a forum for all those who wish to contribute to improvements in international safeguards at a professional level. The principal issues are co-ordination of research, exchange of information and joint execution of R&D programmes. Working groups were established to promote and undertake collaborative R&D and information exchange on activities in various fields. One of them is the discipline-oriented working group on containment and surveillance (C/) which is described in this paper.

2. MISSION AND TERMS OF REFERENCE

The working group's mission is to provide the safeguards community with expert advice on containment and surveillance (C/S) instruments and methods and on their performance; and act as a forum for the exchange of information on such instruments and methods, including unattended and remote monitoring systems. To this end the working group has been charged with 11 individual tasks:

1. Advise the European Commission and IAEA on new and improved instruments and methods and on areas where R&D effort is still needed.

2. Maintain a list of C/S instruments and methods currently used or under development for safeguards purposes.
3. Develop methods for determining the assurance and performance of C/S equipment and contribute to their evaluation.
4. Contribute to determining, on request, the assurance and performance of C/S equipment.
5. Promote cooperation with other working groups and the inspection authorities.
6. Assist in the development of C/S instruments and methods in support of safeguards approaches including the definition of technical requirements.
7. Assist in the development and assessment of data evaluation and decision tools.
8. Collaborate with other ESARDA working groups to develop comprehensive and integrated tools to support safeguards approaches.
9. Promote the exchange of information and experience among facility operators, safeguards authorities, and developers.
10. Study technical characteristics of instruments and devices from other domains (e.g., physical protection) and investigate into the possible transfer of technology from these domains to the safeguards area.
11. Look at other verification regimes and assess to what extent their techniques and methods could be used for safeguards.

3. MEMBERSHIP

In 2006, 21 individuals were admitted to the working group as members or observers. They represented ministries, plant operators, regulatory agencies, research & development establishments, safeguards equipment manufacturers, and safeguards authorities, among them the following ESARDA member organisations: Euratom Safeguards Directorate and Joint Research Centre Ispra representing the Commission of the European Communities, Finnish Radiation and Nuclear Safety Authority STUK, French Institut de Radioprotection et de Sûreté Nucléaire IRSN, German Jülich Research Centre FZJ and Gesellschaft für Nuklear-Service GNS, Swedish Nuclear Power Inspectorate SKI and Uppsala University, British Nuclear Group BNG and Department of Trade and Industry DTI. The following organisations from outside ESARDA contributed: International Atomic Energy Agency (IAEA), Canadian Nuclear Safety Commission CNSC, US American Sandia National Laboratories SNL and Canberra-Albuquerque, Inc., and Australian Safeguards and Non-proliferation Office. Until recently, also the Argentine-Brazilian Agency for Accounting and Control of Nuclear Materials ABACC participated in the working group. It is anticipated that ABACC will resume their participation in the near future.

4. WORKING METHOD

The working group meets twice per year. One meeting is usually coordinated with the ESARDA Annual Meeting taking place in May or June. The second meeting is held in autumn with the Joint Research Centre Ispra being the usual venue. Contributions to the working group are voluntary in nature and depend on the delegating organisation’s capabilities and resources. Other factors are expertise and availability of individual working group members.

In consultation with the ESARDA management and in compliance with its mission and terms of reference, the working group addresses and discusses topics of interest. To this end, individual working group members volunteer to prepare discussion and working papers. Certain topics may be of an interdisciplinary character raising the interest of more than one ESARDA working group. In such cases, two or more working groups enter into a temporary cooperation. The results arising from the working group discussions are primarily published in the ESARDA Bulletin.

5. PAST AND PRESENT ACTIVITIES

In the recent past, the working group addressed and discussed the following topics: Methods to determine the performance and assurance of C/S instruments; electronic safeguards seals; application of "search-by-contents" techniques for the review of safeguards surveillance streams; on-line process monitoring tool for inspecting a reprocessing facility; impact of integration of INFCIRC/153 and INFCIRC/540 safeguards on the use of C/S.

In several joint sessions the working groups on C/S and on NDA discussed guidelines for developing unattended and remote monitoring and measuring systems (URMMS). A one-time joint meeting was held by the working groups on C/S and on verification technologies and methodologies (VTM). Topics of interest were the IAEA Project SGTS-08 "Novel techniques and instruments for the detection of undeclared nuclear facilities, materials and activities", 3 D modelling from satellite imagery and other scientific studies, and geophysical survey and monitoring for safeguards.

In anticipation of the 2006 ESARDA Annual Meeting, the working group had been asked to discuss topics relevant for the implementation of the Additional Protocol which entered into force in the European Union (EU) on 30th April, 2004. The following issues were addressed:

- Methods to determine the performance and assurance of C/S instruments
- needs arising from new Euratom safeguards approaches
- safeguards design and simulation tool
- Global Nuclear Energy Partnership: Reducing the proliferation risk of the civilian nuclear fuel cycle.

Recurrent activities of the working group are: Information exchange and discussions on R&D performed by working group members, maintaining a compendium of C/S instruments and methods, and drafting of technical sheets for publication on the ESARDA website.

6. ACHIEVEMENTS

The working group on C/S has published 25 papers in a total of 33 issues of the ESARDA Bulletin. In the issue of February 2006, the working group contributed a statement on "*The Impact of Integration (INFCIRC/153 + 540) on the Use of C/S*" as well as the following document which was elaborated jointly with the working group on NDA: "*Guidelines for Developing Unattended and Remote Monitoring and Measurement Systems (URMMS)*".

The *Compendium of C/S Products* was first published in Bulletin no. 29, December 1998. Nowadays, the compendium is available on the ESARDA website and updated twice per year [2]. The compilation contains outline information on a range of C/S instruments and methods that would help the potential users to know where to obtain further information and, thus, to judge whether a particular device might meet the requirements of a specific safeguards application. Further publications on the website, prepared by the working group and released by the ESARDA Editorial Committee, are technical sheets on *Electronic Safeguards Seals, IAEA Adhesive Seal, Review of Surveillance Data, Fibre Optic Seal, and Data Transmission* [3].

Methods to Determine the Performance and Assurance of C/S Instruments

The working group on C/S first addressed the question of how to determine the performance and assurance of C/S instruments more than 15 years ago [4], when spent fuel management started to become an important issue with C/S as relevant safeguards measures. The issue was revisited as of AD 2003, in anticipation of the entering into force of the Additional Protocol (AP) in the EU. Apart from Euratom, the users of assessment methodologies would be the IAEA, plant operators, and instrument developers. The availability of methods to determine the performance and assurance of C/S instruments is of relevance for the implementation of the AP, as more and more unattended and remote monitoring and measurement systems will be applied, in order to reduce on-site inspection effort.

The starting point is the design of an appropriate safeguards system for a specific nuclear facility followed by the selection of appropriate safeguards instrumentation that could make up this safeguards system. To this end, a complete set of user requirements has to be defined. Performance and assurance of C/S instruments depend on their specific applications taking into account the specific functions of the C/S instruments in the safeguards approach. Therefore, it has to be determined which type of information provided by the plant operator is to be confirmed by the safeguards system.
Instrument performance aims at the creation of relevant data, whereas assurance aims at the creation of information in support of the inspector's decision process.
A methodological approach for determining the performance and assurance of C/S instrumentation will involve the following steps:
(1) Acquisition of information on facility design and operational characteristics
(2) identification of state specific characteristics such as AP in force and standard of State System of Accounting for and Control of Nuclear Material (SSAC)
(3) assumptions on diversion and misuse scenarios
(4) identification of safeguards requirements
(5) definition of safeguards approach and measures
(6) assessment of C/S instrumentation meeting safeguards requirements.

For C/S instrumentation a facility specific *task profile* has to be defined and checked against its *performance profile*. The generic tasks of C/S instruments can be specified as follows:
- To indicate anomalies
- to support freezing of accountancy data
- to support monitoring of declared nuclear material flows.

A performance profile will involve factors such as instrument sensitivity, data quality, tamper resistance, and reliability which have to be specified for the intended use..
The desired assurance of effective safeguards is a positive conclusion on non-diversion of nuclear material placed under safeguards, and a positive conclusion on the absence of undeclared nuclear activities and materials.

Two proposals on how to address C/S performance and assurance have emerged in the working group's discussions. Both of them foresee to refer to an exemplary nuclear facility and a possible C/S system involving C/S devices with different sensors, i.e., cameras, electronic seals, and radiation monitors. Physical protection systems will not be considered.

One of the proposals is to conduct a feasibility study including the definition of a comprehensive follow-up project. The emphasis should lie on the performance of the C/S system as a whole with the possibility to look at individual sensors within the system and data review. Only active C/S devices will be considered, i.e., yielding measurable signals in contrast to, e.g., passive seals. For the purpose of the study, active devices will be called sensors (e.g., cameras, switches, and radiation monitors). The application has yet to be specified.

The other proposal foresees the development of a methodological approach for the long-term dry storage of spent fuel assemblies. The resulting first approximation methodology concept shall serve as a basis for the discussion of the methodology within all concerned ESARDA working groups. This proposal will be described here in more detail.

As an example, a facility has been chosen that is designed for the dry storage of casks filled with spent LWR fuel assemblies as well as of casks filled with various types of radioactive waste, e.g., vitrified highly active waste (HAW) resulting from reprocessing of spent fuel assemblies. The storage capacity is about 400 casks. HAW casks and spent fuel casks have similar designs. Empty casks may also be stored at the facility. The dimensions of the building are about 200m by 40m with a height of 20m. Figure 1 shows the floor plan of the building which is divided into two parts: (I) reception area and (II) storage hall.

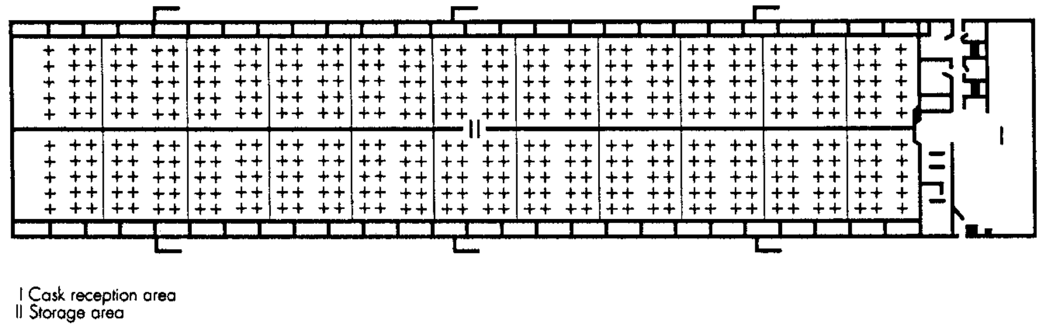

Figure 1. Dry storage facility for spent nuclear fuel and highly active nuclear waste.

The operational characteristics of the facility are as follows. Spent fuel casks arrive at the reception area. Here, they are unloaded from the transport vehicle, prepared for storage, and, by means of a travelling crane, transferred into the storage area. At their storage positions they are placed in upright positions on base plates, where they are permanently monitored for safety reasons. It is foreseen that, in case of leakage, casks are removed from the storage hall for maintenance in the reception area.

Diversion and misuse scenarios and, accordingly, the safeguards measures applied may vary depending on the situation in the state under consideration. If an AP is in force, and if the state as a whole has been evaluated by the IAEA with a positive conclusion, some scenarios may not have the same relevance as in states without an AP in force.

Relevant diversion scenarios in the reception area, i.e., during reception or maintenance work, could be (1) removal of cask after receipt is recorded, (2) declaration of HAW cask as spent fuel cask, and (3) replacement of cask loaded with spent fuel with empty cask, dummy or cask filled with HAW.

Relevant diversion scenarios in the storage area would be related to (1) removal of a cask and replacing it with an empty cask, a dummy or a cask filled with HAW, and (2) lifting a cask, cutting its bottom and emptying it.

The following principles are applied in the safeguards approach. The dry storage facility is categorised as an item facility with the casks being the items. Their nuclear material contents were verified at the shipping facility, and this knowledge is being maintained by means of appropriate C/S measures, a major measure being the attachment of a safeguards seal at the shipping facility. At the dry storage facility, cask identity and integrity can be further ensured by safeguards seals, whereas re-measurement of the cask inventory is not possible. Cask handling can be monitored by unattended optical surveillance. Finally, the safeguards system must provide for distinguishing between spent fuel casks, HAW casks, and empty casks. This can be achieved by unattended radiation monitoring.

With regard to the reception area the following inspection activities are conceivable: Verification of the incoming casks and seal checking; attachment of two safeguards seals to the casks; review of optical surveillance data; review of neutron monitoring data recorded at the transfer point between reception area and storage area to enable discrimination between loaded and empty casks.

In the storage area the following inspection activities are conceivable: Checking of seals and cask integrity; for storage, two different types of seal (metal seal and electronic seal) have been attached to the casks. The metal seal serves as a backup, whereas the electronic seal is applied to facilitate verification of cask identity and integrity during inspections.

The task profile of the C/S instrumentation in the reception area can be specified as follows: Unattended optical surveillance is applied to provide for continuous observation of all cask operations in the reception area. Neutron monitors are installed to support optical surveillance and register, if a cask is empty or not. They must be able to generate unambiguous signals regardless of the radiation background, and allow discrimination between spent fuel and HAW casks.

The task profile of the C/S instrumentation in the storage area can be specified as follows: Seals on the casks are used for verification of cask identity and integrity (preserve the continuity of knowledge). Reading of the seals must be easy and fast to reduce the time of the inspector's exposure to radiation. It should be possible to group casks and secure a group of casks with one seal, in order to cover specific diversion scenarios like lifting a cask and removing the content after cutting off the cask bottom. The seals should be tolerant with regard to radiation and temperature. Alternatively, it could be investigated if the application of optical surveillance would yield better results for performance and assurance.

The methodology to determine the performance and assurance of C/S instruments involves the following steps:
- Compile a list of factors and criteria
- describe task, performance, and assurance profiles for a specific C/S application
- compare and evaluate alternative C/S instruments

On this basis, a first approximation methodology concept will be outlined.

Examples of factors and criteria for a performance profile are as follows:
- Sensitivity (temporal and spatial resolution of images and scene changes)
- discrimination level between signal and noise to suppress irrelevant data
- data redundancy and correlation
- false alarm probability
- environmental qualification
- vulnerability to interferences
- reliability and functional lifetime
- tamper indication
- data review
- user-friendliness
- maintenance requirements.

The methodological approach described here is a kind of expert system with a data base and three types of profile for the assessment process. The data base will contain information on facility design and operation, state specific characteristics, relevant diversion and misuse scenarios, safeguards requirements, safeguards approach and measures, and specifications of C/S instruments.

Figure 2 shows the intended system structure of the assessment methodology.

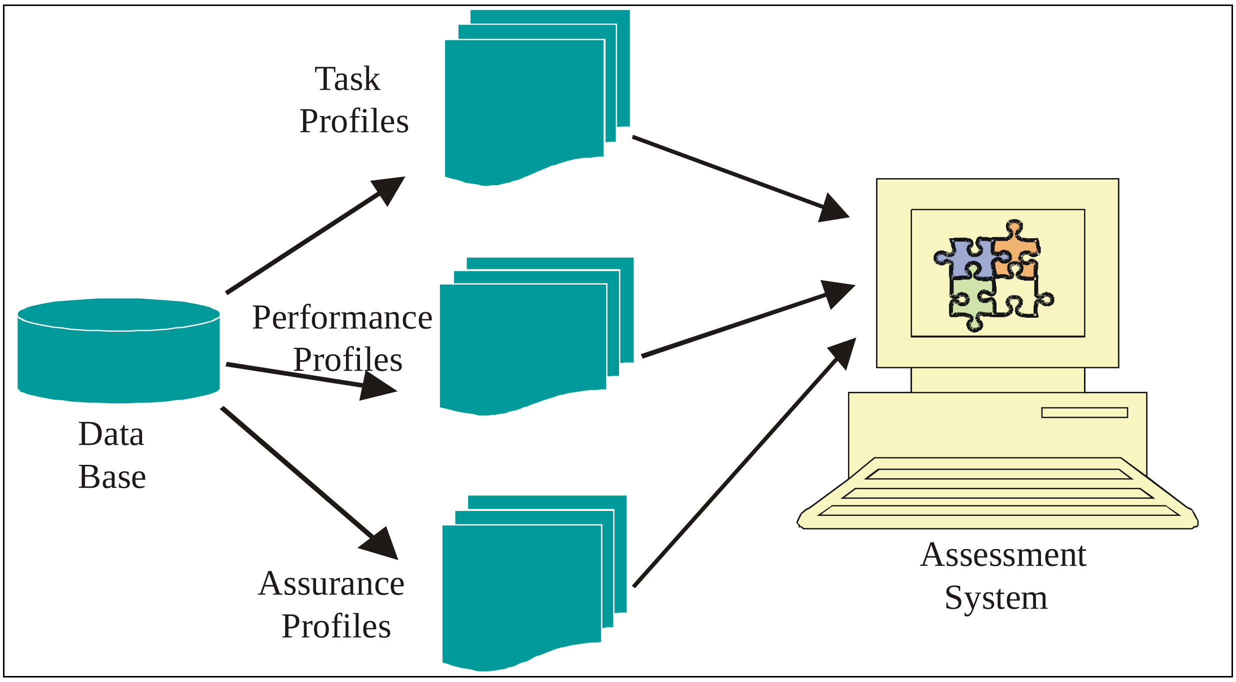

Figure 2. Proposed system structure for an assessment methodology.

Needs arising from new Euratom safeguards approaches

Within Euratom the following current needs in the C/S area have been identified: Standardisation of equipment, which requires an increased involvement of both organisations (Euratom and IAEA) at the conception phase of equipment development; to ensure transparency during equipment development; to maintain a list of equipment that is commonly used by Euratom and IAEA; to agree on a standard encryption protocol for remote data transmission; to develop a common concept for data storage and handling; to revise instrument failure criteria in a dynamic way, i.e. relate directly to actual situation. ESARDA is the appropriate forum for addressing these issues. In the near future, also the following issues should be addressed by the working group on C/S: How to respond to technological changes (security industry); definition of component life-span; reliance on single manufacturer; single inspector interface for all sensors; enhanced use of filters/triggers to minimise workload; authentication of plant operator's instrumentation (branching of signals for safeguards use); wireless data transmission within facility.

Safeguards design and simulation tool

At the JRC Ispra, a safeguards design and simulation tool is being developed based on VIRTOOLS, a commercial Virtual Reality package, and on 3DS Max. The capabilities of the tool are as follows: 3D modeling of the JRC PERLA Laboratory in Virtual Reality for appropriate installation of (a) neutron monitors and (b) surveillance cameras; representation of radiation measurements and sensors status; real-time monitoring and off-line reviewing of sensors measurements; visualisation of the nuclear material container position; interactive tool for camera placement and lens selection. The working group will be kept informed about further progress and, if requested, will discuss and comment on the results of the project.

Global Nuclear Energy Partnership (GNEP)

The information was provided by the colleague from Sandia National Laboratories. The long-term GNEP goals require near-term technical cooperation including advanced safeguards and physical protection research, development and testing, fuel cycle service approaches, and safe and secure spent fuel management. The main thrust of GNEP is international partnership in reducing the proliferation risk of the civilian nuclear fuel cycle. Also here, ESARDA can be involved in a reasonable way.

7. FUTURE ACTIVITIES

In the near-term future, the ESARDA working on C/S will continue to work on the following issues:

- Performance and assurance of C/S instruments
- data review
- aspects of the new safeguards approaches.

Furthermore, the working group intends to address the following issues:

- Wireless in-plant data transmission
- guidelines on sealing and identification systems
- containment verification methods and techniques
- geological repositories, geophysical monitoring for design information verification
- remote system control.

For the ESARDA website it is foreseen to draft technical sheets on cap-and-wire seals, optical surveillance systems, ultrasonic seals, transponder seals, radiation monitoring, encryption and authentication, unattended remote monitoring and measuring systems, mail box systems, and satellite imagery.
Finally, the working group is involved in preparing an ESARDA textbook needed for teaching international safeguards in connection with nuclear engineering at European universities.

ACKNOWLEDGEMENTS

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The IAEA's Next Generation Surveillance System (NGSS): Considerations on the hardware design concept

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Abstract. The Next Generation Surveillance System (NGSS) is a development project of the International Atomic Energy Agency (IAEA) in cooperation with the German and United States Programmes in Support of the IAEA. In March 2005, the project was initiated. The first phase of the NGSS project focused on the conceptual design of the system, especially on the development of the Surveillance Core Component (SCC) comprising design of candidate hardware architectures, selection and irradiation testing of components, prototype design, and performance evaluation. The favoured design was based on a digital signal processor (DSP) of the Blackfin family from Analog Devices as the main processing unit. Irradiation pre-testing with breadboard designs showed that a Blackfin-based system will be capable of handling the expected single-event upset rate through special mitigation techniques that will be implemented in the DSP firmware such as majority vote and execution from flash memory. Firmware prototypes designed for performance evaluation showed that the proposed architecture will be capable of fulfilling the user requirements. The paper gives some details on the user requirements, hardware and firmware designs, and feasibility demonstration by means of a breadboard design.

1. INTRODUCTION

The Next Generation Surveillance System (NGSS) is a development project of the International Atomic Energy Agency (IAEA) in cooperation with the German and United States Programmes in Support of the IAEA. On 2005-03-23, the tripartite contract between the IAEA, Canberra Albuquerque, Inc. (Canberra), and Dr. Neumann Constultants (DNC) entered into force, and, at the same time, the project was initiated.

NGSS will be configured, to the extent possible, from commercial-off-the-shelf components. The crucial hardware module, currently referred to as the Surveillance Core Component (SCC) that will become the standard replacement of the aging DCM-14 surveillance module,

will necessarily be a customised design, as it will be highly safeguards specific and not found on any other market.
This paper quotes the agreed major user requirements and deals with the initial development steps of SCC. They comprised design of the SCC hardware architecture, selection and irradiation testing of components, prototype design, and performance evaluation. Finally, some information is given on the hardware and firmware designs, and on the feasibility demonstration by means of a SCC breadboard design.

2. USER REQUIREMENTS

In October 2003, the US Support Program organised a workshop for the IAEA. The purpose was to bring together surveillance experts from interested Member States Support Programmes, research and development institutions, and private industry. They were asked to advise the IAEA in defining a complete and detailed set of NGSS user requirements, which the IAEA was then able to use as a basis for the bidding process and successive project implementation. The major user requirements for the SCC are:

- Picture taking interval of 1 image per second
- support for high resolution and full colour images
- TCP/IP networking over Ethernet
- scalable removable storage media
- low power consumption (48 hours on battery)
- high reliability under harsh environmental conditions including radiation.

3. SCC HARDWARE DESIGN

The main signal processing unit of SCC will be a digital signal processor (DSP). Other important components will be a microcontroller, data storage units, video interface, power supply unit, and network interfaces. Figure 1 shows the foreseen SCC hardware architecture.

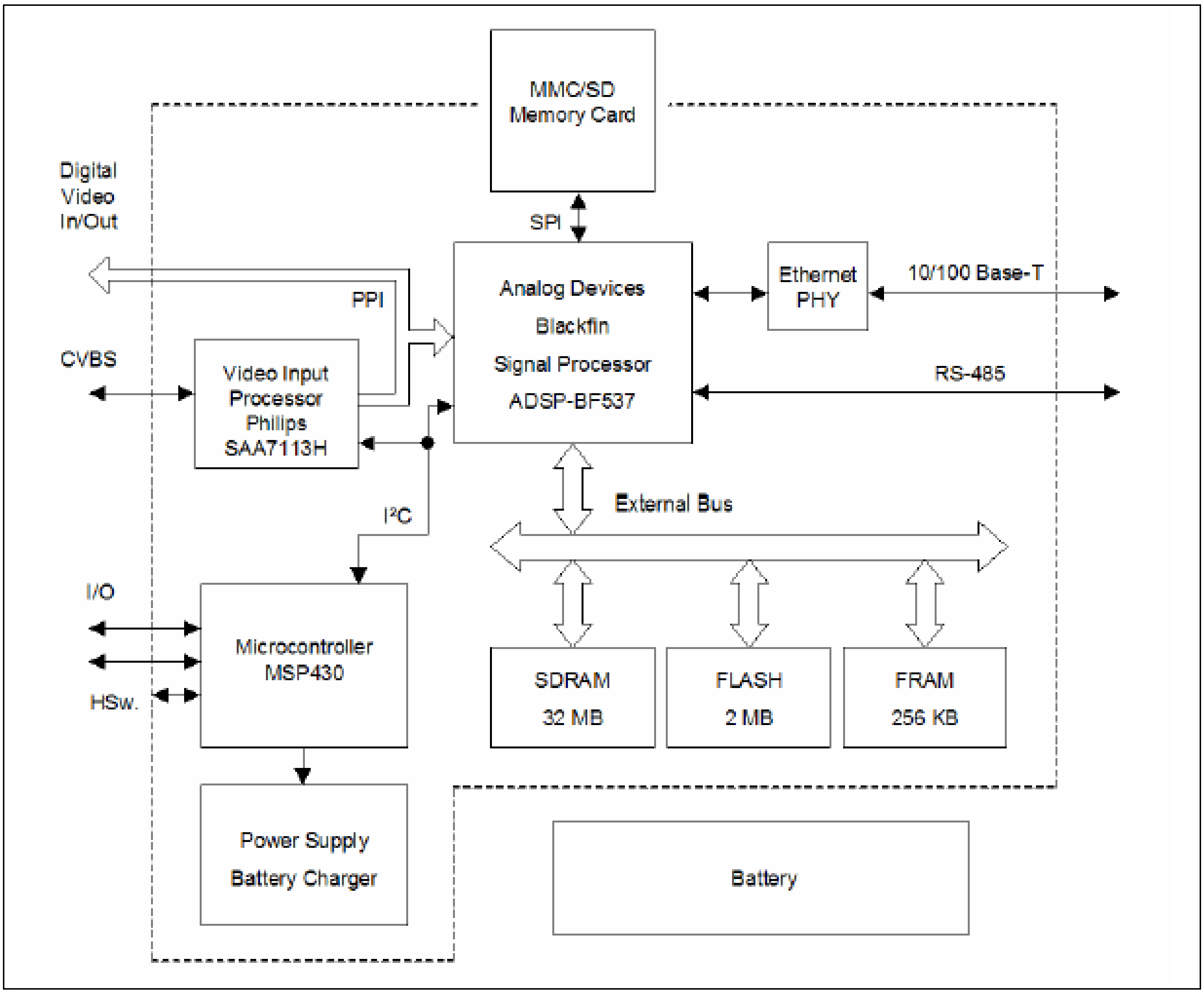

Figure 1. SCC hardware architecture.

3.1 Digital Signal Processor

Several DSPs were subjected to irradiation pre-testing at the Prater Reactor in Vienna/Austria. Finally, a DSP of the Blackfin family from Analog Devices was selected as the main processing unit for the SCC. Breadboard designs showed that a system with such a DSP would be capable of handling the expected single-event upset rate, provided special mitigation techniques will be implemented in the DSP firmware such as majority vote and execution from flash memory. Firmware prototypes designed for performance evaluation showed that the proposed architecture will be capable of fulfilling the user requirements. It was decided to use the Blackfin ADSP-BF537 which is a 16-bit fixed point DSP specialized for audio and video applications [1].

This DSP has the following interesting features:

- Industrial temperature range
- low power consumption
- 500 MHz core clock frequency, 120 MHz external bus frequency
- 132 kB internal memory
- 16/32 bit RISC instruction set and special vector and video processing instructions
- rich set of integrated, direct memory access-enabled peripherals.

The peripherals integrated in the DSP are foreseen for the following purposes:

- The watchdog timer supervises firmware execution and causes system resets upon detection of any anomalies.

- The battery-backed real-time clock provides date and time stamps.
- The Ethernet media access controller, together with an external physical interface (PHY), provides the network interface.
- The I^2C compliant two-wire interface controller connects to external I^2C peripherals such as video processors.
- Several versatile serial interfaces allow for high speed framed communication to external peripherals thus being candidates for open sensor interfaces of SCC.
- The parallel peripheral interface (PPI) allows glue-less connection of video input and output devices such as analogue video input processors, CMOS image sensors, and TFT panels.
- Asynchronous serial communication interfaces can be used as RS-485 for DCM-14 backwards compatibility.
- The synchronous serial peripheral interface (SPI) can be used to connect to MMC/SD memory cards.

A most valuable resource of the BF537 digital signal processor is its internal memory. The processor core can access this memory at its full speed with the DSP reaching maximum performance, if both code and data are stored in the internal memory. In contrast, access to any external memory such as SDRAM and FLASH would be significantly slower. The internal memory is, however, too small for storing complete video frames. Nevertheless, the direct memory access (DMA) controller of the DSP can be used to copy data portions from an external to the internal memory and vice versa without burdening the processor core.

3.2 Microcontroller

A microcontroller of the MSP430F1611-type was selected as a secondary controller for the SCC. It provides the following features:

- Ultra low power consumption and special low-power operating modes
- direct programme execution from internal flash memory
- controller peripherals (digital input/output, analogue-to-digital converter, digital-to-analogue converter).

The main functions of the MSP430F1611 will be intelligent power management and input/output control. As the MSP430F1611 will be active, even if the SCC runs on battery with the DSP resting in sleep mode, it will be capable of triggering the DSP. Furthermore, the MSP430F1611 will permanently monitor the tamper detection circuitry of the SCC. Cryptographic keys could be securely stored in the MSP430F1611’s internal RAM. Upon detection of an attack, the MSP430F1611 could erase the cryptographic keys. The MSP430F1611’s controller peripherals can be used as programmable sensor interfaces. Communication between DSP and MSP430F1611 will be possible through either I^2C or serial peripheral interface.

3.3 External Memory

The proposed design is an external DSP memory consisting of three parts:

- 32 MB SDRAM: The main use of the SDRAM will be the temporary storage of, e.g., video capture data (uncompressed), pre-triggering of images, or receiving/transmitting of buffer data.
- 2 MB FLASH: While the DSP will boot from the flash memory, i.e., the DSP’s boot loader will copy the firmware into the internal memory at start-up, it will also be possible to execute the programme code directly from the flash memory.
- 128 kB FRAM: The major function of the non-volatile FRAM (RAMTRON) will be to store the setup or configuration data. In addition, its virtually unlimited number of

possible erase/write cycles will allow storing of frequently changing state information such as image sequence numbers.

3.4 Interfaces

- Video: The proposal is to use a SAA7113H-type video input processor from Philips to provide connection to standard analogue video sources (CVBS). This processor can be connected in a glue-less way to the parallel peripheral interface port of the DSP. The circuit is I²C-bus controlled. In addition, it is considered to provide an alternative interface to the parallel peripheral interface for high-resolution CMOS image sensors.
- Network: The DSP already provides an internal Ethernet Media Access Controller (MAC). An external physical interface (PHY) will complement the circuit. One of the two asynchronous serial communication interfaces of the DSP will be equipped with a RS-485 transceiver to provide a physical communication interface that is compatible with the Agency’s current DCM 14-based surveillance systems.

3.5 Removable Data Storage Devices

Candidate techniques are multimedia/secure digital memory cards (MMC/SD) and compact flash (CF) cards. Due to rapidly changing technologies, it has been agreed to postpone the decision on the memory card type(s) that will be used in the SCC.

3.6 Power Supply

The SCC will have an onboard power supply that is compatible with external power sources in the range +9...+24 VDC.
As battery backup in case of a mains power outage, it is proposed to use a customised battery pack with sufficient capacity and internal redundancy on cell level (lithium-ion or lithium-polymer technology). Both technologies require an additional protection circuitry to be included in the battery pack (protection circuit module PCM).
According to the user requirements (see section 2.), the SCC shall be capable of maintaining regular operation without external power for at least 48 hours. As the power consumption depends mainly on the number of images to be recorded rather than on the total time span of mains power outage, battery capacity could be reduced, if picture taking intervals could be longer than 1 second between two recorded images. This would allow the SCC to enter into a reduced power mode with the majority of components of the circuitry being temporarily switched off. For the battery life time, the duty cycle would become the dominating factor. In contrast, regular operation of the SCC plus camera sensor would require the capacity of a car battery; i.e., even with state-of-the-art technology the battery pack would be quite bulky.

4. FIRMWARE DESIGN PREVIEW

4.1 Operating System

It was agreed not to use a third-party operating system such as Linux, Windows embedded, or any real-time operating system, in the SCC, the reasons being as follows:

- The system must be able to start up within milliseconds.
- There has to be single-event upset awareness even on the operating system level, in order to facilitate remedial functions such as majority voting.
- There needs to be unrestricted access to and a deep understanding of all source codes.
- There should be the option to adapt the system to changing requirements.

A proprietary operating system will be implemented in the DSP having the following basic features:

- Fast initialisation
- cooperative multitasking

- message passing (interrupt service routines to background task)
- semaphore handling
- wait timers (1 millisecond resolution).

4.2 Image Compression

There are advantages to use both JPEG Baseline [2] and MPEG-2 [3] video compression in the SCC.

4.2.1 JPEG Baseline

JPEG Baseline is a block-based compression algorithm for still images. A digital video input signal undergoes the following processing steps:
(1) Composition of a minimum coded unit (MCU); (2) level shifting; (3) discrete cosine transformation (DCT); (4) quantization; (5) entropy encoding.

The selected DSP is capable of performing MCU composition in hardware. The video input stream is stored in external SDRAM. The resulting MCU is stored in internal RAM. The DMA engine prepares the next MCU in the background, while the DSP core processes the previous MCU.
The level shifting benefits from the DSP's "zero-overhead" loop instruction. The Vector Add/Vector Subtract instruction of the DSP is capable to process two 16-bit samples in parallel.
The used DCT implementation was specially developed for the Blackfin DSP family. The implementation is based on the Chen algorithm. The code benefits from the capability of the DSP to process multiple instructions in parallel. The 8x8 2-dimensional DCT on a Blackfin DSP requires less than 300 cycles.
Quantization is actually an arithmetic division operation. The DSP does not have a built-in division instruction. However, the processor provides an instruction primitive (add/subtract and shift) that can be iteratively used to efficiently compose an integer division operation of the desired precision.
The entropy encoder processes symbols that consist of the number of consecutive zero coefficients (run-length coding) and the magnitude (number of relevant bits) of the next non-zero coefficient. The encoded symbol is then followed by the relevant data bits of the non-zero coefficient. The Blackfin DSP provides a dedicated instruction that can be used to determine the magnitude of a number. The instruction actually delivers the number of redundant sign bits, so that an additional subtraction is required to yield the magnitude. Moreover, the Blackfin instruction set significantly facilitates outputting entropy encoded data that are no longer byte- or word-aligned.
Finally, a JPEG file interchange format is generated. The output will be a file with the file extension JPG.

4.2.2 MPEG-2

MPEG-2 is a block-based compression algorithm for motion video. The output will be a file with the file extension MPG. The basic compression and coding technique of MPEG-2 very much resembles JPEG Baseline. Both algorithms are based on quantized DCT coefficients and use a similar approach for entropy coding.
JPEG processes pictures independently of each other, whereas MPEG-2 also accounts for correlations between successive frames. For instance, if there is a lot of temporal redundancy in a video sequence, MPEG-2 yields a significantly higher compression rate.
In order to save bandwidth and storage capacity (and costs), it is necessary to eliminate temporal redundancy from a picture sequence. MPEG provides several options: prediction, bi-

directional interpolation, and intra-coding. While none of them will be further explained here, it must be stated that the drawback is a loss of fault tolerance.
The MPEG compression algorithm has more processing steps than JPEG Baseline. Comparing the JPEG Baseline and MPEG-2 algorithms from a performance point of view, it can be stated that MPEG-2 processing takes twice as long as JPEG Baseline. In practice, however, the difference may be smaller partly due to the higher colour reduction factor, but especially because the higher compression rate will accelerate all subsequent processing steps such as authentication, encryption, storage, and transmission.
Finally, it must be stated that MPEG-2 is restricted by many patents. It would be advantageous to have the patents relevant for application in the SCC assembled in a licensing pool.

4.3 File System

For the removable storage media of the SCC, it has been proposed to use a file allocation table (FAT) of the FAT32-type which uses 32-bit cluster numbers. A cluster is the smallest unit on a storage volume that can be allocated. A typical cluster size in a 1GB volume is 4kB. The file allocation table provides one entry for each cluster on the disk. The entry is zero, if the cluster is free. Otherwise, the entry is part of a cluster chain (singly linked list).
Another important entity of the FAT file system is the folder structure. A folder is actually a special file with pre-defined 32-byte entries for every file or subfolder.
The FAT32 file system has the advantage of enabling a review of the stored data on a PC with standard software, i.e., a MPG-file can be reviewed using any commercial media player.
The file system can be implemented on a DSP-based system with reasonable effort, although long file names do require a greater effort.
Performance and storage efficiency are reduced, if lots of small files are to be stored in a FAT volume. However, this will probably not be a significant problem, as large MPEG-2 streams are to be created that combine thousands of images in a single file.
On PCs, FAT volumes tend to become heavily fragmented over the long run, so that files do not consist of adjacent clusters. This effect can impair system performance. Fragmentation occurs when files are arbitrarily created and deleted. This will probably not be the normal mode of operation of SCC. The SCC will either stop, if the volume is full, or overwrite old files. The latter will happen in the same order the files are created. Thus, fragmentation will be avoided.

4.4 Network Protocols

The following network protocols are considered for the SCC:

- Ethernet 802.1
- IP (RFC 791)
- ICMP (RFC 792)
- ARP (RFC 826)
- UDP (RFC 768)
- TCP (RFC 793)
- Telnet
- DHCP
- HTTP

5. BREADBOARD DESIGN

Tests were performed on a Blackfin BF537 DSP-prototyping board from Analog Devices (see Figure 2, top part).
An additional media board was developed that hosted the Philips video input processor and a MMC/SD card socket.
With this setup, it was possible to demonstrate that JPEG Baseline and/or MPEG-2 image compression from an analogue video source could be performed in less than one second. Hence, even under worst case conditions such as re-boot or heavy network load a one-second picture taking interval will be achievable.

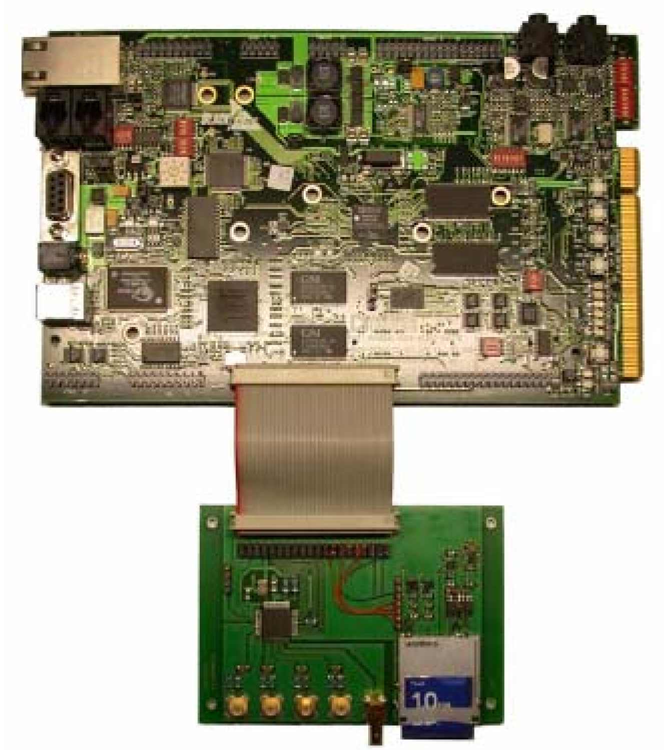

Figure 2. SCC breadboard design.

6. SUMMARY

In Phase I of the NGSS development the Blackfin ADSP-BF537-type digital signal processor was selected and approved as the main processing unit for the surveillance core component SCC. Irradiation pre-testing with breadboard designs showed that such a system will be capable of handling the expected single-event upset rate through special mitigation techniques that will be implemented in the DSP firmware such as majority vote and execution from flash memory.

Firmware prototypes designed for performance evaluation showed that the proposed architecture will be capable of fulfilling the user requirements.

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[3] MOVING PICTURE EXPERT GROUP (MPEG), ISO/IEC 13818-5:2005 (MPEG-2).

Laser surface authentication for containment and surveillance

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Abstract. Containment and Surveillance (C&S) techniques are extensively used for IAEA safeguards activities because they are flexible and cost effective. The two main C&S categories have been optical surveillance and sealing systems. Both techniques are often used to monitor nuclear material storage containers and to maintain continuity of knowledge on IAEA equipment. A new technology developed at the Nanotechnology Laboratories at Imperial College, London and the University of Durham may now provide a new tool for both *sealing systems authentication*, a containment issue, and *laser item identification,* a surveillance issue. The technology - Laser Surface Authentication (LSA) - is based on a laser optical technique and uses the phenomena of laser speckle to recognize and extract the inherent "fingerprint" within all material surfaces such as paper, plastic, metals and ceramics. The physical principle behind LSA and its general applications will be discussed in this paper. The potential application of LSA for metal seals authentication, a component of containment, will also be discussed, and initial test results shown. The paper concludes with a discussion of the application of LSA to a critical safeguards need for monitoring UF6 cylinder movement in an enrichment plant.

1. Introduction

The IAEA is constantly seeking to enhance the efficiency and effectiveness of its Safeguards tools and techniques. This is done primarily through market surveys in the private sector, or with requests for direct support from various Member State Support Programs. In some cases, the IAEA sponsors Technical Meetings in specialized areas, such as the 2004 Technical Meeting on Sealing Systems and Containment Verification. In the case of LSA, IAEA requirements for *laser item identification*, a surveillance task, led to the initial discovery of LSA and an ongoing partnership with Ingenia Technologies on the Laser Item Identification System. Subsequent discussions with Safeguards staff from the Technical Support Division (SGTS) and Ingenia have now led to work on *sealing systems authentication*, a containment task.

2. Laser Surface Authentication Background

2.1. Principles

Ingenia Technology has sponsored the work of Professor Russell Cowburn at Imperial College and Durham University in the development of a revolutionary technology for item level authentication. The technology, known as **"Laser Surface Authentication"** or **LSA™** is a breakthrough security technology, allowing for item authentication by using a low cost scanning laser to rapidly read the equivalent of a natural "fingerprint" inherent within the structure of every item.

LSA reads the surface of the inherent structure of metals, paper and plastics using a low cost laser, with a reliability level exceeding that of DNA in some situations. The reflected laser light from the surface is used in capturing microscopic signatures of surfaces. This opens up a completely new way of authenticating and tracking goods and documents including credit cards, banknotes, ID cards, passports, security seals, and containers with unprecedented levels of security and protection.

At a microscopic level the surface imperfections and irregularities of a credit card, a banknote, a passport, metal cylinder or seal are unique. For decades scientists have been looking for a way to read these unique surface signatures reliably. The LSA approach provides just such a method for the security and authentication industry resulting in covert identification at the item level. The system identifies the DNA-equivalent of materials. The technology is non-destructive and does not require any changes to the manufacturing process or addition of tags, inks or other markers. Critically, it is affordable and does not impact the production line, packaging processes, or require modifications to installed machinery.

At the scale of laser wavelengths, every surface is microscopically different. The LSA system uses a laser to read the naturally occurring surface fingerprint and then stores the information securely in a database. The document or object can be checked by performing a further simple scan which will automatically check against the existing stored 'fingerprint' data and verify its authenticity. In addition, since every surface on every item is microscopically different, the distribution of "like" intrinsic signatures is expected to be well separated from the distribution of "unlike" intrinsic signatures. The separation of these two distributions on some materials is such that estimated false positives are less than one in 10 to the power 100+ using fitted distributions, exceeding the forensic identification and discrimination performance of DNA. Consequently, the LSA system can be used to guarantee the legitimacy of branded packaging and high value products and has been validated on a number of materials including paper, cardboard, plastics, metals, foils, and ceramics.

LSA is not an optical recognition system because it is not recording an image of the surface but rather a digital "fingerprint" based on the microscopic surface imperfections which are unique to the item. In order for counterfeiters to copy the item, and pass the counterfeit forward as a genuine article, they would have to manufacture a product with the imperfections in exactly the same spatial arrangements as an original at the dimension of laser wavelengths of light. Since surface imperfections are manufactured in a totally random process, there are no known manufacturing techniques which can be employed to generate a false positive which would involve the manufacture of items with the same imperfection patterns in exactly the same spatial positioning as an original. Figure 1 shows the level of detail that can be exploited by LSA for both paper and plastic materials.

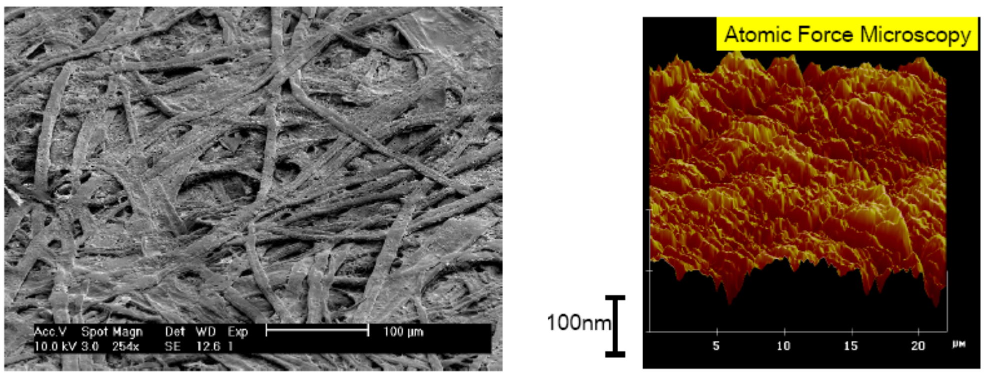

Figure 1. Microscopy Images of Paper (Left) and Plastic (Right).

2.2 Security

The system has significant security implications because it is impossible to replicate the 'fingerprint' of a document, packaging or metal surface. "Nature writes the code". It is incapable of being reproduced and therefore has the potential to bring an end to product counterfeiting. Every LSA scan records a unique "fingerprint" for the item generated from the random imperfections in the surface as shown in the data record in Figure 2. This is basically a physically generated "key". A significant advantage to this "key" is that it typically requires only the order of 200 bytes to store on a computer. This means that a simple laptop computer can accommodate a data base of millions of items.

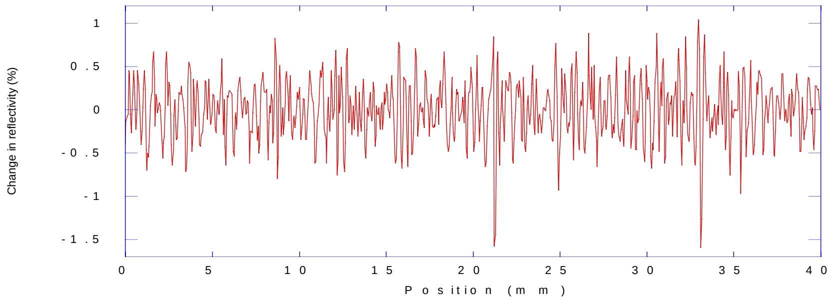

Figure 2. An LSA Data Record.

2.3 Test and Evaluation

Tests on a packaging production line have validated that the system is capable of handling unique scans of items moving at 2m/sec. In principle 4m/sec and even faster should be possible, which covers virtually all industrial scenarios.

Rigorous tests have also been conducted on a range of different materials. Tests include items that have been washed in machines, scratched, scorched and covered in inks and coatings. For some materials, testing has demonstrated that packaging can be damaged from age, crumpling, writing and spilling and the system will still recognise them as the original. Such tests have shown that despite this damage, the system still recognises only the authentic item. Details on some of this work may be found in an article on the new technology recently published in *Nature*[1]. Ingenia Technology is currently engaged with the IAEA in assessments of the technology for containment and surveillance applications, as discussed in the following portions of this paper.

3. LSA Applied to IAEA Metal Seal Authentication.

Sealing systems must satisfy both *authentication* and *tamper indication* requirements. A seal is authentic if we can verify at a high level of confidence that it was the one applied by an IAEA inspector and not a clever counterfeit. A seal must also indicate any tamper properly to be effective. Another issue for sealing systems is whether the declaration of seal integrity can be made immediately in the field, or only after forensic examination in a laboratory. The application of LSA towards in-situ authentication of the IAEA Metal Seal will be discussed.

Figure 3. The IAEA Metal Seal Including Seal Wire.

The IAEA metal (CAP) seal (Figure 3) is the most commonly applied IAEA seal (~18,000 used annually) and is typically applied to maintain the continuity of knowledge on nuclear materials and as a tamper indicating device for IAEA equipment. The seal has two metallic parts which, when engaged, cannot be opened without leaving evidence of tampering. A metal wire is used as a sealing wire and a knot is tied inside the seal body to close the loop. With the knot inside the seal, the loop cannot be opened without tamper indication. The main advantages of the seal are its simplicity, physical robustness, small size and weight, and low lifecycle costs. The main disadvantage is that verification must be performed at the IAEA's headquarters. This is because of the current method of authentication by optical comparison.

The authentication feature of the metal seal resides in the inner surfaces of the two halves of the seal. An optical "scratch 'n solder signature" is manually created and photographed. These images are recorded and kept for later verification by the Seals Lab. Once the seal is applied by the inspector, it is not opened again until it is removed by the inspector who inspects the wire as well. The detached seals are then returned to the Seals Lab for verification by Seals Lab Technicians.

In the CAP seal authentication by the LSA technique, a focused laser beam scatters from the seal surface and a scanner records the resulting laser speckle, creating a unique identity code. This code, like any other type of seal identifier, can be stored in a database along with other information such as date, time of application, inspector identity, and location. The LSA system will use a compact laser device to read this naturally occurring fingerprint and then store the information securely in a database. The seal can be checked at any time by performing subsequent scans, which will automatically check against the existing stored 'fingerprint' and verify its authenticity. The authentication method becomes "intrinsic" as opposed to "applied". The LSA signature of the CAP seal requires no optical signature of the metal seal. The materials of the metal surfaces alone are sufficient to supply a robust and potentially counterfeit resistant signature. Thus, the LSA system can provide a high level of security against seal counterfeiting compared to other technologies, and at a fraction of the cost since the seal itself, already a low cost device, requires no modification.

This enhanced counterfeit resistance is paired by an increase in efficiency. The proposed method can be easily automated for the preparation and verification process in the Seals Lab thereby reducing the human resources required for forensic authentication in a laboratory setting. Given the large number of metal seals to be verified, the Seals Lab can cope with this workload using fewer staff hours than at present. However, the LSA technique is also amenable to in-situ authentication. It is anticipated that a compact field instrument will be developed for the inspector to verify the metal seal in-situ. This has the added advantage of significantly decreasing the time required for seal verification.

A Phase I materials feasibility demonstration has shown that a reliable CAP seal LSA signature may be extracted and remeasured for both halves of the CAP seal under laboratory conditions. Figure 4 shows the discrimination results obtained for the top half (copper portion) of the CAP seal. The results show extremely good separation of data between like and unlike CAP seal authentication measurements.

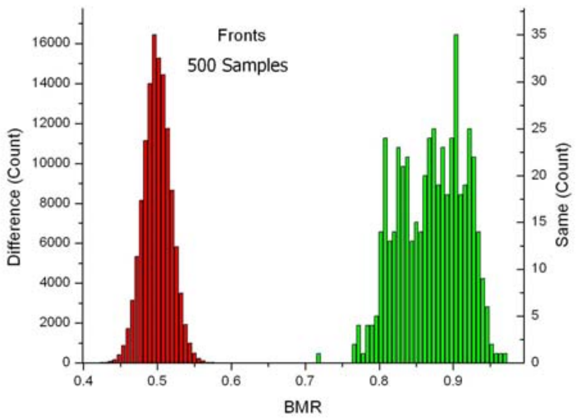

Figure 4. IAEA Metal Seal Signature Discrimination.

The histogram on the left shows the distribution of the comparison statistic (bit matching ratio) when comparing one metal seal to a different metal seal. It is seen that the statistic is narrowly confined between 0.45 and 0.55 with most values at 0.5. In contrast, the histogram on the right of the figure shows the distribution of the comparison statistic when measuring the same metal seal twice. This statistic is primarily confined between the values of 0.75 and 0.95 with central values around 0.88. It is unusual to find such remarkable separation in these types of measurements. In this discrete case, based on 500 seal samples, there is literally no possibility of making an authentication error. Since the left-hand distribution is a binomial distribution, a binomial fit can be made which yields parameters of $p = 0.5$ and $N = 380$. However, 2 parallel scan lines are used that are statistically independent, meaning that $N=760$. Given these values, the probability of a false positive at a decision threshold of

BMR = 0.7 is a number on the order of 10^{-28}. Such a small number is intuitively hard to grasp, but since there are the order of 10^{26} Angstroms in a light year, we see that it is 100 times more likely to find a particular random Angstrom in a light year than it is to generate a false positive from this data.

Further phases of the work include vulnerability assessments to ensure that the seal meets safeguards requirements and can be applied reliably under field conditions. Equipment development for a full scope system used at HQ, Vienna, is foreseen within the next year. The last phase will be devoted to miniaturizing the device for the field.

4. LSA Technology for the Laser Item Identification System Project

LSA addresses another important Safeguards need for the unique identification of UF6 cylinders in enrichment plants. It is difficult to apply any existing IAEA specific tags or seals to those cylinders (30B, IPC, 48Y…) due to the harsh operating conditions. Hence, the use of the LSA surface fingerprint appears to be an interesting and promising technique.

Monitoring the UF6 cylinders in an enrichment plant will consist of three phases:
-1- Acquire all declared cylinders “signatures”, upon delivery of the cylinders to the operator, and store all those unique “fingerprints” in a database.
-2- Have an unattended monitoring scanner (LSA based) on the path of the cylinders entering or exiting the processing area.
-3- Verify, via a hand held device, the cylinder’s “identity” upon inspection and/or inventory (PIV etc).
Since each phase will imply an LSA based laser scanner, the proposed L2IS system described below will therefore be based on three “UNITS”.

4.1 L2IS UNIT1 Laser Scanner

This scanner unit will be operated by IAEA inspector on a yearly basis, and applied on all cylinders declared by the operator to enter the process. This will be the signature initialisation phase of the L2IS system database.

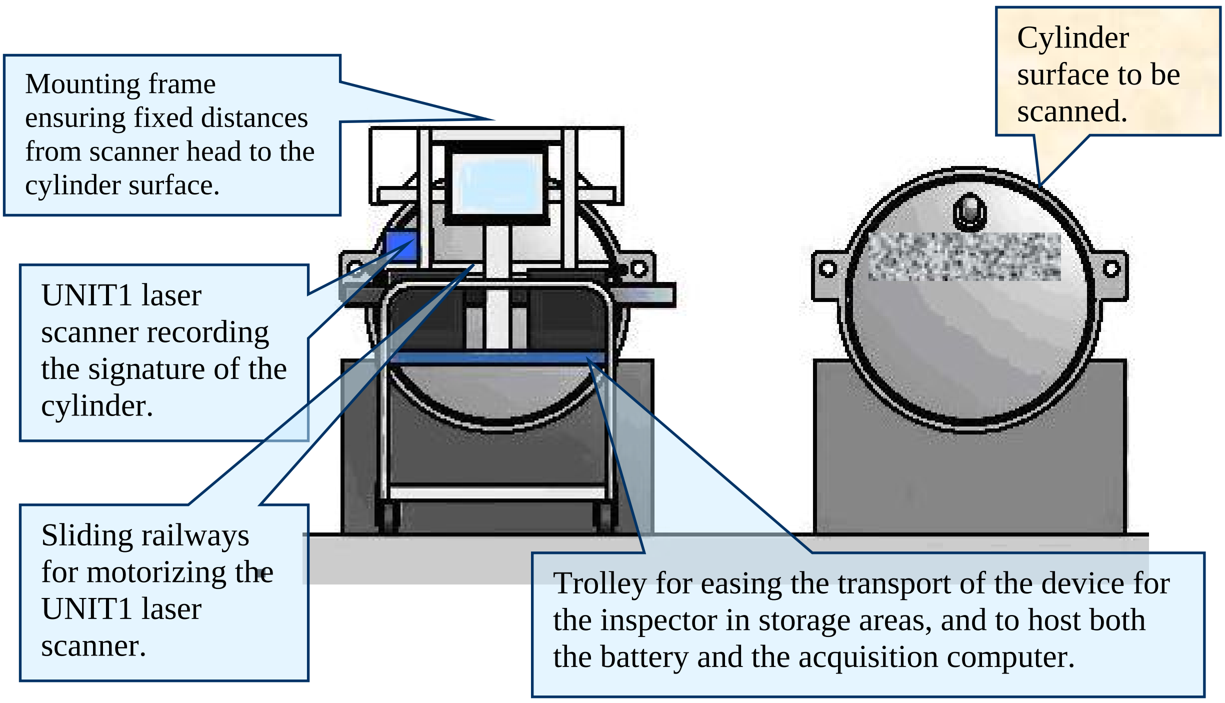

Figure 5. L2IS UNIT1 Principle (Stored UF6 Cylinder: Valve side view).

For all cylinders a wide band of the front surface will be "recorded" as the unique fingerprint of the cylinder. For each cylinder, the same part of its surface will be scanned by the UNIT1 scanner. Signatures from comparable surface areas that are to be compared will be obtained by the use of a mechanical mounting frame developed by the IAEA (Figure 5), and supporting a motorized laser scanner.

4.2 L2IS UNIT2 Laser Scanner

This scanner unit is unattended and linked to video surveillance. It will be located in a strategic position in order to be able to identify all UF6 cylinders entering or exiting the processing area. The UNIT2 scanner is unattended and therefore a static scanner, exploiting the movement of the transported cylinders. Because cylinders are transported on trolleys, the UNIT2 scanner laser beam will scan the surface of the cylinders (see Figure 6), and the recorded fingerprint will then be compared to the recorded signature acquired by UNIT1 during the initialisation phase and stored securely in a database.

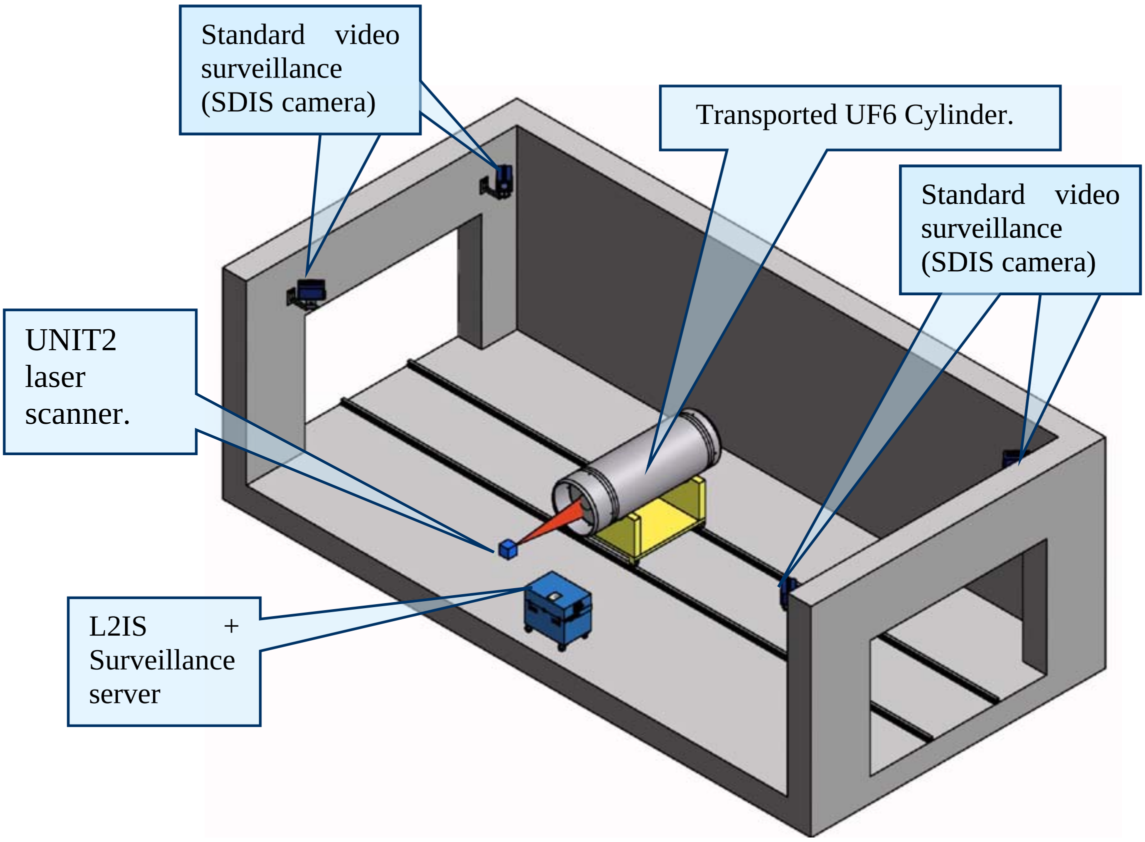

Figure 6. L2IS UNIT2 principle: The laser scanner catches all passing cylinders.

The L2IS UNIT2 laser scanner based on the LSA technology will be a reliable complement to the video surveillance by triggering the camera to a faster frequency of picture taking upon detection of a passing cylinder. Any failure to identify a cylinder presented in front of the UNIT2 laser scanner will lead to an alarm. The video surveillance will in this case be complementing the laser surveillance by indicating a possible misplacement of the cylinder or any other abnormal movement in the room.

One of the challenges the LSA technology is expected to meet is to be able to compensate for a possible cylinder rotation by using a computerised algorithm.

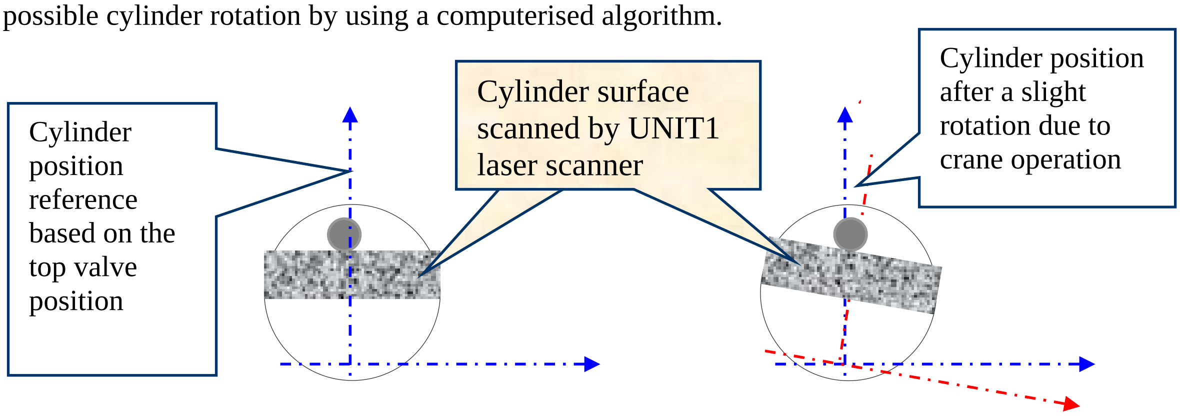

Figure 7-a. Database initialisation scanning position.

Figure 7-b. Worst case scenario: The cylinder has been slightly rotated ***(e.g. when lifted onto the transport trolley)****.*

4.3 L2IS UNIT3 Laser Scanner

This scanner unit will be a hand held device, allowing inspectors to identify UF6 cylinders in storage yards by scanning a subset area of each cylinder's surface. This has two advantages. By reducing the total area to be scanned, the data analysis proceeds faster (a smaller surface scanned implies less data to be checked) but also a random subset of the signature is chosen by not always scanning the complete surface.

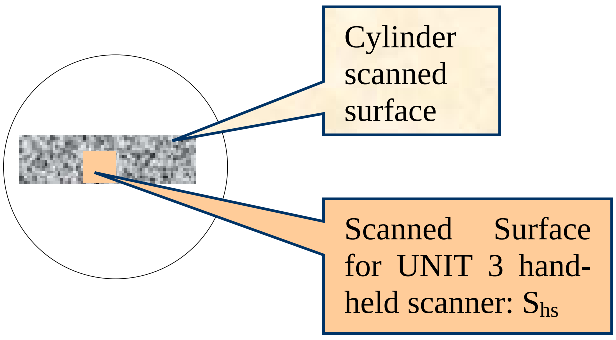

Figure 8. UNIT3 laser scanner unit will be performing "sampling" scanning.

Unlike all previous techniques the L2IS project does not rely on cylinder labelling and does not add additional IAEA tags or seals to the cylinders, but rather exploits the native unique surface microstructure of the cylinders to provide unique item identification.

LSA Technology is usually applied to a small surface area of smaller items, with high speed scanning of large quantities. The technology challenge of the Laser Item Identification System (L2IS) is the need to acquire a surface fingerprint from a larger surface, from larger physical items, at slower transit speeds, and from a further distance from the surface than the usual LSA scanning scenario. For the IAEA, the LSA based identification will allow inspectors to uniquely identify UF6 cylinders, enabling easy logging of transfers as well as easier inventory activities (gain in time and accuracy). The small size of the generated data (no pictures but bit-stream signatures) also will greatly ease the integration of the L2IS with existing surveillance systems, which could lead to entirely new concepts for future remote monitoring modes.

TheL2IS system is scheduled for proof of concept during the last half of 2006.

5. Conclusion

The new LSA technique, developed at Imperial College, London and the University of Durham, which exploits surface intrinsic features existing at the micron level, is a very promising technique for Safeguards applications in the area of containment and surveillance. Two possible applications have been identified at present and both applications are expected to increase the efficiency and effectiveness of the IAEA’s Safeguards tools for containment and surveillance measures. Proof of principle has already been achieved for the materials compatibility study for the IAEA Metal Seal. For the L2IS, proof of principle is expected during the last half of 2006.

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Remote monitoring of radio frequency tamper indicating devices in international safeguards applications

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Abstract. The IAEA has placed large quantities of direct use nuclear material under international safeguards at a dedicated location. To increase efficiency and effectiveness, it was decided to install remote monitoring systems (RMSs) that would allow active radio-frequency tamper indicating devices (seals) to be interrogated and video surveillance system data to be reviewed, at any time, from dedicated computers at IAEA headquarters in Vienna. To reduce cost and improve response time, it is also possible to place review stations at the IAEA's regional offices.

This paper describes the work performed to develop, install and test the RMSs that are being used to reduce inspection costs while increasing the availability and accuracy of critical safeguards information. In this application, the new RMSs replace an interim safeguards solution whereby seals and video surveillance data were reviewed on site by IAEA safeguards inspectors at each (monthly) inspection. The IAEA is reviewing a cost-benefit analysis of these systems, and it is expected that physical inspection frequency can be reduced from twelve to five inspections annually. The video surveillance and seals RMSs are completely independent and data are separately forwarded to IAEA headquarters. This is the first application where the IAEA has implemented an independent RMS for a large number of seals. It is also the IAEA's first implementation of radio frequency communication between a large group of seals and a remote data collection system. During 2005, the new RMSs were installed and the transfer of data to IAEA headquarters was tested. The systems have successfully completed a trial period with no safeguards significant system failures. System architecture, data protection methods, implementation and results of the field test are discussed.

1. Background

In many IAEA Member States large quantities of nuclear material are stored in vaults or at other high-density storage facilities. In these facilities it is common to use a large number of seals in close proximity. In one such application, the IAEA is using a new seal which uses a radio frequency link to communicate with a seal reader. This enables a small number of permanently installed seal readers to communicate, nearly continuously, with all of the seals in the facility.

Using this configuration the IAEA has demonstrated a new type of remote monitoring system (RMS). The new system enables real time monitoring of the 150 seals, currently installed at a facility, from IAEA headquarters in Vienna. This system is used in conjunction with a remotely monitored digital multi-channel optical surveillance (DMOS) system. When used together, the two systems make it possible to assess the facility's safeguards status on a daily basis, thereby significantly reducing the number of on-site inspection days without compromising the attainment of safeguards goals.

2. The sealing system

The seal used in this demonstration is the T1 seal developed by Sandia National Laboratories, USA. The IAEA refers to this seal as the T1 radio frequency seal (TRFS). This seal is a stand-alone tamper indicating device, which uses a fiber-optic cable to sense when a protected container is opened. The fiber-optic cable is passed through a series of hasps that serve to sever the cable if the container is opened. Optical pulses are passed from the seal's transmitter through the fiber-optic sensor cable to the seal's receiver. The seal interprets an unauthorized interruption of the optical pulse train as a tampering event. As with all electronic seals, the T1 records the date and time of all events. The seal works in conjunction with the material management system (MMS) software. The MMS provides the required system database and data review tools.

The system uses a radio frequency communication link between the seal and the seal reader. The seals continuously monitor their own state-of-health. In this application, the seals send a state-of-health message every six hours, and would immediately send a message if a safeguards significant event occurs.

3. System description

Three computers are used in the TRFS RMS: the buffer computer (buffer), the remote key management computer (RKMC), and the data review station (DRS). The computers are 1 GHz class industrial computers. Each computer is configured with numerous availability protections programmes and devices. Key features include a programme to restart critical software if it terminates unexpectedly and a watchdog timer that reboots the system if the required file movement stops for more than one hour. Both the RKMC and the DRS are also configured with an external 160 GB universal serial bus (USB) hard disk. The system performs the following functions:

(a) Receive TRFS seal data files from the operator;

(b) Archive the data files;

(c) Transmit the data files to IAEA headquarters over a secure virtual private network (VPN);

(d) Process these data files including cryptographically authenticating each individual seal message;

(e) Import this information into a SQL 2000 database; and

(f) Display the seal information for safeguards evaluation purposes.

The main functional elements of the system are: the network, the buffer computer, the RKMC, the DRS and the uninterruptible power supply (UPS). Each of these functional elements will be discussed below.

Every 30 minutes each computer pings the other computers in the network and sets its system time to the buffer computer’s system clock. This is done to ensure overall network operation and to keep a consistent time reference across all of the computers on the network

In order to protect the TRFS RMS from unauthorized access and tampering, all of the equipment required to implement this RMS is contained in a sealable tamper indicating cabinet, which is maintained under IAEA seal. The equipment rack is located in the IAEA office at the facility.

3.1. Remote monitoring network

The network is the backbone of the system, providing all of the required data communication and security. A Netscreen 2500 firewall appliance is used in conjunction with a four-port Ethernet hub to provide an Ethernet network. The Ethernet line to the operator’s LAN is connected to the untrusted input of the firewall. The firewall isolates the buffer computer from the operator’s network, allowing only access for FTP incoming files. The buffer computer is isolated on a separate subnet from the DRS and the RKMC computers, allowing unlimited access from the RKMC and the DRS to the buffer computer but no access from the buffer computer to either the RKMC or the DRS computers. The firewall is also used to establish a VPN connection between the RM equipment and the Safeguards LAN in Vienna. Seal data files are pulled from the RKMC computer to Vienna four times a day. A diagram depicting the network is shown in Figure 1.

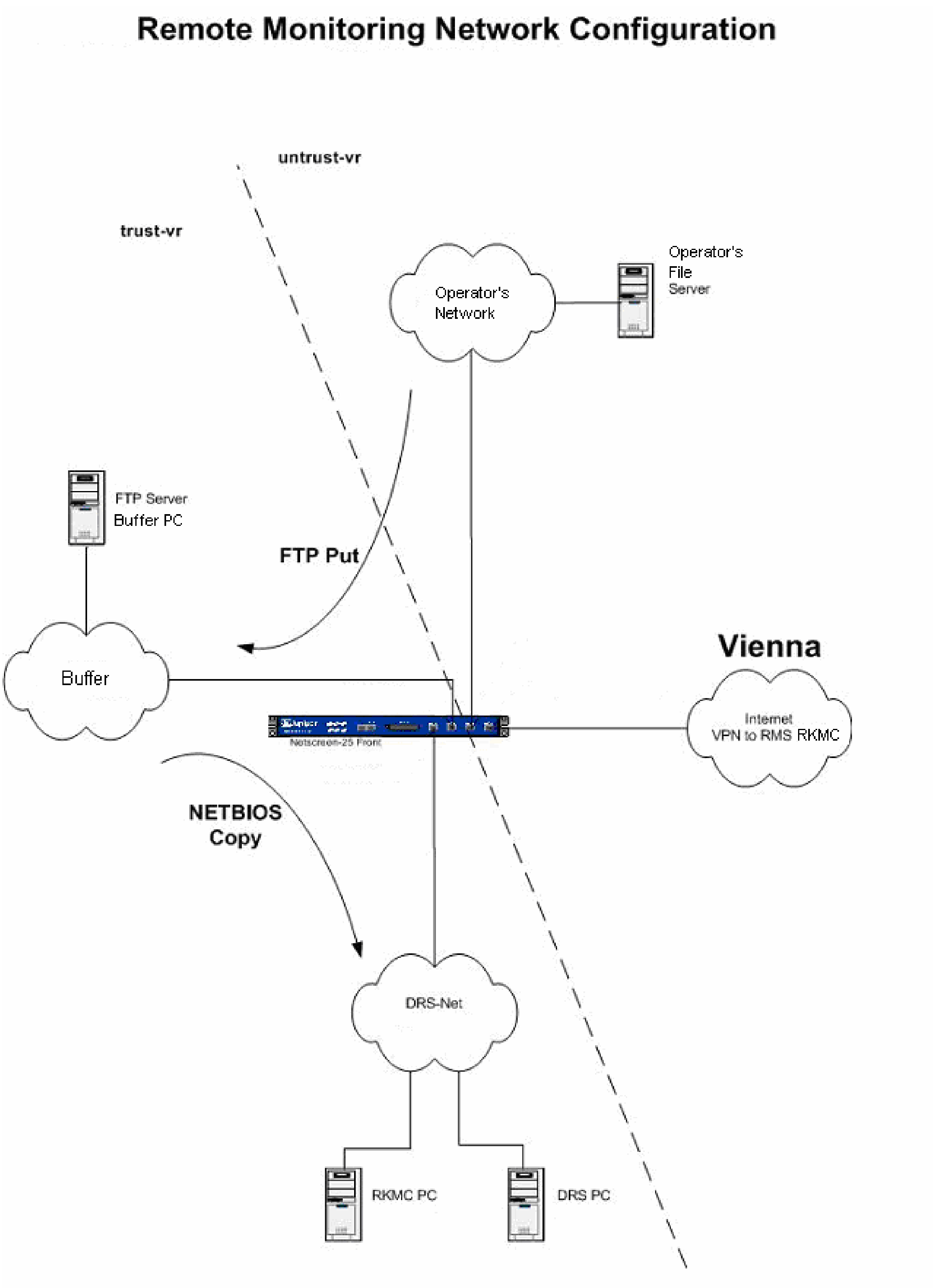
Remote Monitoring Network Configuration
untrust-vr
trust-vr
Operator's Network
Operator's
File
Server
FTP Server
Buffer PC
FTP Put
Buffer
Vienna
Internet
VPN to RMS RKMC
Netscreen-25 Front
NETBIOS
Copy
DRS-Net
RKMC PC
DRS PC

FIG. 1.

3.2. Buffer computer

The function of the buffer computer is to provide isolation between the untrusted operator network and the trusted RMS network. The buffer computer is configured with an FTP server to allow the operator to deposit seal data files onto the system. Operator access is limited to the use of the FTP put command. The RKMC computer copies only seal data files from the buffer computer and distributes them as required to other components of the system. In this way the buffer computer serves as a firebreak to make it more difficult to penetrate the trusted RMS network.

3.3. Remote key management computer

The RKMC is the heart of the RMS. It provides the required file movement by periodically moving the seal data files from the buffer computer to the following locations:

(a) Vienna directory,

(b) Local archive, and

(c) DRS working directory.

Once the seal data files are copied, the RKMC deletes all files from the buffer computer incoming directory. This is done to provide further protection from malware. As the buffer computer does not execute any files located in this directory and all files located there are deleted, any malware planted there will not have an opportunity to infect either the buffer or other computers on the trusted network. The seal data files are never executed and provide no vector to executable code; any viral infection of the files will not be effective.

The Vienna directory is used to hold files that will be transferred to Vienna. There are no subdirectories. The Safeguards LAN pulls all files located in this directory to Vienna. Upon completion of the file transfer all files located in this directory are deleted.

The Archive directory is provided to archive all files imported from the operator. The files are organized in a series of subdirectories that identify the exact date and time the files were imported. These files are deleted on a yearly basis.

The RKMC also provides a major security function by authenticating the TRFS seal messages. The seals attach a hash code to the end of each message. The RKMC recalculates the hash code and compares the calculated code with the transmitted code. If the message is authentic the two codes will be the same. The T1Verify programme authenticates the seal messages.

The T1Verfy programme is critical to the operation of the system. The MPSS programme is used to ensure that the T1Verify programme is running continuously. If it terminates for any reason, the MPSS programme will attempt to restart it.

3.4. Data review station computer

The MMS working directory is located on the DRS computer and is used to import the seal data files. The MMS data storage component (DSC) imports these data files from the working directory into the MMS database. As the data files are imported, the DSC passes TRFS seal messages to the RKMC via an RS232C connection. As discussed above, the RKMC authenticates these seal messages and the results are passed back to the DSC via the same RS232C connection. After the authenticated messages are imported into the MMS database, the data archive review component (DARC) provides an inspector friendly interface.

The DSC and the DARC are critical programmes running on the DRS computer. To ensure that they are continuously available, the MPSS programme is used to constantly monitor their status. If for any reason either programme aborts, the MPSS programme will attempt to restart it.

Every night at 12:30am the system generates and archives a full database backup. The backups are archived on an external disk drive. The Delphi programme is used to delete all backups older than one year.

3.5. Uninterruptible power supply

The UPS provides backup power for a period of two hours in the event of main power failure. A power selection relay is used to ensure that power is supplied from the main power source in the event of an UPS failure. The UPS is a commercial off the shelf product.

4. Field test results

During the first year of operation, the RMS encountered a few normal start-up problems. However, these issues were easily resolved without the loss of safeguards information. A list of these failures and actions taken to resolve them is shown below in Table 1.

Table 1. RMS problems and solution actions.

Problem	Action taken
Communication with the remote computers was lost from 2005-09-01 to 2005-09-23 due to hardware/software problems at the local computers.	Maintenance work at the facility was done by SGTS on 2005-09-23 and these problems were resolved. No data was lost.
Transmission of data files was stopped on 8 November between 16:00 and 17:00. IAEA asked the facility operator to check the RM support system	At 16:04 on 2005-11-08, an electrical test was performed which interrupted power to the IAEA link system data computer. The data computer did not restart. This fact was discovered the following day. Resetting the computer restored data transmission. No data was lost. An UPS was placed on the IAEA link in the Control Room to avoid this problem in the future.
IAEA was informed during PIV that power would be interrupted approximately 1hour on 18 November 2005 for system maintenance. Transmission of data files was stopped on 18 November between 07:00 and 08:00.	At 07:50 on 2005-11-18, there was an electrical outage for system maintenance. Automatic file transmission was interrupted due to a loss of the Media Converter. This was discovered the following Monday. Resetting the Media Converter restored normal operation. No data was lost. Three Auto Reset Media Converters were installed to avoid this problem in the future.
Transmission of data files was stopped on 16 April at 16:50. The watch-dog-timer log of the RKMC computer indicated that it had been rebooted. It was decided to check system operation during the 2006/005 Interim Inventory Verification (IIV). The facility operator confirmed that RM data files were being transmitted to the IAEA Seals RMS at that time. During the IIV, it was found that the IAEA RKMC computer had failed.	A sealing expert visited the facility and replaced the failing computer on 2006-05-30. Data files accumulated on the IAEA Buffer computer. Upon replacement of the RKMC, normal data transmission resumed. No data was lost.

5. Conclusions

This is the first fully functional seals specific RMS in use by the IAEA. In the first year of operation of the RMS there were a few equipment issues, which were easily repaired and represent typical system start-up problems. At no time was there a safeguards significant loss of data. With remote monitoring capabilities for both containment and surveillance systems now established, the IAEA has a new tool to maintain effective safeguards while reducing the cost of inspection for both the IAEA and the facility.

3D Technologies in safeguards applications

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Abstract. Three Dimensional – 3D – technologies have been proposed for Safeguards applications already in the early 90's, the major advantage being increased robustness in terms of verification. Recently major technological developments enabled the wide use of 3D technologies. The paper overviews different 3D technologies and discusses their potential in safeguards. The paper discusses the use of 3D technologies for improved continuity of knowledge in several application areas including site modelling and verification of design changes both indoors and outdoors, identification of nuclear containers by unique surface maps, the verification of containment and closure welds, and finally the augmentation of traditional 2D cameras with depth measurements for robust and accurate surveillance, including in-front-of-lens authentication. It presents some novel 3D surveillance sensors and 3D self-authentication techniques.

1. Introduction

The Additional Protocol to the Non-Proliferation Treaty foresees improved verification of existing nuclear installations. To be effective new advanced capabilities must be developed and fielded to increase the accuracy of verification and detection of changes in the facilities. New systems need to be portable, simple to use and, yet, highly accurate and dependable.

3D laser technologies proved to be effective in Design Information Verification (DIV). IAEA successfully uses a laser based DIV system at the Rokkasho Reprocessing Plant. The system allowed IAEA to carry out rapid and accurate DIVs far faster and more accurately than had been possible in the past [1]. A further application of 3D laser technologies is to perform the verification of the facility buildings. Typical plants are located on sites of few square kilometres with tens of buildings, housing process and storage facilities. This requires systems that are capable of measuring and verifying long distances and easy to handle in an outdoor environment.

This paper presents an overview of the different 3D technologies and discusses its potential use in safeguards applications:

- Design Information Verification [2].
- 3D Surveillance (overcomes the flatten world of classical 2D Surveillance and provides accurate quantitative (i.e., distance) measurements.
- Object self authentication (spatial forensics), including the verification of closure welds on containers [3].
- Outdoor verification System or verification of the facility buildings and outdoor perimeters [4, 5].

2. Design Information Verification for indoor and outdoor verifications

2.1. Rationale

Countries are required to declare design information on all new and modified facilities, which are under safeguards, and to ensure that the accuracy and completeness of the declaration is maintained for the life of the facility. It is the obligation of the United Nations' International Atomic Energy Agency (IAEA) to verify that the design and purpose of the "as-built" facility is as declared and that it continues to be correct. These activities are referred to as Design Information Examination and Verification (DIE/DIV). Although methodologies have been available for DIV, they have not provided the level of continuity of knowledge needed for the lifetime of the facility. Also, the size of some facilities as well as the complexity of their design and process poses an insurmountable challenge when considering 100% verification before the facility comes into operation and during its operational life.

Recently, new 3D laser-based tools emerged [1,2] enabling a quick and accurate modelling of large areas both inside buildings and outside on the plant area. The tools accept different types of laser range scanners covering distances from 1 mm to some 400 meters and software programs exist making possible to acquire and accurately reconstruct a 3-dimensional model of the scanned area. Other sources provided to the system apart from the data from laser scanners can be in the form of CAD models or other 3D representations. The final reconstructed model is then used for distance measurements and virtual inspection tasks as well as inspections where different epoch's of 3D models are compared to each other. The physical difference between these models can be identified and presented for an operator.

2.2. System overview

As an in-depth verification of all areas is beyond the Inspectorates' resources, a structured, methodical approach is taken prioritizing equipment, structures and activities and randomizing the lower priority items. Even with prioritized tasks and with the application of a random approach, the verification activities, especially for cell and piping verification remains a tedious and costly activity. Also the fact that DIV activities must take place over several years represents additional problems, the issue of maintaining continuity of knowledge of the previously verified equipment and structures being with no doubt the most important one, not only during the construction phase but also for the entire life of the plant. This leads to efficient modelling in different environments such as indoor and outdoor areas.

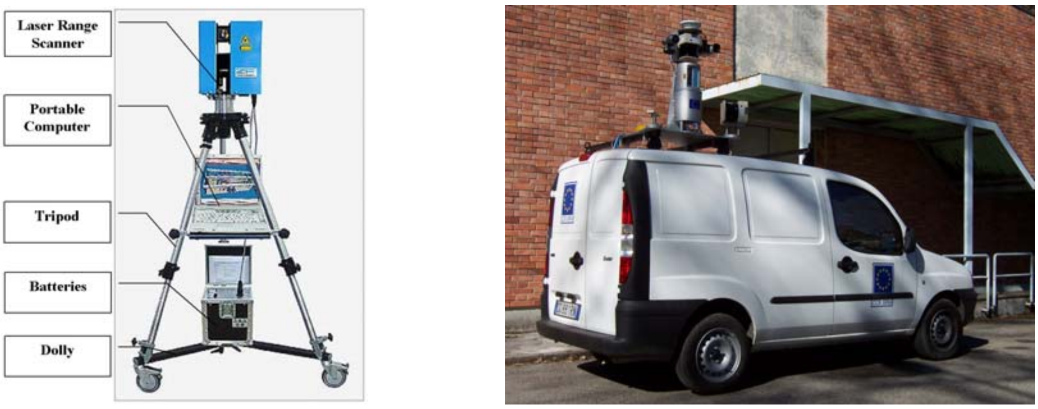
2.2.1. The hardware used in DIV

The basic meaurement device in all systems makes use of a portable commercial off-the-shelf laser range scanner and a portable computer. The laser range scanner is composed of a laser beam, a rotating mirror and a rotating base. The laser beam is deflected in two dimensions: vertically by the rotating deflection mirror and horizontally by the rotating movement of the entire measuring device. The accuries which can be acquired for a typical scanner lies in the range of a couple of millimeters and the maximum working range from 50m to 350m depending on embedded technology.

For DIV indoor operations, we need a highly portable system with a high level of autonomy. Such system is based on the above mentioned components mounted on a tripod with a dolly,see ***Figure 1***a. Different scan-positions are always needed to model an environment due to object occlusions. The movement of the scanner and its tripod is carried out by physically moving the scanner to suitable positions by the operator(s). For flexibility, the system is battery operated.

Considering outdoor scenarious, the requested area to be covered increses by magnitudes. In this case, one can make use of a vehichle for the transportation, see ***Figure 1***b. Additional components for a vehicle mounted system are an GPS and an Intertial System. These extra componentes keeps track of the acctual scanning position and orientation in order to identify better knowledge for the documentation of the acctual scanning. Given a site area of some resonable size, one can consider two operational modes for the vehicle mounted system; i) Stop-and-Scan and ii) Continuous Scanning.

For the Stop-and-Scan case, the car is stopped in a suitable position, the vehicle position and heading is automatically noted in the digital documentation. During all the laser scanning, the vechicle is remaing still. This technique is directly equivalent in terms of accuracy with the indoor scanning technique.



(a) **(b)**

Figure 1. (a) Laser scanner on tripod with dolly for indoor opertion (b) Laser scanner mounted on vehicle for outdoor operation.

In situations where the requested accuracy and detection capablitis are more relaxed to centimeter or decimeter level, one can make use of the scanning platform in a Continuous-Scanning mode. During this operational mode, the laser scanner is continuously scanning perpendicular to the vehicle driving direction which at the same time the vechicle position and orientation is saved. By driving the vehicle around the site, which operating the scanner-system, a large coverage area can rapidly be modelled.

2.3. Working methodology

The DIV procedure is divided into three distinctive phases (see ***Figure 2***):

1. Building a 3D Reference model by acquiring multiple scans from the scene. The quality of the DIV activities is highly dependent on how accurately and realistically the 3D model documents the “as-built” plant. As such, the reference model should be acquired with the best possible conditions considering the requested detection capabilities for the final results. Important apsects includes a) high spatial resolution, b) low measurement noise and c) multiple views to cover possible occlusions. The number of required scans depends on the complexity of the scene.

2. Initial verification of the 3D models cells versus the CAD models or the engineering drawing provided by the plant operator. The process is fully automatic. In the case of

unavailability of CAD models the software is provided with tools allowing the measurement of distances for verification of lengths, heights, pipe diameters, etc.

3. Re-verification. At that time, new scans of the selected area are taken and compared with the reference model in order to detect any differences with the initial verification. The automatic detected changes can be further analysed by an operator. The re-verification phase can occur at any time after the reference model is constructed.

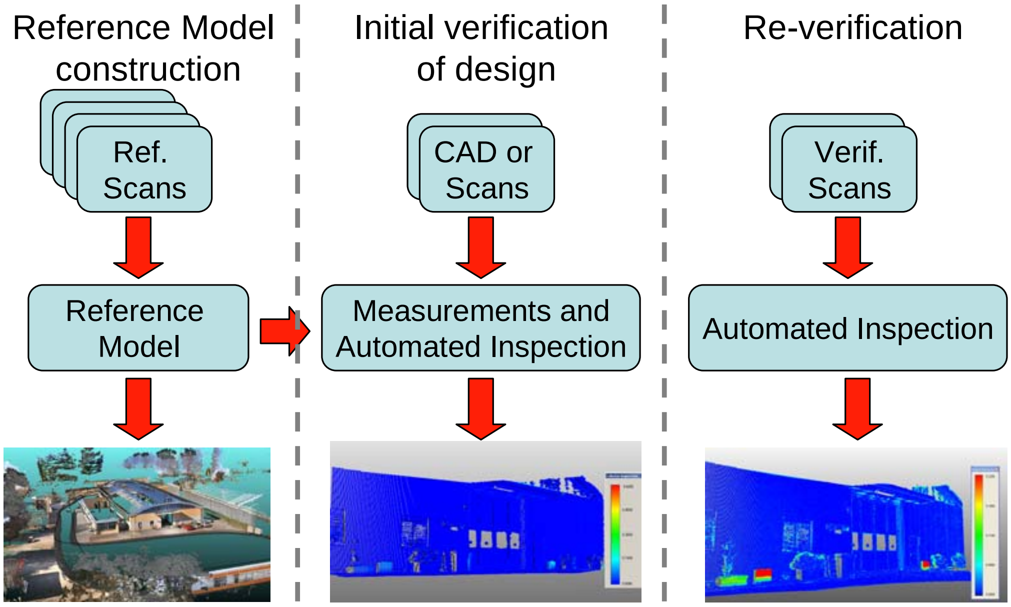
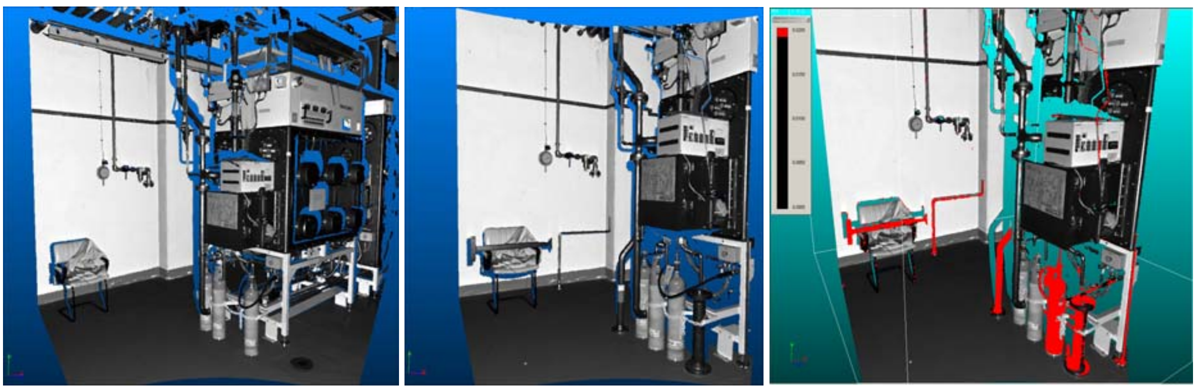


Figure 2. Working methodology for DIV.

2.4. Examples

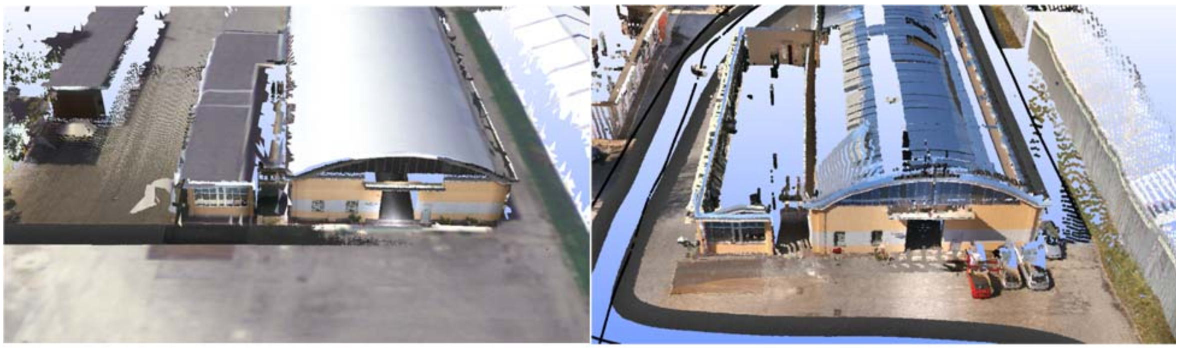
For indoor situations or when the outdoor Stop-and-Scan methodology has been used, changes of small scale objects can be detected. In ***Figure 3***a, we see a small part of a Reference Model. In ***Figure 3***b, the same area, now slightly modified, is observed from a new scan position. By visual inspection it is very difficult to identify all changes introduced. After an inspection-operation has been carried out with the DIV system, a view as the one in ***Figure 3***c can be presented to an operator/inspector. Detected changes between the models in ***Figure 3***a and b can be seen with the alarm-colouring schema, where changes are marked red. The smallest details change clearly identifable in this example is pipe to the right in the figure having a diameter of 6 mm.



(a) **(b)** **(c)**

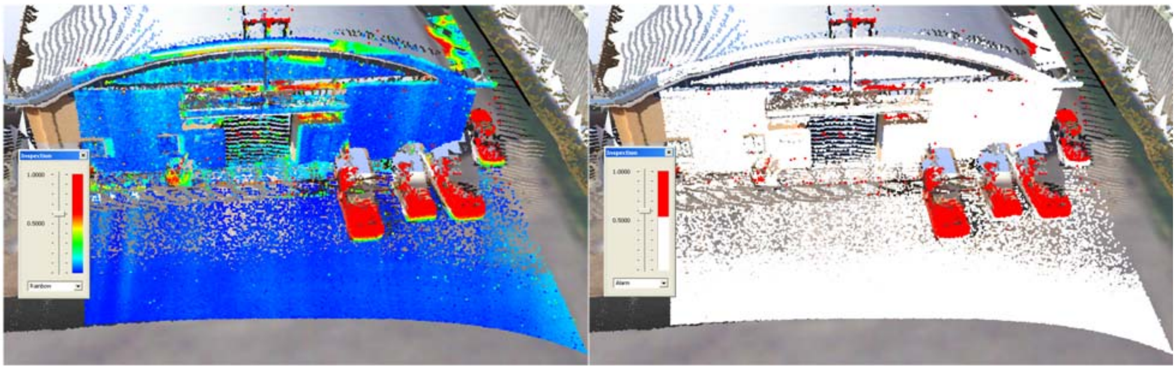
Figure 3. (a) Reference model, (b) verification scan and (c) automatically detected differences (in red).

For larger scenarios which has been modelled with the faster but slightly less accurate modelling technique, Continuous Scanning, Figure 4a presents a model acquired with this technique. Figure 4b shows the verification scan acquired at a later time. In Figure 5a and Figure 5b, the result of the inspection-operation is presented. The different colouring schemas can aid an inspector to focus on the areas changed.



(a) **(b)**

Figure 4. (a) detailed view of the Reference Model; (b) detailed view of the Verification Model.

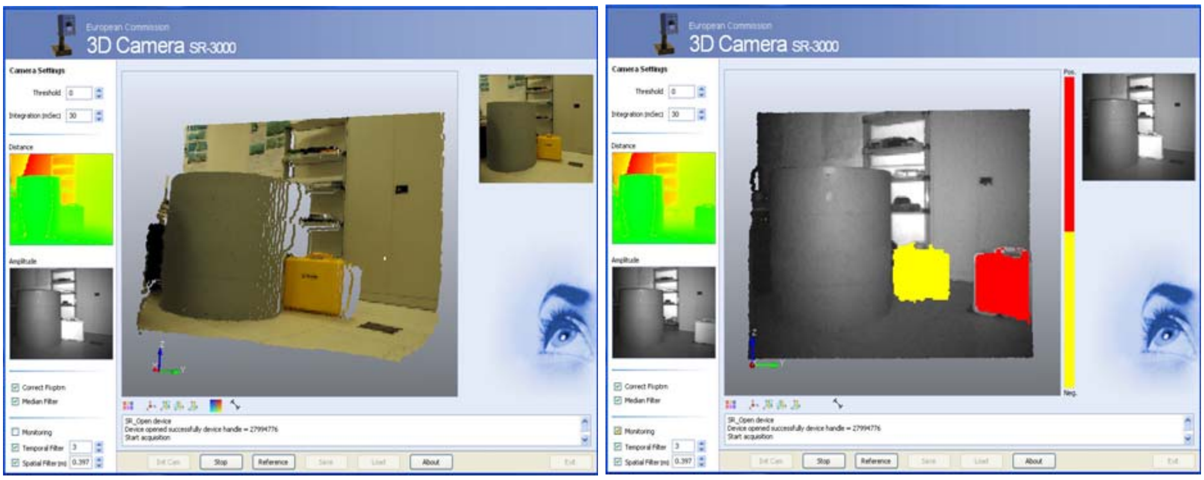


(a) **(b)**

Figure 5. Results of Scene Change Detection (a) with Rainbow colouring (Blue=0m, Red=1.0m difference); (b) with Alarm colouring (White = 0-0.5m, Red >0.5m differences).

3. 3D Surveillance

Conventional camera surveillance is based on 2D data. 3D information (i.e., distance or depth measurements) complements and enhances detection capabilities, including in-front-of-lens authentication. 3D detection capabilities enables robust alarm triggering, allowing for a fast and accurate determination of the position and measurement of objects in the field of view.



(a) (b)

Figure 6. (a) 3D and optical image combination (b) Alarm: in yellow object removed; in red object added.

New miniaturized systems capable of measuring an array of distances in real-time and without any mechanical scanning parts ("scanner less") are now entering the market. One way of achieving this is by using a custom solid-state area image sensor allowing the parallel measurement of the phase, offset and amplitude modulated light field that is emitted by the system and reflected back by the camera surroundings. Depth measurement is based on the time-of-flight principle.

3D surveillance algorithms analyse the 3D data collected from the real world by comparing newly incoming data with a reference 3D mapping. Distance discrepancies above a given absolute tolerance (e.g. a few centimetres) are immediately detected and an alarm generated. Figure 6 illustrates some measurements made in our laboratories where a suitcase was moved from on location (in yellow) to a new location (in red).

4. Containment and Weld Verification

3D laser imaging techniques are also used for containment and weld verification on containers with nuclear materials. Opening a sealed container and closing it again will inevitably cause changes to the three-dimensional structure of the surface which can be detected using 3D laser techniques. ***Figure 7*** (left) shows a system for the verification of the identity and integrity of closure welds on Pu containers. The container is rotated in front of a high-resolution laser line scanner which acquires a range image of the weld (***Figure 7***, right). The range image is compared to a previously acquired and archived reference.

Figure 7. Imaging setup for verifying the identity and integrity of closure welds on Pu containers (left), and range image of a weld acquired with this setup (right). The gray levels in the range image correspond to the elevations of each point above the container surface.

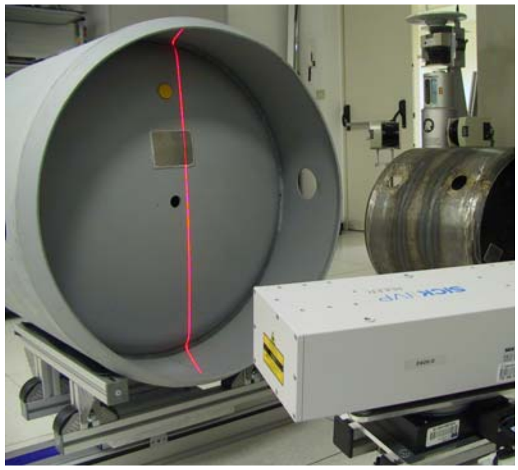


Figure 8. Measurement setup for the unique identification of UF_6 cylinders.

This is done by extracting a set of features for each 3D profile along the weld, i.e., for each row of the image in Figure 7 (right). Figure 9 (left) shows the features width, height and area extracted for a 100 mm long portion of a weld. This set of features serves as a unique "fingerprint" of the weld which cannot be easily copied or forged. Opening and re-welding a Pu container will completely change the fingerprint. Figure 9 (left) shows the result of the verification of a weld fingerprint against a reference fingerprint: An error measure is computed for each feature and for each possible shift between the two scans. Where the two weld profiles match (about. 40 mm in the example), all three feature functions exhibit a pronounced minimum. A weld different from the reference weld would not have such a minimum.

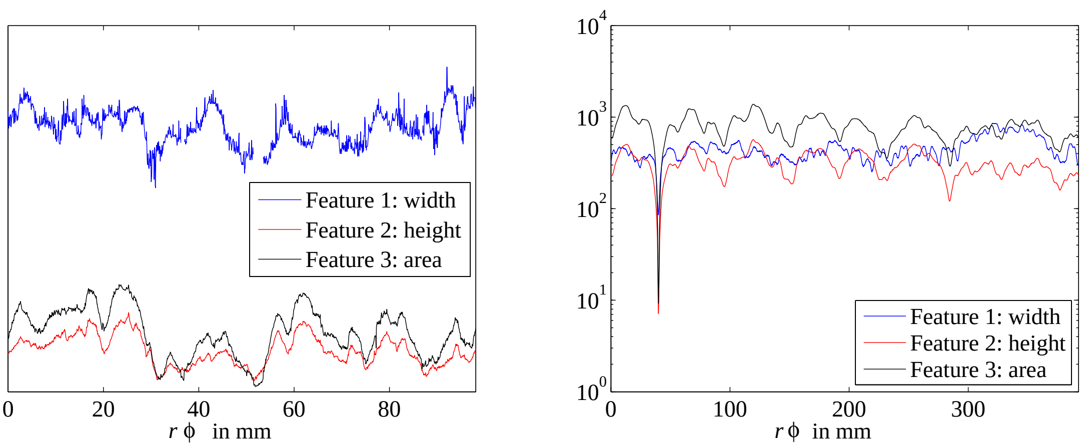


Figure 9. Width, height and area features extracted from each profile along a weld serve as its "fingerprint" (left). The error functions between this fingerprint and a previously acquired reference exhibit a pronounced minimum at the shift where the welds match (right).

When combining the error measures of the three feature function, the separation between a matching and a non-matching fingerprint is typically at least three orders of magnitude (factor of 1,000), thus allowing a reliable verification with low false alarm rate. It is sufficient to scan approximately 50 mm of a weld in order to achieve robust results. A single verification takes a few seconds and does not require any accurate mechanical alignment or calibration; the scanner and Pu container can be set up manually. The correct rotation angle between the reference scan and the verification scan is automatically determined by the system.

5. Unique Identification

Another application is the unique identification of objects: ***Figure 8*** shows a setup for scanning the face surfaces of UF_6 cylinders using a laser line scanner. This application requires a “self-authentication”, exploiting only the 3D surface structure of the cylinder face “as is”. Adding any dedicated identification tags, markers or transponders is not acceptable, both for operational reasons and due to the rough environmental conditions (heat, water, radiation) that the cylinders are exposed to.

The system in ***Figure 8*** scans the entire cylinder surface in a few seconds and exploits both the three dimensional shape of the cylinder face itself and the structure of the metal plate welded to the cylinder.

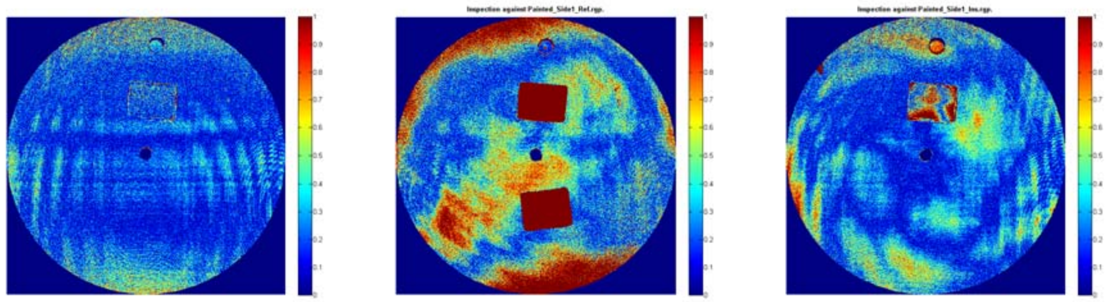


Figure 10. Results of matching a verification scan against the correct reference (left) and against two other reference scans (centre and right). Each point indicates the error between the reference and the verification surfaces, from dark blue (0 mm), to green (0.5 mm), to dark red (1 mm or above).

Figure 10 shows the results of matching a single scan against three different reference scans. The colours visualize the distances between the reference surface and inspection surface, ranging from dark blue (0 mm), to green (0.5 mm), to dark red (1 mm or above). The left-hand image shows the match against the correct reference while the centre and right-hand images show the results of matches against wrong references. In the left images, deviations are low and correspond to measurement uncertainties and calibration inaccuracies only. In the other two images, there are distinct red or yellow areas, indicating excessive deviations well above the measurement uncertainties. In the centre example, the deviations are very obvious since the metal plates were placed in different locations. But also in the right-hand example where the metal plate was carefully aligned, there are distinct deviations, both in the 3D structure of the metal plate and in the structure of the cylinder surface itself. Such differences cannot be perceived by the human eye; visually the two cylinder surfaces look identical. In the example, the cylinder surface can uniquely be identified as the left-hand reference.

6. Conclusions

– The use of 3D technologies in DIV allowed IAEA to carry out rapid and accurate DIVs in a complex facility, Rokkasho reprocessing plant in Japan, far faster and more accurately than had been possible in the past [6]. The system was successfully extended for outdoor applications to perform the verification of the facility buildings or modifications to a site.

- 3D Techniques were also successfully applied to large (> 1m of diametre) containers for self-authentication of welds and containment verification.

- JRC is working with IAEA to enhance their approved surveillance cameras with real-time 3D detection capabilities.

- The paper illustrate the potential of 3D technologies for many other applications in the areas of surveillance, containment, self-authentication and verification activities both indoors and outdoors. Further to those, applications are being considered on the use of realistic and dimensionally accurate 3D models from real-scenes as a way to visually integrate and present information from multiple sources, various types, different resolutions and time frames.

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Developing new techniques for detecting undeclared nuclear material and activities: UF_6 cylinder tracking system for uranium enrichment plants

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Abstract. During the uranium enrichment processes conducted in a gas centrifuge or gaseous diffusion enrichment plant, handling and storage of uranium hexafluoride (UF_6) is accomplished through the use of cylinders. This includes all UF_6 delivered to an enrichment facility, introduced into the cascade, withdrawn as product and shipped to the customer, withdrawn as tails and stored, and material that is sampled for analysis. Current methods used to track UF_6 cylinders require extensive manual entry of tracking/accountability data by operators. This manual entry is subject to transcription error and delays in timeliness and requires labor-intensive inventory procedures. Therefore, a need exists for real-time systems to automate this function to improve efficiency and effectiveness and to mitigate the risk of incorrectly entered or improperly manipulated data. A system that provided reliable, accurate, and automated cylinder identification could greatly improve the efficiency in maintaining a facility's inventory of nuclear material, support international safeguards, and increase the probability of earlier detection of UF_6 diversion and undeclared production activities.

The key to a more effective and efficient material accountability system is the development of a reliable, automated, tamper-resistant process for tagging and monitoring the location and status of UF_6 cylinders. This document proposes the use of a cylinder tracking system (CTS) to reduce the risk of false or incorrectly reported cylinder tare weights, diversion of nuclear material, concealment of excess production, use of undeclared cylinders, and misrepresentation of cylinder contents. The suggested system should include the capability to track UF_6 cylinders within uranium enrichment process areas. Additionally, a CTS could have the flexibility to incorporate other process data points as well as alarms and/or processing interlocks to detect and prevent abnormal operational conditions that are indicators of proliferation activities.

1. International Atomic Energy Agency Safeguards Objectives

More than 90% of International Atomic Energy Agency (IAEA) inspection effort at uranium enrichment plants involves verifying inventories, shipments, and receipts of Uranium Hexafluoride ($UF_{6)}$ cylinders. While the IAEA has been conducting verification activities at declared enrichment facilities for decades, growth in both the size and number of enrichment plants has increased the burden on inspectors, which necessitates changes to current inspector practices to improve effectiveness through the use of newer safeguards techniques. At a Technical Meeting on safeguards held at the IAEA in Vienna in April 2005,[1] working groups reviewed current practices for the purposes of identifying opportunities for increasing the IAEA's effectiveness. This review resulted in

[1] November 2005 Final Report on the IAEA Technical Meeting on Techniques for IAEA Verification of Enrichment Activities, Vienna Austria, 18-22 April 2005, Department of Safeguards, Division of Technical Support (SGTS), Division of Concepts and Planning (SGCP).

a strong recommendation for the development and use of "smart tags" for tracking material flow, inventory, and cylinders. Therefore, a cylinder tracking system (CTS) must support the following IAEA safeguards objectives for an enrichment facility:

1. The timely detection of the diversion of significant quantities of natural, depleted, or low-enriched UF_6 from declared flows throughout the plant and the mitigation of such diversion by the risk of early detection.
2. The timely detection of the misuse of the facility to produce significant quantities of undeclared product (at the normal product enrichment levels) from undeclared feed and the deterrence of such misuse by the risk of early detection.
3. The timely detection of the misuse of the facility to produce significant quantities of UF_6 at enrichments higher than the declared maximum, in particular highly enriched uranium (HEU), and the deterrence of such misuse by the risk of early detection.

2. UF_6 Cylinder Tracking System Objectives

The objective of a CTS is to help mitigate the potential for diversion or undeclared production of low or high enriched uranium at an enrichment facility.[2] The current IAEA practice for conducting enrichment facility safeguards is through the review of design information, (limited) observation of the cascade hall, measurement verification of shipper/receiver weights, performance of nondestructive assay (NDA) tests on cylinders containing UF_6, and evaluation of environmental samples taken inside and outside the cascade halls through particle and bulk analyses.

A CTS should be designed to track UF_6 cylinders from entry into the enrichment facility, throughout processing, and upon exit from the facility or into interim/final storage. The ability to monitor cylinders at critical process points will enhance the IAEA's ability to verify material weights and NDA activities within the enrichment facility. It should be noted that while the scope of this paper is to address the tracking of cylinders within an enrichment facility, the tracking capability can be expanded to encompass shipping/receiver differences from conversion facilities to enrichment facilities and to fuel fabrication facilities (from site to site or from country to country).

A CTS should be designed to support effective accounting of nuclear material and safeguards at a large enrichment facility. Cylinder tracking must provide immediate identification and verification of a cylinder selected either individually or as part of an inventory of cylinders. System capabilities must include the ability to track cylinders throughout operational processing cycles and to provide the possibility for alerts/alarms to indicate abnormal operational conditions. To meet these overall objectives, a CTS should be designed to meet the following specific objectives.

- **Real-Time Cylinder Tracking -** The system must have the ability to monitor the processing and storage of all cylinders that enter and are processed at the facility such as feed cylinders, parent product/intermediate cylinders, tails cylinders, customer cylinders, and sample containers.

- **Data Security and Access -** The system must have the ability to provide the cylinder data collected at critical process points to the IAEA and have the ability to be remotely monitored. The system must have the appropriate level of security, be tamper resistant, and be resistant to "spoofing." Furthermore, the operator must be able to track cylinders independently of the IAEA.

- **System Robustness -** The system must be able to survive harsh enrichment facility process environments while maintaining the capability of tracking cylinders and providing reliable access to safeguards data.

- **Increased IAEA Efficiency and Effectiveness -** The system must result in improved effectiveness and efficiency of IAEA inspections through the use of automated inspection tools and remote monitoring capabilities.

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[2] ORNL/TM-2006/98, *UF_6 Cylinder Tagging System for a Uranium Enrichment Plant*, August, 2006.

- **Planning for Defense-in-Depth –** The system design must take into consideration integration with other technical and programmatic elements for the purpose of building defense-in-depth into the safeguards system. Design elements can include, but are not limited to, interfaces with camera systems in key locations, incorporation of IAEA monitored scales where possible, incorporation of tamper-indicating devices such as seals, and incorporation of non-destructive analysis measurements. The system should also be integrated with limited-frequency unannounced access inspections, physical inventory verifications, short notice random inspections, etc., which are considered programmatic elements contributing to defense-in-depth.

- **Alarms and Interlocks -** The system must have the potential to interface with alarms and/or interlocks from process/operator facility equipment to indicate whether untagged cylinders are used at feed, withdrawal, sampling, or analysis process points. This system must be under the control of the operator but also have the ability to independently notify the IAEA that undeclared activities may have occurred at the facility.

- **Use of Existing COTS Technology –** The system design must maximize the use of existing commercial off-the-shelf (COTS) technology to reduce costs provided safeguards requirements can be met.

3. Cylinder Process Flow Requirements

A CTS should provide the capability to monitor the movement of all cylinders within an enrichment facility, including feed cylinders, parent product (or intermediate) cylinders, customer cylinders, sampling containers, and tails cylinders. Therefore, understanding the process flow is an essential element of the design of a CTS. Table 1 lists the primary cylinder types and their specialized characteristics. Cylinders are typically processed through an enrichment facility in the following manner.

1. Full feed cylinders are received at the site and stored temporarily.
2. The full feed cylinders are then loaded into the autoclaves (or heat chests) to feed the contents into the cascade.
3. Tails cylinders are filled at the tails withdrawal station.
4. Parent product (or intermediate) cylinders are filled at the product withdrawal station.
5. Material from parent product (intermediate) cylinders is then transferred to empty cylinders that have been received previously from the customer. These are called product cylinders.
6. Product cylinders containing enriched uranium are shipped from the enrichment facility to the customer.
7. Sample containers are filled during the transfer of the product from the parent product cylinders/intermediate to the customer cylinders.
8. Sample containers are transported to the laboratory for analysis (establishing assay and purity for parent product cylinders and customer cylinders).
9. Samples are also taken from feed and tails cylinders to support statistical sampling programs.
10. Empty feed cylinders are returned to suppliers or are filled with tails material.

3.1 Environmental Considerations

Cylinder tracking devices must have multiyear durability and the ability to withstand repeated steam baths in feed and transfer autoclaves (or heating cycles in electrically heated chests), being washed in cylinder cleaning facilities, exposure to corrosive gases, and operations in withdrawal-facility cold boxes. Detailed specifications for survivability in the operating environments should be developed and documented in a future System Requirements Document (SRD). These specifications should include, but are not limited to:

- survivability at –40°C to 140°C in both wet and dry conditions;
- survivability after exposure to hydrogen fluoride gas and to other corrosive gases;
- immunity to electromagnetic/radiofrequency interference and to ac/dc magnetic fields;
- survivability from vibration and from dropping;

- resistance to software viruses;
- tamper resistance and long battery life (greater than 5 years), if applicable.

Table 1. Cylinder types for handling UF_6[3]

Cylinder model	Typical Use	Weight of UF_6 [lb (kg)][a]	Nominal Diameter [in. (cm)][a]	Max psig[a]	Length [in (cm)][a]	Isotopic content limit (% ^{235}U)	Photo
48G	Tails	28,000 (12,701) approx. 14 tons	48 (122)	100	146 (370)	1	
48X	Feed/Tails	21,030 (9,539) approx. 10 tons	48 (122)	200	119 (302)	4.5	
48Y	Parent (or Intermediate) Product	27,560 (12,501) approx.14 tons	48 (122)	200	150 (380)	4.5	
8A	Product	255 (115.67)	8 (20)	200	56 (142)	12.5	
12B	Product	460 (208.7)	12 (30.5)	200	49.5 (126)	5.0	
30B	Customer Product	5,020 (2,227) Approx. 2 1/2 tons	30 (76)	200	81 (206)	5.0	
5A, 5B	Product	55 (24.95)	5	200	36 (914)	100	
1S	Sample	1 (0.45)	1.5 (38)	200	11 (280)	100	
2S	Sample	4.9 (2.22)	3-5/8 (90)	200	11 1/2 (290)	100	

[a]Units as designated unless otherwise indicated.

4. Radiofrequency Identification Devices: One Potential Technology

One promising technology being considered for application to a CTS is radiofrequency identification devices (RFID). RFID technology lends itself very well to deployment in automatic identification and data capture systems. RFIDs have many functional advantages over simple bar codes. The most significant advantage is that the RFIDs can be read automatically by interrogators (transceivers) and

[3]USEC-651, United States Enrichment Corporation, *Uranium Hexafluoride: A Manual of Good Handling Practices*, Revision 8, 1999.

require no operator interaction. This feature contributes significantly to the goal of a completely automated system. Additionally, certain types of RFIDs (e.g., active RFIDs) have the ability to receive and store data themselves so that specific information on changes to the cylinder's attributes can be stored internally.

A CTS can integrate data from cylinders with other sensors such as radiation detectors, gamma spectrometers, pressure and temperature sensors, accelerometers, limit switches, cameras, accountability scales, and other process information devices. This combination of data can provide enhanced verification of the material in cylinders, information related to cylinder processing, and insights on how cylinders and their contents have been used or modified. Figure 1 depicts a conceptual model of a CTS based on RFID technology for feed cylinders.

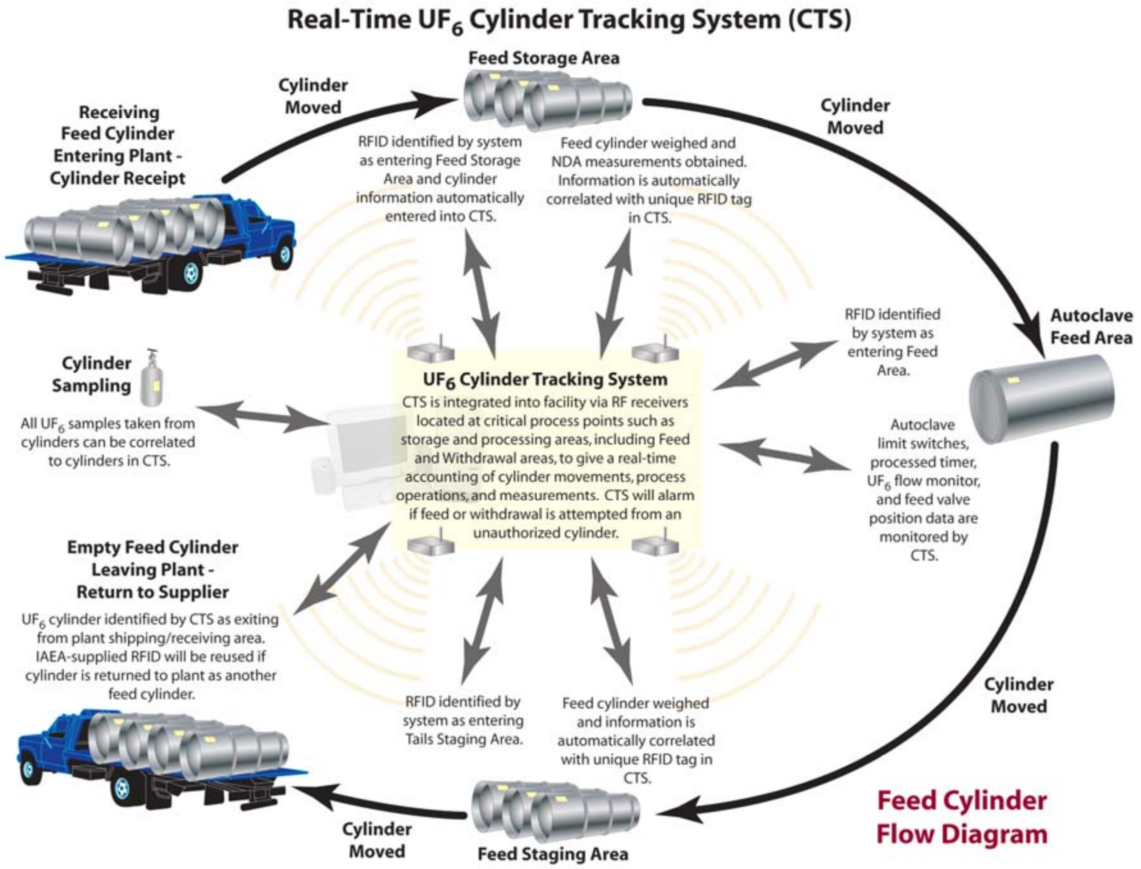

Figure 1. Conceptual Model of a CTS based on RFID Technology for Feed Cylinders.

There are three primary categories of RFID tags: active, passive, and semipassive. Each category of RFID tag consists of a silicon chip integrated with an antenna for receiving the transceiver's signal and responding with its unique information.

Active RFIDs are read-write tags that transmit information to strategically placed interrogators. These tags are battery powered and can be programmed for continuous, real-time signal transmission. The maximum working range for active tags is ~100 m from the tag to the interrogator. Active tags use a unique identifier similar to an encryption key with a time stamp. Active RFIDs offer the most advanced capabilities for combining data from other sensors, such as accountability scales, radiation detectors, and other process information devices. These tags, which can store larger volumes of data, are typically used for high-value-asset tracking.

Passive tags, which do not contain an internal battery, rely upon radiofrequency (RF) energy absorbed from the interrogator to supply the power for signal transmission. Because a passive tag relies on incoming energy, it has a much lower read range (about 9 ft). Most passive tags are either read-only or have very limited data storage capabilities. Rather, it responds to the interrogator with a unique identification signal. These tags are much simpler in construction than active tags and are more cost-effective. The disadvantage of passive tags is that they do not provide real-time, continuous tracking and are used primarily for discrete tracking through portals and discrete process points.

Semipassive tags behave like passive tags in that they do not continuously transmit signals. Instead, they await communication requests from interrogators. The semipassive tag is a read-only tag but differs from a passive tag in that it contains a battery. The battery is used by the tag circuitry to transmit a signal and thus increases its reading distance to up to 100 m. Like the passive tag, the semipassive tag is incapable of encapsulating additional sensors and may have battery life issues similar to those of active tags. Table 2 summarizes some of the characteristics of the various RFIDs and contrasts them with conventional bar-code technology.

One very important feature of RFID technology is that they can be easily integrated into a rules based software architecture where functional, operational, and design requirements are defined by the user.

Table 2. Tag specifications

	Active RFID	**Semipassive RFID**	**Passive RFID**	**Bar code**
Tag power source	Internal battery	Internal battery	Energy transferred from reader via RF	None
Power source lifetime	Up to 5 years	Potentially up to 10 years	Infinite (Reader is hardwired into electric grid)	Infinite (Reader is hardwired into electric grid)
Practical communication range	~100 m	~100 m	Up to 3 m	Up to 1 m
Type of communication	Can actively announce its presence (i.e., beacon); can be read remotely	Can respond only to a received signal; can be read remotely	Can respond only to a received signal; can be read remotely	Must be manually scanned; limited or no remote capability
Data storage	Read/write with the ability to store data from multiple sensors	Limited read/write abilities	Limited read/write abilities	Read only
Temperature limits	Thermal noise begins to occur around 40ºC, system failure at 80^{o}C	80^{o}C for most tags	80^{o}C for standard tags, 200^{o}C for special high-temperature tags	400^{o}C for special high-temperature tags
Sensor capabilities	Accelerometers, radiation detectors and other sensors can be included on the tag	Generally none, some special tags may include temperature sensors, etc.	Generally none, some special tags may include temperature sensors etc.	None
Cost per tag	~\$50–\$100	~\$50	~\$5 (more for special high-temp tags)	~\$2–\$5

4.1 Issues and Limitations of RFIDs

Each RFID tag type has strengths and weaknesses depending upon the operating environment and the desired results. The primary issues to consider when using RFID tags include security, susceptibility to tampering (i.e., “spoofing”), electromagnetic interference with other plant systems, environmental exposure, reading distance, and read interference from metal environments.

5. Preliminary Environmental Testing Results

Some preliminary testing was performed at the Oak Ridge National Laboratory (ORNL) to benchmark the performance of commercially available RFIDs and to determine their capability to withstand the harsh environments inside an autoclave.[4] The tags tested were all identified and represented by the manufacturers as high-temperature tags that could withstand the ORNL environmental testing parameters. The tags were tested over a temperature range from -40° C to +140° C to simulate the cold temperatures in the freeze boxes and the high temperatures in the steam heated autoclaves. The temperatures were held at +140° C for a period of 24 hours and then reduced back to ambient over a period of 4 hours. During the cold test, the tags were cooled to -40° C and held for a period of 72 hours before returning to ambient. After each test, the tags were read with the corresponding reader system to determine whether their performance was negatively affected by the environmental exposure. Figure 2 shows the RFID tags before and after the heat test.

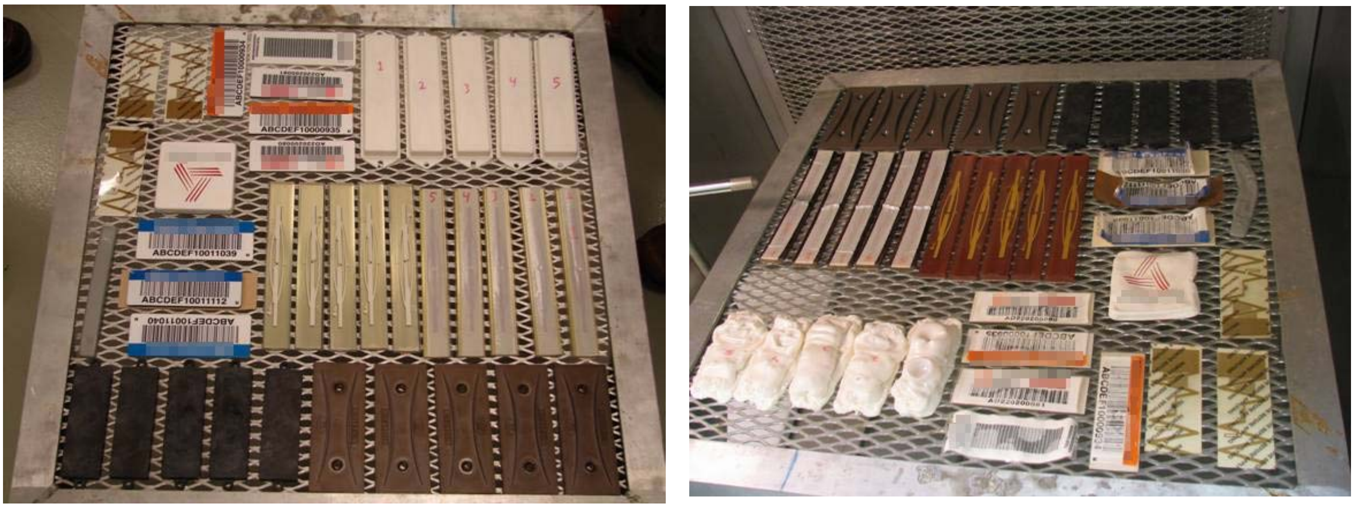

Figure 2. RFID Tags Before & After Environmental Testing +140° C.

The following list summarizes the results of this preliminary testing.

1. Two of the thirteen vendor supplied types of RFID tags tested survived the environmental testing with acceptable mechanical and performance characteristics.
2. Although some tags failed the heating cycle due to mechanical deformation, the tags themselves continued to be read into the system once they were removed from the environmental chamber. This indicates that the underlying electronics of the RFID tags are quite robust and capable of withstanding significant environmental cycles.
3. Standoff height between the RFID tag and the metal cylinder proved to be critical to minimizing the interference caused by the electromagnetic properties of the metal cylinder.
4. Not all the tags complied with industry established RFID formats. This is a serious problem because it can affect the ability of the tags to be integrated into a complete system. As the RFID industry moves forward, there is a natural convergence of standards. Large corporations are driving suppliers to use the GEN 2 protocol for inventory, thus ensuring technical support for the format.
5. There was generally good consistency in the read data for the tags. The tags were read consistently by the readers and did not fluctuate significantly within the time period the individual read testing was being performed.

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[4] ORNL/TM-2006-127, *Phase I Environmental Test Results of Radiofrequency Identification Devices for Application to a UF6 Cylinder Tracking System*, September 2006.

6. The size (i.e., surface area) of the antenna used by the reader system significantly affects the distance at which the tags can be read. The larger the antenna the better the read capability.
7. Only tags protected with a thick ceramic or ceramic-like coating were unaffected by the heating test.
8. For future applications, it may be necessary to obtain the most favorable RFID tag technology based on performance and then develop a custom environmental enclosure for the tags that will withstand the extreme environmental conditions required.

6. Conclusions and Recommendations

Reductions in inspection efforts could be achieved through the use of electronic identification tags, such as the RFIDs discussed in this report. Automated cylinder tracking provides an opportunity to reduce resource demands on the IAEA by using techniques to reduce the number and size of inspections necessary to verify plant operations, inputs, outputs, and inventories. Such techniques include the following:

- unattended monitoring of UF_6 cylinder accounting and handling operations;
- unattended monitoring to confirm consistency of process operations with declarations;
- remote monitoring, to the extent feasible; and
- integration with short-notice random inspections to confirm operational and material accounting information.

Additionally, the automated tracking of cylinders should provide a more effective accountability of material because it should provide immediate identification and verification of a cylinder that is selected either individually or as part of an inventory of cylinders. Automated cylinder identification can be used to verify the transfer of cylinders into and out of feed and withdrawal stations. The proposed CTS approach is designed to effectively meet the IAEA safeguards objectives for verifying material flows and inventories, the absence of excess production, and the absence of undeclared enrichment levels. In addition, the system can provide increased verification of the enrichment plant operations and material flows because all feed, product, and tails cylinders can be verified and the plant can be continuously monitored to ensure compliance with declared operational values.

To move forward efficiently with development of a CTS, the following actions should be considered.

1. <u>Develop detailed SRD.</u> To fully capture the requirements of an automated CTS, a concerted effort has to be made to consider all the operational, safeguards, security, safety, and other parameters to ensure that the final system will perform as intended and meet expectations.
2. <u>Perform Environmental Laboratory and Field Testing to Determine Best Available COTS Solution</u> Technologies that facilitate automated asset tracking, such as RFIDs, are experiencing a period of rapid growth and development. Several key industries are fully backing automated asset tracking and setting standards for compatibility and performance. Based on the SRD, tagging technologies should be identified, evaluated, and tested to see whether they are suitable for application to UF6 cylinder tracking. If none are available, some R&D may have to be undertaken to develop a suitable tracking system.
3. <u>Demonstrate Authentication of Data.</u> The CTS must be designed and tested to demonstrate to assure the Agency of its ability to authenticate safeguards data.
4. <u>Obtain Acceptance by Operators and Interface with IAEA</u>. Operator acceptance is vital to the success of a CTS. Therefore, a system administration approach must be determined that will best meeting the needs of both the IAEA and the operators.
5. <u>Field Test to Demonstrate Applicability of a CTS for Real-Time Cylinder Tracking.</u> A “proof-of-concept” field test must demonstrate that the proposed CTS can be used to perform real-time tracking of UF_6 cylinders in an operational facility.
6. <u>Evaluate Suitability for Integration with Other Plant Data Points and Process Systems for Safeguards Defense-in-Depth</u>. It should be determined whether a CTS can be integrated with other systems and plant data points and management controls for the purpose of building defense-in-depth into the safeguards regime. This can be accomplished by testing a CTS in a pilot facility.

7. Evaluate Increased IAEA Efficiency and Effectiveness from CTS Deployment. Additional effort must be expended to determine the effectiveness and efficiency improvements to the current IAEA safeguards regime at enrichment plants.