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CONTENTS

Illicit trafficking of radiological & nuclear materials: Modelling and analysis of trafficking trends and risks (IAEA-CN-148/127P)

D. York, G. Rochau, V. Cleary

Resolution of open core anomalies at LWRs (IAEA-CN-148/128P)

Y. Abushady, H. Dupreez, H. Cha, R. Carchon, G. Lee

IAEA safeguards implementation at dry spent fuel storage at Zaporozhe nuclear power plant (IAEA-CN-148/129P)

I. Perez Herrera, K. Murakami, H.-J. Schreiber, C. Olivieri, Y.-G. Lee, A. Ignachenko, V. Kulbakova

Successful implementation of an unattended safeguards approach for the transfer of spent fuel to interim dry storage (IAEA-CN-148/130P)

D.H. Hanks, A. Tolba

Strengthened safeguards system implementation at Serpong Nuclear Research Center — Batan (IAEA-CN-148/131P)

M. Imron, D.T. Jatmiko, M. Kadarusmanto

To achieve a nuclear material inventory in case of emergency (IAEA-CN-148/132P)

B. Massendari, F. Lemoine

A field exercise course to train IAEA safeguards inspectors in implementing the additional protocol and performing complementary access activities (IAEA-CN-148/133P)

B. Boyer, J. Valente, C. Gazze, J. Gilbert

The road to safeguards quality: An e-learning tutorial (IAEA-CN-148/134P)

K. Desson, D. Martin, Q.S. Truong, R. Keeffe

Education on nuclear safeguards for European nuclear engineering students (IAEA-CN-148/135P)

G. Janssens-Maenhout, A. Hamilton, T. Jönter, G. Stein, E. Martikka, A. Håkansson, K. van der Meer, B. Autrusson, R. Howsley, J. Baute, B.A. Burrows, M. Franklin, P. Funk, M. Hunt, J. Joly, C. Jorant, M. Kalinowski, P. Peerani, A.E. Poucet, K. Mayer, P. Schwalbach, L. van Dassen, L.-V. Bril, J. Gonçalves

Experiences in Germany with the implementation of the additional protocol (IAEA-CN-148/136P)

H.H. Remagen, A. Rezniczek, B. Richter

Preparation for implementation of the safeguards system in the Republic of Tajikistan (IAEA-CN-148/137P)

U. Mirsaidov, J. Salomov

Experience of the preparations for implementing integrated safeguards in Kazakhstan (IAEA-CN-148/138P)

G. Zh. Yeligbayeva

Experience of safeguards implementation at Rokkasho Enrichment Plant (REP) and an innovative safeguards approach for new plant (IAEA-CN-148/139P)

T. Iwamoto, D. Langlands, S. Aiuchi, A. Honma

- JNFL MOX fuel fabrication plant (J-MOX): Plant overview and safeguards considerations (IAEA-CN-148/140P)
K. Hiruta, M. Deguchi, K. Fujimaki
- Tools and techniques for safeguards training and knowledge management (IAEA-CN-148/141P)
N. Herber, D. Martin, Q.S. Truong, R. Keeffe, K. Desson
- Safeguards system in the Republic of Belarus (IAEA-CN-148/142P)
O. Gurko
- Experience of Thailand on conclusion of additional protocol (IAEA-CN-148/143P)
S. Biramontri, M. Aramrattana, C. Warawas
- Development of a safeguards database (IAEA-CN-148/144P)
S. Khan, M.S. Mulla
- A real-time system of accounting and control of nuclear materials and radioactive sources (IAEA-CN-148/145P)
I. Badawy, M.K. Mobasher, V.M. Naguib
- Proposed national actions for safeguards improvements in the United Republic of Tanzania (IAEA-CN-148/146P)
D. Shao
- ALOS satellite imagery utilization for safeguards (IAEA-CN-148/147P)
T. Tadono, M. Shimada, T. Hamazaki, N. Matsuura
- Potential applications of synthetic aperture radar (SAR) satellite imagery to nuclear safeguards (IAEA-CN-148/148P)
R. Saper, Q.S. Truong, M. St-Hilaire, K. Permiakova, J. Mulvie
- VITA-6.2: Advanced visual tool for information management (IAEA-CN-148/149P)
Z. Jacobson, Q.S. Truong, B. Houston, V. Taylor, N. Herber, A. El Gebaly
- Towards a strategy and conceptual framework for safeguards exploitation of optical satellite imagery (IAEA-CN-148/150P)
G.A. Borstad, Q.S. Truong, J. Lim, L.N. Brown
- The IAEA illicit trafficking database programme: Trends and patterns in confirmed incidents involving nuclear material (IAEA-CN-148/151P)
V. Turkin, W. Hammond
- Integrated information communication technology (ICT) tool for inspectors (IAEA-CN-148/152P)
R. Gaetano, J.-M. Becar, B. Sirajov, P. Titov
- An integrated approach to material balance evaluation (IAEA-CN-148/153P)
C. Norman, L. Bevaart, W. Fuhr
- Advances in safeguards IT security: Matching protection and actions to risks (IAEA-CN-148/154P)
J. Barton, L. Allen, J. Haluza, C. Schneider
- Creating an XML schema for enhancing nuclear safeguards information (IAEA-CN-148/155P)
C. Hobbs, J. Kidd

A geo-portal for data management and integration in the context of the additional protocol (IAEA-CN-148/156P)

E. Wolfart, L.-V. Bril, F. Bellezza, A. Ussorio, S. Contini, F. Mazza

IAEA safeguards information system re-engineering project (IRP) (IAEA-CN-148/157P)

G. Whitaker, J.-M. Becar, N. Ifyland, R. Kirkgoeze, G. Koevesd, L. Szamosi

Accounting and verification of data from the operators (IAEA-CN-148/158P)

I. Lovjagina

Measurements and verification challenges of drums containing HEU/LEU nuclear waste under safeguards using IQ3 in South Africa (IAEA-CN-148/159P)

L.J. Shayi, G. Marcon, M. Rasweswe

Advances in NDA of Pu in Pu-Be neutron sources (IAEA-CN-148/160P)

L. Lakosi, J. Bagi, C.T. Nguyen

Determination of the age of research-reactor fuel rods (IAEA-CN-148/161P)

J. Zsigrai, C.T. Nguyen, I. Sziklai-Laszlo

Statistics of the neutrons and gamma photons emitted from a fissile sample with absorption (IAEA-CN-148/162P)

A. Enqvist, I. Pázsit, S.A. Pozzi

Gamma-spectrometry for characterization of Pu-Be neutron sources (IAEA-CN-148/163P)

C.T. Nguyen

Development of an XRF analyser with preliminary energy selection filter for screening environmental samples (IAEA-CN-148/164P)

D. Donohue, V. Maiorov, Y. Ito, A. Nirschl, W. Raab

Performance and validation of COMPUCEA 2nd generation for uranium measurements in physical inventory verification (IAEA-CN-148/165P)

H. Ottmar, S. Abousahl, N. Albert, P. Amador, H. Eberle, H. Schorlé

Quantification of Cerenkov light for safeguard applications (IAEA-CN-148/166P)

V. Pratt, L. Bourva, R. Carchon

Safeguards and environmental measurements using Compton suppressed Ge detectors (IAEA-CN-148/167P)

T.-F. Wang, W.D. Ruhter, G. Price Russ

Superconducting ultra-high energy resolution Gamma spectrometers for nuclear safeguards applications (IAEA-CN-148/168P)

S. Friedrich, S. Ali, T.R. Niedermayr, I.D. Hau, S.F. Terracol, O.B. Drury

Autoradiography using optically stimulated luminescence for verification and safeguards applications (IAEA-CN-148/169P)

S. Miller, J. Tanner, M. Conrady, D. Anderson

Simulation smuggling study for shielded sources of ^{235}U , ^{238}U , ^{232}Th , ^{137}Cs and ^{60}Co by using γ -detection (IAEA-CN-148/170P)

S. Al-Rashdi, R. Al-Marbouei, S.A. El-Mongy

Prototype tomographic partial defect tester: Project status update (IAEA-CN-148/171P)

T. Honkamaa, A. Turunen, F. Levai, M. Larsson, R. Berndt, A. LeBrun

Analysis of U-particles by fission track-etch method and quadrupole ICP-MS (IAEA-CN-148/172P)

M. Lipponen, R. Zilliacus

Application of airborne gamma spectrometry to the detection of nuclear materials: Example of the French Helinuc system (IAEA-CN-148/173P)

S. Gutierrez, L. Guillot

Investigating the applicability of anions as indicators for verification of consistency of declarations (IAEA-CN-148/174P)

M. Wallenius, V. Badaut, K. Mayer

Improvement of non-destructive radioisotope search devices of smuggling by application of switchable neutron source Am-LiF (IAEA-CN-148/175P)

L. Meskhi, G. Cirekidze

Determination of uranium by the Brazilian Safeguards Laboratory — LASAL using "Davies & Gray/NBL" potentiometric method (IAEA-CN-148/176P)

P.D. de Barros, R.M.S. de Araújo, J.W.S. da Silva, F.M. Andrade

Simulation and test of spent fuel verification system for CANDU dry storage canister: A feasibility study (IAEA-CN-148/177P)

I.J. Park, G.H. Ahn, J.S. Kim, G.S. Min, S.-Y. Leeb, M.C. Miller

Towards the re-verification of in-process tank calibrations (IAEA-CN-148/178P)

J. Howell, M. Caviglia

Strategic safeguards considerations of the global nuclear energy renaissance (IAEA-CN-148/179P)

C. Mathews, C. Kessler, O. Elkhamri

Enhancement of proliferation resistance of a fuel cycle by applying the INPRO methodology (IAEA-CN-148/182P)

J.H. Park, M.S. Yang, J.K. Kim, E. Haas

Expected new role of IAEA in the area of transparency and proliferation resistance in advanced nuclear fuel cycle (IAEA-CN-148/183P)

M. Hori, N. Inoue

Giving transparency concepts a face-lift: Bridging the gap between old and new (IAEA-CN-148/184P)

C. Mendez, V. Cleary, G. Rochau, D. York

Nuclear fuel cycle transparency: An approach to support the global deployment of nuclear power (IAEA-CN-148/185P)

G. Rochau, D. York, C. Mendez, V. Cleary

Implementing a new reporting system to improve the security and the processing of accounting information (IAEA-CN-148/187P)

R.O. Nicolás

Safeguards approach to a new nuclear power plant in Argentina (IAEA-CN-148/188P)

L.I. Valentino

Determination of uranium contents and isotopic compositions in estuarine sediment standard reference material candidates by isotope dilution thermal ionization mass spectrometry (IAEA-CN-148/189P)

D. Guo, J. Cui, J. Tan, Z. Wu, Y. Liu

Approaches to strengthen China nuclear material control system (IAEA-CN-148/190P)

Q. Yang

Study on plutonium isotopic analysis in environmental samples (IAEA-CN-148/191P)

Z.Y. Chang, L.L. Li, Y.G. Zhao

A multilingual text mining based content gathering system for open source intelligence (IAEA-CN-148/192P)

F. Neri, N. Baldini

RADAR and CRISP — Standard tools of the European Commission for remote and unattended data acquisition and analysis for nuclear safeguards (IAEA-CN-148/195P)

P. Schwalbach, A. Smejkal, E. Roesgen, T. Girard

Tunable diode laser spectroscopy as UF₆ monitoring technique (IAEA-CN-148/197P)

N. Peter, M. Stein, A. Nadezhdinskii

Illicit trafficking of radiological & nuclear materials: Modeling and analysis of trafficking trends and risks

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Abstract. Concerns over the illicit trafficking of radiological and nuclear materials were focused originally on the lack of security and accountability of such material throughout the former Soviet states. This is primarily attributed to the frequency of events that have occurred involving theft and trafficking of critical material components that could be used to construct a Radiological Dispersal Device (RDD) or a rudimentary, improvised nuclear device (IND). However, with the continued expansion of nuclear technology and the deployment of a global nuclear fuel cycle these materials will continue to become increasingly prevalent, affording a more diverse inventory of dangerous materials and dual-use items. To further complicate the matter, the list of nuclear consumers has grown to include:

- Nation-states that have gone beyond the IAEA agreed framework and additional protocols concerning multiple nuclear fuel cycles and processes that reuse the fuel through reprocessing to exploit technologies previously confined to the more industrialized world.
- Terrorist organizations seeking to acquire nuclear and radiological material due to the potential devastation and psychological effect of their use.
- Organized crime, which has discovered a lucrative market in trafficking of illicit material to international actors and/or countries.
- Amateur smugglers trying to feed their families in a post-Soviet era.

An initial look at trafficking trends of this type seems scattered and erratic, localized primarily to a select group of countries (Figure 1). This is not necessarily the case. The success with which other contraband has been smuggled throughout the world suggests that nuclear trafficking may be carried out with relative ease along the same routes by the same criminals or criminal organizations.

Modeling and Analysis of Illicit Trafficking Trends

Clearly, the generation of an accurate analysis to model trafficking routes and nuclear trafficking trends requires:

- Cooperative, accurate coverage of incidences and the material involved, including the origination of the material, the seizure site and the proposed destination.
- Modeling of previous nuclear trafficking trends, routes, and smuggler involvement to generate an understanding of the supply and demand.
- Identification of the material in relation to risk for a particular material threat, trend, or local/regional activity.
- Noise filtration of inaccurate intelligence from the analysis that may impede the incident interpretation of route and risk.

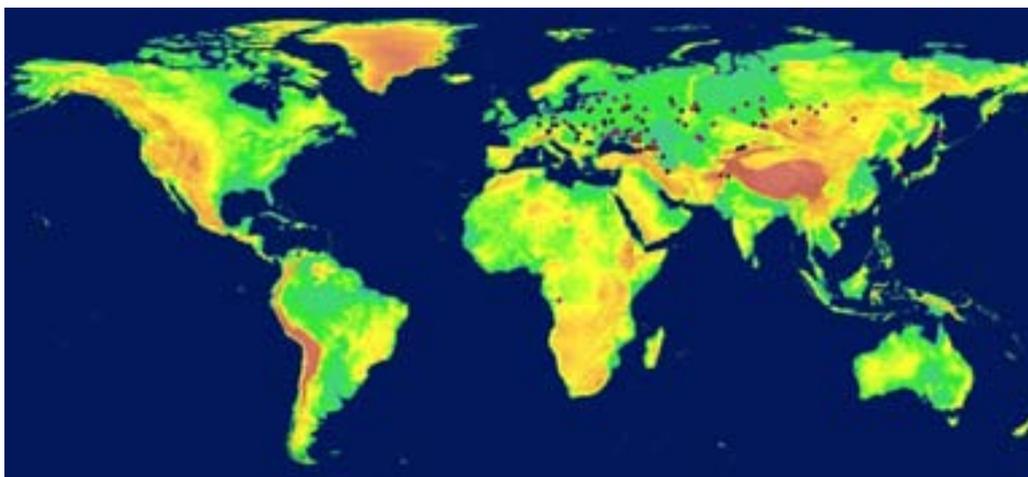


FIG. 1. Global Trafficking Incidents (2002).

In collecting reports on illicit trafficking of nuclear and radiological material or related equipment, it is expected that some information or authenticity may be lacking. However, it is important to note that the collection and modeling of multiple incidences to provide an activity analysis will convey an overall supply and demand trend indicative of illicit nuclear activity and intentions.

Illicit Nuclear Material Trafficking Analysis Software

Software to analyze trafficking routes should incorporate cognitive modeling and advanced information processing using compilation of multi-source intelligence and previously designed probabilistic risk assessment (PRA) techniques. The backdrop of such an analysis is weighted towards previous trafficking activity (Figure 2).

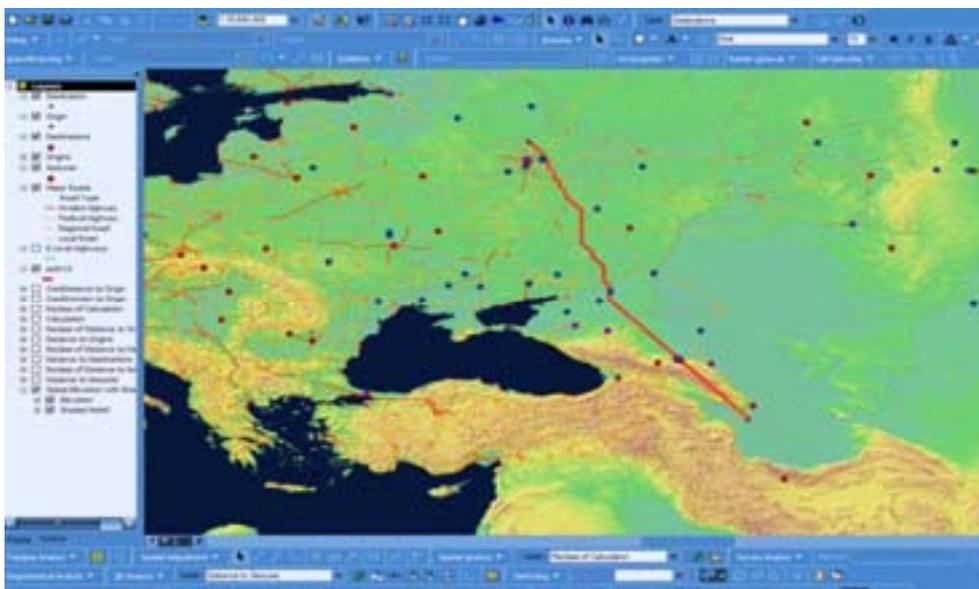


FIG. 2. Route analysis based on previous trafficking trends.

This analysis will also include an objective determination of proliferation and trafficking trends via current methodologies for predicting the potential for development of weapons of mass destruction. This can be achieved via modeling of trafficking routes and trends, while incorporating the material risks associated with the particular nuclear or radiological materials involved. It is then possible to design a risk based probability for the deployment of an unconventional weapon.

Figure 3 is a 3D model of a radiological dispersal device using a specified amount of cesium-137. The detonation site is marked by the red circle and the color-coded columns indicate levels of radiological dosimetry. The model is also confined to environmental conditions that would affect the spread of radiological material.

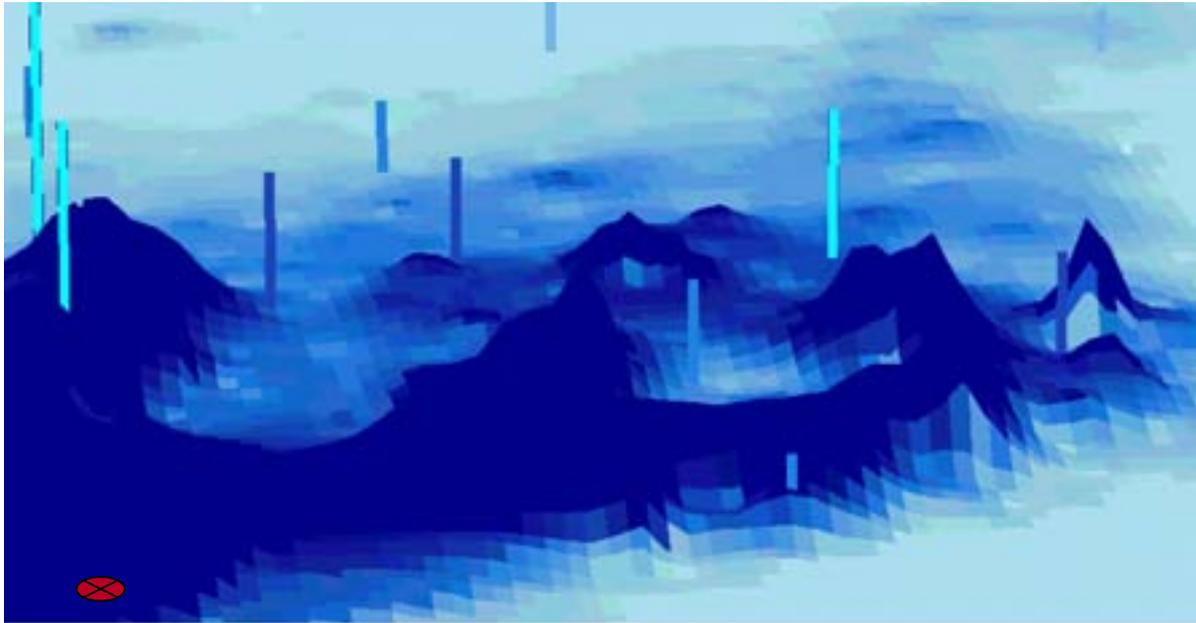


FIG. 3. Potential RDD risk analysis.

Preliminary Conclusions

The development of nuclear energy will amplify the accessibility to nuclear and radiological material. This may further increase the potential for latent proliferation. It is important that enough information and intelligence is collected under a cooperative framework to efficiently address and combat the spread of illicit trafficking. This intelligence and information must be analyzed using modeling schemes that accurately depict illicit trafficking activity. Accomplishing this will allow us to understand the scope and intentions of illicit nuclear and radiological trafficking, and be prepared for future trafficking incidences.

D.L. York et al. (2006) "Global Modeling and Risk Analysis of Illicit Nuclear and Radiological Material Trafficking," *Article Submitted for Publication*.

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Resolution of open core anomalies at LWRs

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Extended Summary: Safeguards anomalies caused by surveillance failures or inconclusive surveillance during the open core period at light water reactors (LWRs) are the most difficult anomalies to resolve, as they are usually detected after refuelling and core closure and the re-start operation of these reactors. The absence of diversion of irradiated fuel, whether in the core or in the pond, with the use of undeclared fresh fuel, dummies or even without replacement — which is less likely — needs to be confirmed. The IAEA has no authority to request immediate stoppage of reactor operation to verify the reactor cores, as such action would cause a high interruption of power production and cost. Therefore, resolution of these types of anomalies is of high safeguards value, particularly before the broader safeguards conclusion of the absence of undeclared nuclear material and activities in the State, in particular the absence of undeclared reprocessing activities or spent fuel storage, has been drawn.

This paper discusses the resolution of three different types of open core anomalies at LWRs where different methods, mainly based on analysis and advanced non-destructive assay (NDA) methods, are successfully used.

The first type of open core anomaly is related to the late detection of the failure of open core surveillance after refuelling, core closure and restart operation. The core fuel under surveillance is not verified by NDA but only by item counting and item identification in the core. This is the most common anomaly type, where the removal of either irradiated core fuel or spent fuel could be replaced by fresh fuel assemblies in the core. The resolution of this type of anomaly is done through the use of an advanced NDA method, mainly a specially designed fork detector irradiated fuel measurement system (FDET), to assess the irradiation period of different core fuel assemblies upon discharge at the end of the cycle. This solution is more likely to be applied to pressurized water reactors (PWRs) where normally the entire core fuel assemblies are discharged temporarily to the spent fuel pond before refuelling.

The second type of anomaly is related to the failure of surveillance during the core opening period until the discharge of the entire core fuel assemblies and the verification of the entire irradiated fuel (core fuel and spent fuel) in the spent fuel pond — 100% verification — using an improved Cerenkov viewing device (ICVD). The resolution in this case is based on detailed analysis of the possibility to discharge and reload the entire core, introduce the number of undeclared fresh fuel assemblies in the core, close the reactor, perform undeclared operation of the reactor, shutdown, open the core again, and discharge the entire core. In addition to this analysis, the ICVD is used to confirm the absence of unirradiated dummies or very low irradiated spent fuel assemblies.

The third type of anomaly is related to an undeclared emergency core opening and closure with the declaration that no core fuel assemblies have been discharged, while the surveillance was inconclusive shortly before this undeclared core opening. The resolution of the anomaly in this case is based on the prompt verification of all spent fuel assemblies in the pond using the advanced ICVD, namely the digital Cerenkov viewing device (DCVD), to confirm mainly the cooling time of the assemblies and consequently the absence of undeclared core fuel assemblies discharged during the undeclared emergency core opening.

IAEA safeguards implementation at dry spent fuel storage at Zaporozhe Nuclear Power Plant

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Abstract. In 2001, a dry spent fuel storage was put into operation at the Zaporozhe nuclear power plant in the Ukraine. The IAEA developed a safeguards approach for spent fuel transfers based on the use of a fork detector (FDET). In addition, containment and surveillance (C/S) measures were put in place to maintain continuity of knowledge over the verified assemblies. The implementation of the safeguards approach was carried out with excellent cooperation from the facility operator. The FDET measurement system enables the IAEA to verify the operator's declaration on the nuclear material content. The results are shared with the operator in order to check the correctness of the calculated burn-up with regard to the nuclear safety concern. This paper describes the approach implemented and discusses the results obtained.

1. Introduction

The dry spent fuel storage area at the Zaporozhe nuclear power plant began operation in August 2001. The first part has a capacity of 110 ventilated storage containers, which is expected to increase to 350 containers in the future. This is an open-air area of approximately 50 x 50 m located within the plant perimeter. The containers are stored in parallel rows, separated by approximately 70 cm. Number/letter coordinates identify the locations.

During the second half of 2001, the prototype fork detector (FDET) for WWER-1000 fuel, with a distance of 410 mm between the arms, was delivered and commissioned with positive results at the Zaporozhe nuclear power plant. Sufficient neutrons were detected from the spent fuel, as low as 12 GWxd/tU (only one cycle in the core). The results of the testing are shown in Figure 1.

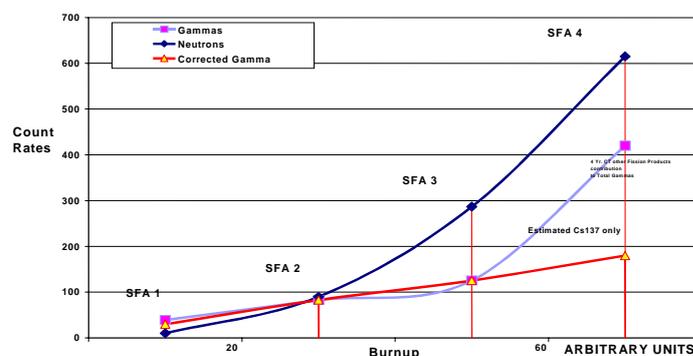


FIG. 1.

2. IAEA verification activities

The IAEA Safeguards Division of Operations C developed a safeguards approach, based on stepwise verification activities to be carried out during the loading, transport and final storage of the spent fuel assemblies. The approach comprises two parts: (1) continuity of knowledge (CoK) measures and FDET verification measurements; and (2) application of dual containment and surveillance (C/S) systems at the dry spent fuel storage area.

The CoK measures include the following:

- (a) The installation of 2 all-in-one systems (ALIS) and 1 digital single camera optical surveillance system (DSOS) with high frequency picture taking (see Figures 2 and 3);
- (b) Attachment of 2 VACOSS electronic seals (see Figures 2 and 3);
- (c) Serial number identification of individual assemblies before loading;
- (d) FDET measurements of the 24 spent fuel assemblies (see Figure 2);
- (e) Verification by item counting and serial number of the fuel assemblies already loaded into the container;
- (f) Sealing, with 2 IAEA metal seals, the all-weather lid of the ventilated storage container (VSC) containing a full MSB.

TOP VIEW OF THE REACTOR HALL AT WWER-1000 SHOWING FDET POSITIONING AND C/S DEVICES

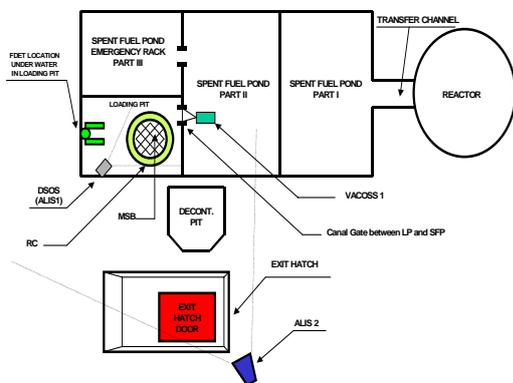


FIG. 2.

SIDE VIEW OF TRANSPORT CORRIDOR AT WWER-1000 SHOWING C/S MEASURES ATTACHMENT 2

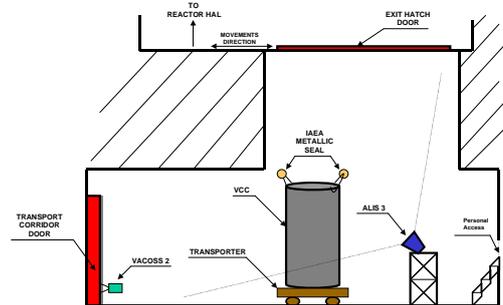


FIG. 3.

2.1. Verification of operator declared burn-up, FDET measurements

From 2001 to mid- 2003, data collection from GRAND-3 was performed using the GrandCollect software. In October 2003, a new software FDMS, developed by Los Alamos National Laboratory, was tested and is currently being used on a routine basis. FDMS software provides real time graphical representation of the data being collected.

Three analysis options are available: neutron versus burnup; gamma/burn-up versus cooling time; and neutron versus gamma.

3. Evaluation of measurement results

Figures 4 and 5 show the analysis results for neutron versus burn up and gamma/burn up versus cooling time tests, respectively. The sensitivity of verification was illustrated by the capability of the IAEA to detect inconsistencies in the operator declared burnup. The cases are described below:

- (a) Incorrect burn-up declaration of a spent fuel assembly;
- (b) Unnoticed swapping of the loading of 2 fuel assemblies; and
- (c) Some typing mistakes.

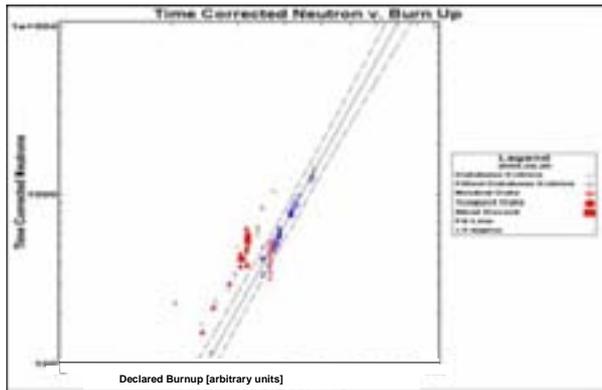


FIG. 4.

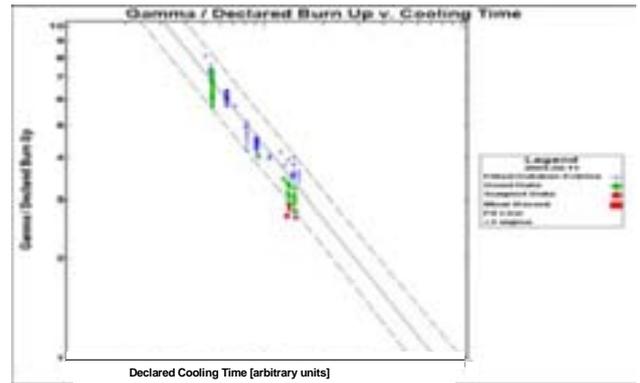


FIG. 5.

Successful implementation of an unattended safeguards approach for the transfer of spent fuel to interim dry storage

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Abstract. In today's commercial nuclear environment spent fuel bundles are being removed from wet storage areas, encased in containers or baskets and stored at interim dry storage facilities all over the world. Growing demand on resources are challenging the IAEA to develop better, more inventive ways for maintaining continuity of knowledge of the spent fuel while it is being transported by operator-controlled vehicles to facilities for storing large caches of spent fuel. This formidable task becomes less intimidating through inspector pre-planning, approved procedures and professionally installed, reliable containment and surveillance (C/S) equipment. Safeguards measures for transferring spent fuel under traditional safeguards include the on-site inspector who monitors the loading of spent fuel into transport/storage containers and constantly observes the item's physical location throughout transport to an interim dry storage facility. However, currently the number of inspector days in the field has been drastically reduced through the installation and operation of unattended monitoring systems that maintain continuity of knowledge of the spent fuel. Under a State-level approach to safeguards, spent fuel transfers may be monitored by a regime of unannounced inspections, with a low probability for post-transfer detection of a diversion of nuclear material. Through a combination of low frequency unannounced inspections and an unattended spent fuel transfer safeguards approach, the probability of post-transfer detection for a diversion are above 50%. The IAEA's unattended approach to the monitoring of spent fuel transfers has already eliminated the need for continuous inspector coverage and minimized the number of person-days required per facility, thereby allowing valuable inspector resources to be redirected to other, more demanding areas of the world for safeguarding efforts.

Cernavoda Nuclear Site located in Romania is currently the first of its kind being safeguarded under an unattended spent fuel transfer approach. This paper examines the approach, viewed as a successful case study that has demonstrated the viability of the improved method for monitoring and maintaining continuity of knowledge of spent fuel being transferred from an operating CANDU reactor to an interim dry storage facility.

1. Introduction

Finalizing an implementation plan for a new concept, such as an unattended approach for safeguarding nuclear spent fuel transfers, is usually difficult relative to determining relativistic failure mechanisms and mitigating their impact. Aspects of each suspected mode of diversion associated with an unattended spent fuel transfer approach were fully addressed in the case of the spent fuel transfer from wet pool storage to an interim dry storage module at Romania's CANDU reactor. Allowing the process of transfer to proceed according to the operator's scheduled needs is essential for an on-load reactor's continuous need for storage space in the spent fuel pool.

Strategic planning, procedural compliance and quality management are all key elements that were used in establishing unattended monitoring along the outdoor transport path, previously observed by a resident IAEA safeguards inspector. Person-days of in-field work by inspectors could then be redirected toward other areas of the world so as to meet safeguarding requirements resulting from the expanding global nuclear power supplied electricity market. As an additional benefit, this process also lowered individual radiation dose exposures previously experienced by inspectors who were in constant proximity to highly irradiated spent fuel bundles.

Technological advancements in flow monitoring of irradiated spent fuel components, using neutron/gamma radiation peak detection devices, have provided inspectors with a greater probability of detecting a diversion, post transfer. Reproducibility of a sequence of events for each basket/container of spent nuclear fuel transferred is now possible, using stand-alone equipment placed at key measurement points between material balance areas. Utilizing capabilities such as remotely monitoring surveillance equipment, the IAEA maintains nuclear material under constant observation, ensuring continuity of knowledge over each transfer.

Successive planning has helped to attain the strategic objective of safeguarding CANDU spent fuel being transferred at the Cernavoda site in Romania, while the State undergoes a transition to its State-level approach (SLA) for integrated safeguards. Under the successive plan, traditional safeguards measures are gradually replaced by a more comprehensive approach, as outlined in Romania's SLA which aims to promote enhanced efficiency and effectiveness, especially in the area of unannounced inspections. Numerous routine inspections planned during each spent fuel transfer campaign may be reduced under the SLA to only a few unannounced inspections performed in a randomized fashion.

2. Implementation

2.1. Strategic plan

The strategic implementation plan for Cernavoda is modelled after an approach discussed in a presentation at the 2004 annual meeting of the Institute for Nuclear Materials Management (INMM)¹. The establishment of Cernavoda's spent fuel transfer safeguards system and the transition to an unattended approach resulted in significant savings of IAEA expenditures during the first year of implementation. Each step of the transfer process for CANDU spent fuel bundles required that newly designed effective safeguards measures be put into practice, to ensure continuity of knowledge over the route from the spent fuel's point of origin to its final position in the interim dry storage facility (Figure 1). Cernavoda's containment and surveillance (C/S) requirements for each of these stations are summarized below:

- Spent fuel previously sealed by underwater ultrasonic seals is moved to a dedicated staging area of the spent fuel pool for transfer, after being unsealed and item counted by a visiting inspector. Underwater surveillance maintains continuity of knowledge over the spent fuel transfer staging area until CANDU spent fuel bundles are ready to be moved by the operator to a tilting device for basket loading. Gross defects detected by non-destructive analysis (NDA) is performed by means of an installed system, such as a core discharge monitor or bundle counter in the case of a CANDU-type facility, prior to the spent fuel being sealed.
- Cernavoda's operators send an electronic or "mailbox" declaration to the IAEA headquarters in Vienna via an encrypted and authenticated transmission, describing their intended programme of transfers. A post-transfer mailbox declaration is sent to IAEA headquarters in order to convey nuclear material accountancy updating. However, should there be an unexpected delay in the programme, a well-clarified amendment may be accepted.
- Inspectors remotely monitor re-batching of spent fuel bundles, using underwater surveillance, when they are loaded into dry storage baskets from the tilting device. Verification measures at the time of re-batching include item counting of each assembly and maintaining continuity of knowledge over the spent fuel basket after it is covered. Underwater cameras installed at the basket loading area eliminate the need for the presence of a resident inspector in order to observe basket movements and on-site item counting.
- An operator-owned transport vehicle equipped with an IAEA surveillance system is loaded with the transport flask that carries the welded closed spent fuel basket. A mobile camera is mounted on the transport vehicle, using a specially reinforced camera-mounting bracket that has greater

¹ Hanks D., Liguori C., Tolba A., Olivieri C., Aparo M., Implementation of an Unattended Safeguards Approach to the Transfer of Spent Fuel Into Interim Dry Storage, 45th Annual Meeting of INMM (2004).

resistance to vibration than a normal mounting bracket. The camera's field of view is directed toward the spent fuel transport flask and continuously monitors its presence throughout transfer to the interim dry storage.

- Mounted on top of the transport flask is a neutron monitoring device that tracks the movement of spent fuel baskets leaving Cernavoda's spent fuel wet storage pool to the site's interim dry storage facility. This apparatus, known as a mobile unit neutron detector (MUND), consists of a detector, memory chips and batteries encased in a self-contained box that can be serviced by a visiting inspector, as needed. Time stamped data are then retrieved and reviewed by an inspector performing an unannounced inspection or technical service visit, thereby eliminating the possibility of a nuclear material substitution.
- Remotely monitored surveillance covering Cernavoda's interim dry storage facility records all movements around its storage modules 24 hours a day. Review of this transmitted information is performed at IAEA headquarters in Vienna and can be recalled at any time to recreate a sequence of events (SOE) for each spent fuel basket transfer being deposited to a particular silo.
- A silo entry gamma monitor (SEGM) is mounted at the mouth of each of the declared storage silos to be used during the spent fuel transfer campaign. The presence of spent fuel and direction of flow are verified as each basket passes by a short string of gamma detectors connected to an encrypted recording apparatus. Directional data are collected in a digital compilation device that can be downloaded during an unannounced inspection or a technical service visit.
- Engineering reliable equipment is important in assuring that continuity of knowledge is maintained throughout the spent fuel transfer route. High quality reliable equipment demands a proven routine of maintenance and servicing that will provide a guaranteed minimal failure rate over the course of its useful life. Equipment protection from the environment using special design techniques is necessary for trustworthy monitoring. Effects of rain, snow and heat must be addressed in the establishment of a comprehensive, reliable system of monitors and data collection components. Cabinets or connectors must have the ability to withstand extreme environmental changes, because most are subjected to outside weather conditions.

2.2. *Procedures compliance*

Specific procedures apply the IAEA's safeguards criteria concerning spent fuel designated to be transferred to difficult-to-access storage. Cernavoda's spent fuel stored in each unit's wet storage, under dual C/S systems, require that the fuel be NDA verified prior to being sealed, and will not require re-verification by NDA measurements at the time of basket loading. Item counting at each re-batching of spent fuel bundles into spent fuel baskets is therefore a sufficient means of verification for this site. Procedures include precautions and limitations that inform the inspector of the safeguards criteria, operator requirements, sealing arrangements and other essential information that improves the probability of successfully safeguarding nuclear material during transfer of spent fuel.

A sequence of events is documented for each basket as it is filled, welded, transported to interim dry storage and then finally deposited into its storage silo, through the use of remotely monitored surveillance equipment and manually downloaded unattended monitoring systems (UMS). Comprehensive comparisons are made between the IAEA's SOE constructed from UMS and the remotely monitored surveillance equipment to the operator's declared SOE sent to the IAEA via the mailbox declaration.

Unannounced inspections are included in Romania's SLA for integrated safeguards, which encompasses the venue of spent fuel transfers at the Cernavoda site. Determination of the frequency and monitoring characteristics of such a programme is explained in internal IAEA documents associated with unannounced inspection principles.

2.3. *Quality management*

Quality management measures were developed by outlining a set of goals to be achieved while maintaining good corporate acceptance of IAEA expectations by Cernavoda's operators. Inspector competency, immediate problem resolution and improved efficiency are objectives that need to be met

before the new requirements could be introduced to the Romania State authority, CNCAN, and Cernavoda's operational management.

Safeguards techniques and minor modifications to installed equipment that might cause significant safety issues were evaluated in-house prior to presentation to the operator, to eliminate possible safety related concerns. IAEA inspectors demonstrated competency during the on-site implementation of these techniques, equipment servicing and the conduct of safeguards activities. This was done to avoid any concerns on the part of the operators during the spent fuel transfer campaign and resulted in significant confidence in the IAEA's ability to remotely monitor all related activities.

Immediate problem resolution was enhanced through continuous coordination between IAEA inspectors in Vienna and operators in Romania via mailbox declarations and automated acknowledgment. Instant messages were sent to the IAEA through an encrypted Internet information conduit, which protects dispatched information and feeds confirmation of receipt back to Cernavoda. Problems were thereby communicated directly to the inspector's computer in Vienna, who then chose the best course for clear problem resolution.

Improved efficiency of an unattended monitoring approach has given the operator control over his own spent fuel transferring schedule and allowed the IAEA to conduct remote observation throughout each step of the process during the spent fuel transfer SOE. Spent fuel attribute confirmation, using gamma and neutron detectors, allowed for the recreation of an attempted diversion event which could be detected post-transfer once the nuclear material had been deposited into the assigned silo. These permanent records, along with digital surveillance recordings, helped to ensure that any questionable event could be reviewed later at IAEA headquarters for possible anomaly detection.

Error reduction during review of acquired data is only possible when an inspector is adequately trained in all the new site-specific safeguards techniques. Hazardous nuclear materials transported by operators at Cernavoda are now being successfully tracked through a combination of sound safeguarding solutions and hands-on training of equipment operation. Decline in the overall error rate can also be attributed to improved monitoring equipment components and their task engineered man-machine interface. Some of these improvements have help to minimize the impact of extreme environmental changes caused by outside weather conditions, using water resistant and self-cooling cabinets/boxes.

Another element of quality management is process ownership by the inspector, which is vital to maintaining continuity of knowledge over the SOE encountered when up to one hundred basket transfers are performed during a six-month spent fuel transfer campaign. Details of constant daily movements, reported by the operator to the IAEA via mailbox declaration for each basket, must be reviewed on a daily basis. The IAEA site officer, facility officer and their alternates are responsible for spent fuel basket post-transfer reviews and have taken ownership of safeguarding the entire process.

3. Successive planning

Drastic reduction of inspector days in the field has been experienced at the Cernavoda site since the implementation of the unattended spent fuel transfer safeguards approach during the spent fuel transfer campaign in 2005. The continuation of this approach under an SLA for integrated safeguards would vastly improve the post-transfer detection of a possible nuclear material diversion when compared to other unattended procedures. This safeguards regime, which has already provided full continuity of knowledge over each spent fuel transfer throughout a six-month spent fuel transfer campaign, would be further enhanced through the introduction of unannounced inspections in Romania's SLA.

Planning for the implementation of unannounced inspections, as part of Romania's SLA for its CANDU nuclear power plants, must also take into account additional unannounced inspections required for Cernavoda's spent fuel transfer campaigns. Spent fuel transfer campaigns that include multiple transfers of individual baskets composed of re-batched irradiated CANDU spent fuel bundles randomly selected for inspector verification would be evaluated according to their frequency of

occurrence. A larger number of spent fuel basket transfers per campaign means more unannounced inspections must be performed in order to maintain an equivalent probability for possible diversion detection.

Site officer responsibilities are frequently turned over to another inspector, as dictated by an active personnel rotation plan at the IAEA. Although this turnover of responsibilities in Romania includes all aspects of the natural uranium fuel cycle, the Cernavoda site incorporates additional safeguards techniques for monitoring spent fuel transfers in an unattended mode. The site officer at the Cernavoda site must therefore be trained to a high standard, and must be able to address complicated aspects of mailbox declarations, remote monitoring, sequence of events tracking and unannounced inspections. Successive plans, based on process knowledge, must be comprehensive and up to date, to assure proper execution of responsibilities at the site officer level.

4. Conclusions

Programmatic use of strategic planning, procedural compliance and quality management have allowed the successful establishment of an unattended safeguards system along the outdoor transport path of CANDU spent fuel at the Cernavoda Site. IAEA safeguards inspectors that were previously required to be resident during the spent fuel transfer process are now free to perform duties in other areas of the world so as to meet safeguarding requirements resulting from the expanding global nuclear power supplied electricity market. Lower radiation exposures and reduced monetary expenditures normally associated with positioning an inspector in close proximity to these baskets of irradiated spent fuel have been achieved using this proven method of remote monitoring of each transfer and the establishment of real-time SOE.

Continued goal attainment at the Cernavoda site will require resourceful successive planning during the implementation of Romania's SLA. The new approach will further enhance safeguards unattended spent fuel transfers through the introduction of unannounced inspections performed during a spent fuel transfer campaign.

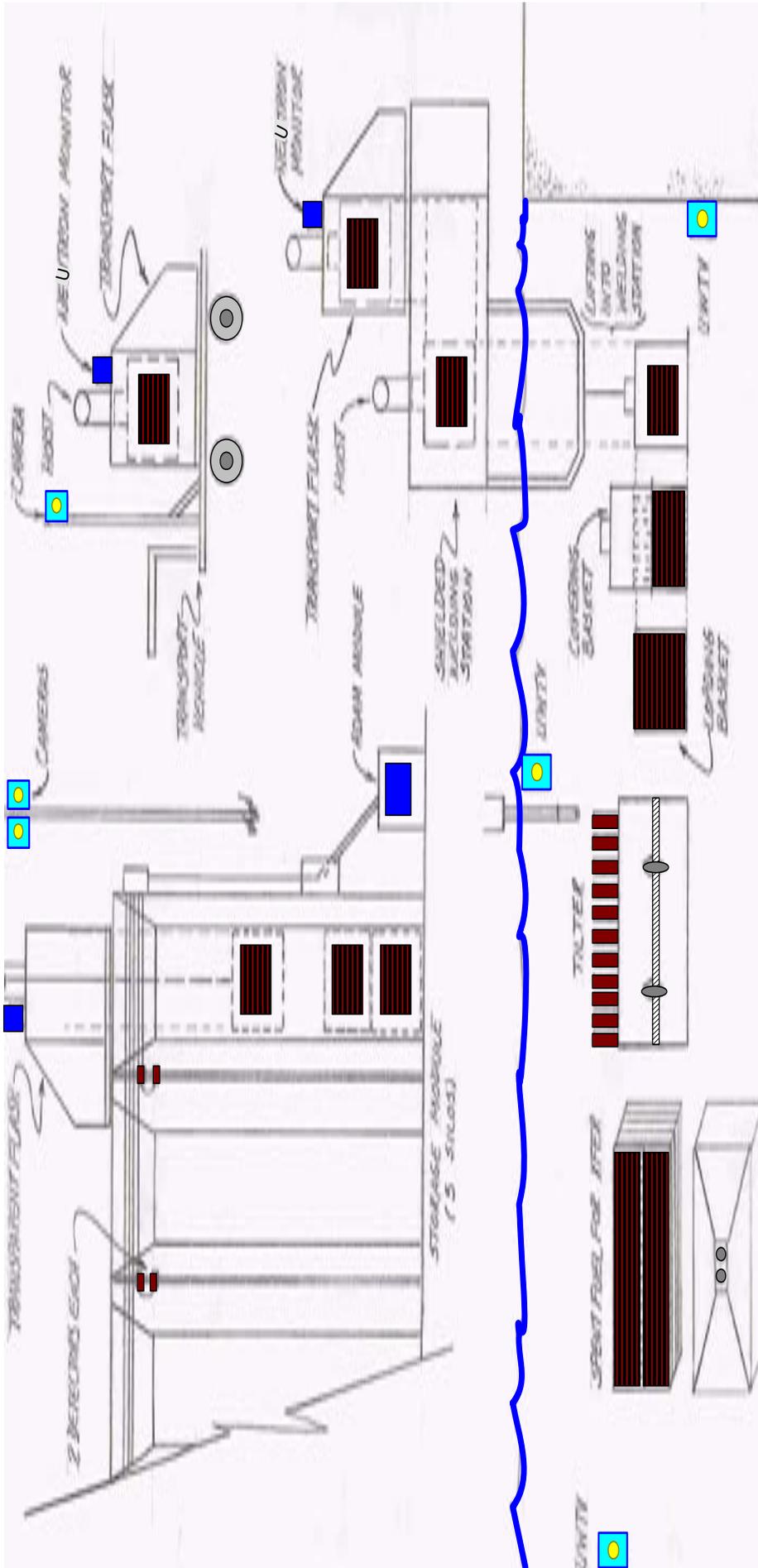


FIG. 1. Spent fuel transfer pathway for candu SF.

Strengthened safeguards system implementation at Serpong Nuclear Research Center – Batan

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Abstract. The International Atomic Energy Agency has conducted safeguards to nuclear material for more than 30 years. Indonesia has signed and conducted the safeguards agreements since 1980. Since the end of “Cold-War” some events have changed procurement and safeguards system requirements. The following, some procurements made member states and Secretary of IAEA desire to strengthening the safeguards system: Detection of Iraq’s nuclear weapon clandestine program in 1991, followed by difficulties to verify beginning report of Democratic People’s Republic of Korea after in forcing safeguards agreements in 1992, and decision of Government of South Africa to abandon their nuclear weapon program and merge on Non Proliferation Treaty (NPT) in 1991. In 1997, Boards of Governor of IAEA have approved Additional Protocol model to Safeguards agreement (published as document of INFCIRC/540 Corr.). Up to 19 July 2005 there are 102 states signing Additional Protocol and which have ratified there are 69 states. Government of Indonesia has signed and ratified Additional Protocol to Safeguards Agreement on 29 September 1999. Indonesia is the 8th state in the world that signed this protocol. By signing this protocol, it means Indonesia has commitment to implement the strengthened safeguards system. Strengthened safeguards system consists of 3 elements are related to:

- (a) Additional information
- (b) Broader access to location
- (c) Administrative measures^[1]

This paper explains conducted activities at Serpong Nuclear Research Center relate to implementation of strengthened safeguards system.

1. INTRODUCTION

Non Proliferation of Nuclear Weapon Treaty (NPT), and the comprehensive safeguards agreements regime on IAEA model INFCIRC/153 Corr., nuclear safeguards systems have been operated for over three decades. Indonesia ratified the NPT agreement by Act No. 8 Year 1979. The government of the Republic of Indonesia is committed to general contribution in achieving a condition of safe, secure and peace the world in relation of nuclear energy utilization and to continue its strong support for the principles of the treaty. In that time Indonesian nuclear program was not as big as present programs. By time changes, the utilization of nuclear energy for peaceful purposes was significantly increasing based on the world’s nuclear research and technology development. In the same time, Indonesia also enhanced of control capability by restructuring the Regulatory Body. Nowadays, Indonesia has three research reactors and other nuclear installations for research activities. The first nuclear power plant is planned will operating on year 2016.

Since the comprehensive safeguards agreements (CSA) INFIFCIRC/153 Corr. introduced, Indonesia finalized to have agreement with the IAEA on 1980. From that point many efforts have been done in order to realize the broader political decisions. The traditional safeguards with a good practices and familiarization and Indonesia took forward challenge to the comprehensive safeguards implementation. In intention to the main purpose to assure

correctness and completeness of information the attendance nuclear material from member states, the Agency establishing a report system related to CSA implementation. Indonesia carried out an informal series discussion with Agency and also the Agency inspectors during their stay in regular routine inspections. The information about nuclear material accountancy and design information has been covered under CSA. In conclusion that we did to proceed implementing the new regime called Additional Protocol to the Safeguards Agreements and entered onto force in the same time on September 1999. Based on CSA, Indonesia should inform to the Agency: facilities design, nuclear material inventory and their change, and any other special reports. Nuclear material inventory in each facilities cover all nuclear depleted natural and enriched form of uranium, thorium and plutonium isotopes. The inspection activities to strategic points by the Agency inspectors during the last few years draw into conclusion that there is no diversion of nuclear material from declared activities.

After entered into force the Additional Protocol (INFCIRC/540 Corr.), in the first stage implementation of Additional Protocol declaration had been done since year 2000 until mid of 2003. Under this protocol, the Agency in commitment with Indonesia is focusing more on expanded declaration, complementary access and broader environmental sampling. The coverage of information declaration is specific information concerned to nuclear related R&D, nuclear related equipment and non-nuclear material and other location of non-nuclear material. Based on additional protocol first stage, implementation trial during three years, the Agency concluded that there is no absence of undeclared nuclear material and activities such as clandestine nuclear program in Indonesia. By continuing the consistency and transparency nuclear activities to fulfill the requirements set forth in the agreements, the IAEA General Conference on September 2003, Indonesia has been declared as the third member state in the world to implement the Strengthened safeguards system after Australia and Hungary, and become the first in the ASEAN region. From this point, Indonesia will continue to maintain and to keep own mission on the track to contribute for broader sharing-view and promoting on safeguards implementation in the region^[2].

2. NUCLEAR FACILITIES AT SERPONG NUCLEAR RESEARCH CENTER - BATAN

Indonesia has seven nuclear installations at three different locations. Each facility is as one Material Balanced Area (MBA). The oldest MBA is for TRIGA MARK II reactor located in Bandung called MBA RI-A and TRIGA Mark II reactor located in Yogyakarta called MBA RI-B. The other biggest nuclear complex is located at Serpong, southwest Jakarta with 4 different nuclear facilities such as fuel fabrication facility, experimental fuel element facility, metallurgical laboratory, and Interim storage for spent fuel facility. The MBA's names and location are listed below. Table 1 shows the Indonesian nuclear facilities under IAEA's safeguards:

Table 1. Indonesia's nuclear facilities.

No.	Name of Facility	MBA	Location
1.	The 2000 kW TRIGA Mark II Research Reactor	RI-A	Bandung
2.	The 250 kW TRIGA Mark II Research Reactor	RI-B	Yogyakarta
3.	The 30 MW MTR reactor GA Siwabessy reactor	RI-C	Serpong
4.	The LEU Fuel Fabrication Installation	RI-D	Serpong
5.	The Experimental Fuel Element Installation	RI-E	Serpong
6.	The Radiometalurgy Installation	RI-F	Serpong
7.	The Interim Storage for Spent Fuel	RI-G	Serpong

Most of MBA above has only a nuclear material category III for physical protection purposes. Those locations will be covered to the information declaration for additional protocol to the safeguards agreements concerned.

3. COMPREHENSIVE SAFEGUARDS

Based on the signing of Safeguards Agreement referred to INFCIRC/153 Corr. document^[3], Indonesia has obligation to implement safeguards system called as SSAC (State Systems of Accounting for and Control of Nuclear Materials) to all nuclear material under safeguards based on definition in the document. Therefore, the State Regulatory (BAPETEN) has released conducting regulation for State Level was Decree of Chairman of BAPETEN No. 2/2005 about SSAC (State Systems of Accounting for and Control of Nuclear Materials).

Systems are based on three measured elements: Nuclear Material Accounting for, Containment and Surveillance System with characteristic of conducting safeguards are:

- (1). Action of safeguards implementation based on facility level with structure of MBA (Material Balance Area).
- (2). IAEA's routine inspection is limited only at strategic point in declared nuclear facility.
- (3). Main objective of safeguards is early detection of important nuclear material from peaceful purpose to nuclear weapon or other unknown explosive materials.
- (4). IAEA verifies nuclear materials data and declared facility (did not look for undeclared information)

Based on description above, it can be known rights and obligations of BATAN (National Nuclear Energy Agency) as a Research and Development Institution in nuclear technology field. On implementing safeguards based on safeguards agreement (as INFCIRC/153 document), it can be mentioned that there are no problem with those. BAPETEN has completed all regulations. Systems made all job units in BATAN an obligation to manage nuclear facilities based on structure of MBA definitively and to give ease for IAEA inspectors.

Conclusion of safeguards implementation result is based on that there is no deviation of declared nuclear materials reported by state. It is compared by verification result that conducted by IAEA's inspectors.

4. STRENGTHENED SAFEGUARDS SYSTEM IMPLEMENTATION AT SERPONG NUCLEAR RESEARCH CENTER – BATAN.

Government of Indonesia has signed and ratified Additional Protocol to Safeguards Agreement on 29 September 1999 (as INFCIRC/283 add. 1 document). NPT safeguards scope covers all nuclear materials in all peaceful purposes nuclear activities have extended its scope by Additional Protocol to Safeguards Agreement (as INFCIRC/540 Corr. document). The document requires declaration completely included all activities nuclear related in past, present, and the future. Peaceful purpose or no, and the most important thing is to require Member State permitting IAEA inspectors to access information and location on facility and location other facility that have declared. Briefly, the safeguards action is addressed especially to ensure absence of undeclared nuclear activities. Information that reported to the IAEA based on INFCIRC/540 Corr. included all R&D activities nuclear related information. It has ever conducted discussion between IAEA's inspectors and researcher at Serpong Nuclear Research Center about some papers related R&D activities nuclear related. Some information

is got from open source and analysis result of sample withdrawal. Member State conducts all R&D activities nuclear related information or all activities are indicated by:

4.1. Specific equipment nuclear related

One of specific equipment nuclear related is plant for the fabrication of fuel elements. At Serpong Nuclear Research Center, there is LEU Fuel Fabrication Installation (MBA RI-D).

4.2. Supporting infrastructure

IAEA recommended setting the fences around the Nuclear Research Center in Serpong consist of:

- (a) Reactor fence
- (b) Yellow fence
- (c) Batan fence
- (d) Puspipstek fence

4.3. Tell-tale traces environment

IAEA has ever performed swipe sample/Pap smear test at the Radioisotope Hot-cell Laboratory, at The Radiometalurgy Installation, and at the High Level Waste Storage in Serpong Nuclear Research Center area.

4.4. Nuclear material utilization prediction.

In conducting of verifying of nuclear material, if needed, IAEA's inspector conducted nuclear material measurement. Some technique is used to conduct Non Destructive Assay (NDA) measurement. The main wares are used, such as MCA equipment, ICVD, HM-5 and the others are based on gamma ray and some nuclear material emits neutron.

Approachable things by preparing base information covers:

- (a) An expanded declaration by Member State
- (b) Information evaluation by IAEA
- (c) New technical measures
- (d) Enhanced inspector access

These things enable for the IAEA to evaluate consistency of declaration conducted by Member State by checking with all IAEA's other information from:

- (a) Open sources
- (b) Inspection data
- (c) Withdrawal result of Environment sample and
- (d) Complementary access

to make available data on verification or detection undeclared nuclear activities.

Special arrangement for Short Notice Inspection (SNI) for Multipurpose Reactor (MBA RI-C) at Serpong Nuclear Research Center had been conducted since the Integrated Safeguards System was caused to be effective. Since MBA RI-C classified into group III that has 1 SQ inventory required at least one interim inspection simultaneously with short notice inspection. For others MBA required one random inspection. During inspection time, the Agency inspector may add another access to specified location by short notice at least 2 hours before start and access. Both fax and direct phone contact should deliver those notices to certain official. This position will also have impact to the BAPETEN routine inspection to each

facility. And BAPETEN recommends the operators that always keep on update their record and report all material and activities.

CONCLUSION

Based on description above, it can be concluded that there are no problems on implementing strengthened safeguards system at Serpong Nuclear Research Center. All have gone better.

SUGGESTION

There are several things can be conducted to enhance strengthened safeguards implementation at Serpong Nuclear Research Center as mentioned below:

- (1).BAPETEN shall publish as soon as possible conducting regulation about Strengthened Safeguards System and socializing it to all job units in BATAN that have obligation to arrange appropriate declaration. So, the requirements can be fulfilled immediately and it will be helpful for BAPETEN to give access to IAEA's inspectors.
- (2).For internal BATAN behalf, It is preferable that there is one job unit in BATAN to monitoring implementation of declaration so that no misperception on implementing declaration. Especially, concerning the future nuclear program policy.

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To achieve a nuclear material inventory in case of emergency

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The poster details the aim and context of such a type of exercise. The main component elements are described and the experience feedback based on 10 exercises involving 14 nuclear plants is addressed. The different items presented in the poster are as follows :

AIM AND CONTEXT

- To test and check the organisation of crisis in the event of suspicion of theft or diversion
- To prepare the various actors involved for complicated and realistic interventions
- To test and check the organisation of crisis in the event of suspicion of theft or diversion
- To complete procedure assessment and inspections
- To maintain a level of vigilance against malevolent or terrorist threats on nuclear facilities

COMPONENT ELEMENTS

SCENARIO

The different possible themes

- Theft or sabotage
- One or several operators
- One or several sites
- +/- media pressure

Examples

- Nuclear item shifting
- Picture trafficking + blackmail
- Database falsification
- Use of traps

Basis of elaboration

- Security studies
- Design basis threats
- Identification of nuclear material flows
- Identification of potentially threatening nuclear materials

CRISIS ORGANIZATION

Emergency committee on National Level

- Authority : Ministry in charge of industry
- National direction operator(s)
- Technical support IRSN

Emergency committee on local Level

- Local Direction operator(s)
- Technical teams for physical protection system integrity, Inventory, Accountancy data, and measurement checking

Media pressure on local and national level

RELIANCE ON REGULATIONS

A regulation on the follow-up and the accountancy of nuclear material with, at least, a text dealing with the need to carry out inventories.

In France-Decree n°81-512 May 12 1981:

- To know all the input-output, localization, use, transformation of nuclear material
- All these provisions under standard "Management of quality"

SCHEDULE

D-7 months

- Presentation meeting at the operators'

D-4 months

- Meeting of official launching between: Authority, Operators and Technical support (IRSN)

From D-3 months to D-2 weeks

- meeting for preparing scenarios:
- Media pressure preparation
- Working out an exercise convention

D

- Launching of the exercise and immediate assessment

D+1 month

- Debriefing for evaluation and identification of the improvements

EXPERIENCE FEEDBACK

Feedback to achieve that type of exercise

SCENARIO

- The essential involvement of the operator in the development of the scenario
- Non-proliferation is no longer the only assumption, sabotage could be included

SCHEDULE

- The preparation must imperatively consist of:
 - Pre-briefing of the operators several months in advance
 - The issuing of an exercise agreement settlement
 - Before the exercise, the delivery of relevant documentation
- A first assessment must be done immediately after the end of the exercise in each emergency committee
- An assessment of the comprehensive feedback must be carried out approximately one month after the exercise with representatives of all the involved actors

Feedback on the improvements to be brought in case of a real crisis

- The need for defining an indicator allowing us to communicate on the inventory progress
- To have a dedicated Internet site to provide journalists with information in order to avoid network overload
- The deadlines for the validation of the press releases must be compatible with the requirements of media timing
- The need to set up dedicated and protected networks to guarantee the confidentiality of information

BENEFITS GAINED

- Confidence of the authorities in the control of crisis situations

- Validation of the written procedures for crisis situations
- Realistic evaluation of the duration in order to achieve a partial inventory
- Skills acquired by the teams working with the media pressure
- Confidence in the reliability of technical communication equipment
- Verification of the nuclear materials data

A Field Exercise Course to Train IAEA Safeguards Inspectors in Implementing the Additional Protocol and Performing Complementary Access Activities

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Abstract

The International Atomic Energy Agency (IAEA) Department of Safeguards has the task of implementing the Additional Protocol (AP) in the Member States that have signed agreements bringing that treaty into force. The IAEA inspector under the traditional INFCIRC/153 [1] safeguards has been an accountant focused on the declared nuclear material stores of a Member State. The INFCIRC/540 [2] Strengthened Safeguards System (SSS) provides the Agency and its inspectors with the right to investigate a Member State's nuclear programme to see if all declared activities are in order and no undeclared activities exist. This broadening of the scope of the inspector's responsibilities has changed the training of the inspectors to orient them to being an investigator compared to an accountant [3]. The Safeguards training department has created a curriculum of courses that provides the background to train the inspectorate into this new inspection regime [4]. The United States Support Program (USSP) has contributed to this curriculum by putting together a course at Brookhaven National Laboratory (BNL) in Additional Protocol Complementary Access (APCA) to give the IAEA the opportunity to provide inspectors a necessary field exercise in a realistic environment at a research site.

Brookhaven National Laboratory contains three shutdown nuclear research reactors, operating particle accelerators, hot cells, radioactive waste storage, laser laboratories, and magnet production facilities on a large site very similar to numerous research facilities around the world situated in non-nuclear weapon states (NNWS). The USSP Team created an Article 2 declaration containing annotated maps of the site, descriptions of the buildings on site, satellite and aerial photographs of the area, and a declaration of research activities on the site. The declaration is as realistic to actual BNL research except that proprietary and security concerns of the BNL site have been taken into account. The USSP Team felt the best training vehicle provides a realistic scenario and that adding manufactured fictional situations weakens the scenario for the student. The facilities at BNL provide enough ambiguities for the student to investigate without any contrived scenarios.

The BNL site will be part of a fictional country called Freedonia. Freedonia consists of the real Long Island of the U.S. state of New York consisting of the actual Nassau, Suffolk, Kings (Brooklyn), and Queens Counties which for our purposes we assume to be an independent NNWS. The real Long Island contains numerous interesting open source material for the IAEA students to investigate. One can show that "Freedonia" had a flourishing aerospace industry that could provide technical expertise

B. Boyer et al.

for a gas centrifuge uranium enrichment programme. There is also the decommissioned and decontaminated Shoreham Nuclear Power Station about 10 km north of BNL. On the outskirts of BNL there also exist various industrial structures and businesses that the students may investigate as being functionally related to BNL.

The BNL course will focus mainly on the techniques of complementary access activities. In implementing the AP, the inspector will examine the open source research provided to him, the country report about the State's past nuclear activities and possible future nuclear ventures including any areas of proliferation concern, the State's AP declaration to the IAEA, and satellite photographs of the site. The inspector must then create a picture of the State's capabilities as declared and a programme to investigate ambiguities in the State's nuclear ambitions including clarifications of the State's declaration. IAEA Department of Safeguards Training Staff will train the inspectors in the AP declaration evaluation, complementary access equipment use, and rationale for site selection for complementary access in Vienna during a week-long preparatory part of the course to prepare them for the next week's field exercise in "Freedonia" at BNL. When the inspectors arrive at BNL, the BNL training team will act as the State System of Accounting and Controls of nuclear material (SSAC) personnel and actual BNL staff will act as the operator.

The USSP Team will be forcing the IAEA inspectors to use expertly the tools of complementary access in a thoughtful manner. They will stress that there exist within the bounds of INFCIRC/153 safeguards means to do design verification and other activities where they would not need to exercise the rights of complementary access. The USSP Team will also stress how to handle the State's and operator's right of managed access of the site in a manner so that the inspector can answer the questions he has about a site without trampling the State's rights or being deceived by a State intent on hiding undeclared activities.

While the course will encourage the inspectors to investigate all aspects of the BNL site, BNL will have the student inspectors focus on certain unique facilities on the site so that a fruitful and controlled exercise can be conducted. The inspectors will access the waste management areas with its hot cells, the laser test facilities, and magnet production areas, the shutdown Brookhaven Medical Research Reactor and off-site at the decommissioned Shoreham Nuclear Power Station. These areas will provide the inspectors with possible areas of nuclear activity that will need the exercise of AP tools such as complementary access. The inspectors should have an appreciation of how to do AP declaration analysis, choose complementary access locations, handle managed access, and understand the limits of the rights and responsibilities of an inspector under the AP inspection regime.

1. Introduction

The IAEA Department of Safeguards has the task of implementing the AP in the Member States that have signed agreements bringing that treaty into force. The IAEA inspector under the traditional INFCIRC/153 safeguards has been an accountant focused on the declared nuclear material stores of a Member State. The INFCIRC/540 Strengthened Safeguards System provides the Agency and its inspectors with the right to investigate a Member State's nuclear programme to see if all declared activities are in order and no undeclared activities exist. This broadening of the scope of the inspector's responsibilities has changed the training of the inspectors to orient them to being an investigator compared to an accountant [1]. The Safeguards training department has created a curriculum of courses that provides the background to train the inspectorate into this new inspection regime [2]. The United States Support Program has contributed to this curriculum by putting together a course at Brookhaven National Laboratory in APCA to give the IAEA the opportunity to provide inspectors a necessary field exercise in a realistic environment at a research site. A USSP Team including BNL staff and Science Applications International Corporation (SAIC) experts in managed access techniques worked to create the desired AP declaration for the BNL site and realistic proliferation scenarios for the student inspectors to investigate at the BNL site.

2. Background

The BNL site is similar to many large sites all over the world that contain nuclear and scientific endeavours of varying size and scope such as Ispra in Italy, Karlsruhe in Germany, and Swierk in Poland. These three sites as well as many others of a similar nature will be declared to the IAEA under the Additional Protocol. BNL contains three shutdown nuclear research reactors, operating particle accelerators, hot cells, radioactive waste storage, laser laboratories, and magnet production facilities on a large site very similar to these research facilities around the world situated in non-nuclear weapon states (NNWS). The IAEA inspectors, who examine the declarations, determine where to conduct complementary access (CA), and analyze the results of such CA activities to determine that all declared nuclear activities are reported properly and that no undeclared activities are occurring, need guidance and training in executing this task. The USSP Team determined that an exercise at BNL, a course in Additional Protocol Complementary Access (APCA), would be an excellent tool for training the inspectors in CA.

The USSP Team working in concert with the IAEA Department of Safeguards Training Staff created an Article 2 declaration containing annotated maps of the site, descriptions of the buildings on site, satellite and aerial photographs of the area, and a declaration of research activities on the site. The declaration is as realistic to actual BNL research except that proprietary and security concerns of the BNL site have been taken into account. The USSP Team felt the best training vehicle provides a realistic scenario and that adding manufactured fictional situations weakens the scenario for the student. The facilities at BNL provide enough ambiguities for the student to investigate without any contrived scenarios.

The BNL site will be part of a fictional country called Freedonia, shown in Figure 1. Freedonia consists of the real Long Island of the U.S. state of New York consisting of the actual Nassau, Suffolk, Kings (Brooklyn), and Queens Counties which for our purposes we assume to be an independent NNWS. The real Long Island contains numerous interesting open source material for the IAEA students to investigate. One can show that “Freedonia” had a flourishing aerospace industry that could provide technical expertise for a gas centrifuge uranium enrichment programme. There is also the decommissioned and decontaminated Shoreham Nuclear Power Station about 10 km north of BNL. On the outskirts of BNL there also exist various industrial structures and businesses that the students may investigate as being functionally related to BNL. A Nuclear Weapons State (NWS), Sylvania, which is the actual United States of America (USA) minus Long Island, borders Freedonia. Maplonia and Medvedia, which are, respectively, NNWS and NWS, and are the actual Canada and Russia, respectively, are near neighbours to Freedonia. Freedonia is a signatory to the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) and has a Comprehensive Safeguards Agreement (CSA) in place and the AP is in force with the implementation of AP occurring in 2006 Freedonia to coincide with the course activities.



FIG. 1. The Fictitious State for Freedonia constructed for APCA Course at BNL

B. Boyer et al.

The BNL course will focus mainly on the techniques of complementary access activities. In implementing the AP, the inspector will examine the open source research provided to him, the country report about the State's past nuclear activities and possible future nuclear ventures including any areas of proliferation concern, the State's AP declaration to the IAEA, and satellite photographs of the site. The inspector must then create a picture of the State's capabilities as declared and a programme to investigate ambiguities in the State's nuclear ambitions including clarifications of the State's declaration. IAEA Department of Safeguards Training Staff will train the inspectors in the AP declaration evaluation, complementary access equipment use, and rationale for site selection for complementary access in Vienna during a week-long preparatory part of the course to prepare them for the next week's field exercise in "Freedonia" at BNL. When the inspectors arrive at BNL, the USSP Team will act as the State System of Accounting and Controls of nuclear material (SSAC) personnel and actual BNL staff will act as the operator.

3. Course Goals

The USSP Team will be forcing the IAEA inspectors to use expertly the tools of complementary access in a thoughtful manner. They will stress that there exist within the bounds of INFCIRC/153 safeguards means to do design verification and other activities where they would not need to exercise the rights of complementary access. The USSP Team will also stress how to handle the State's and operator's right of managed access of the site in a manner so that the inspector can answer the questions he has about a site without trampling the State's rights or being deceived by a State intent on hiding undeclared activities. Specifically, the USSP Team defined three major course goals for the APCA with emphasis on various subtopics defined below:

1. Understanding of INFCIRC/540(corr.) and the Strengthened Safeguards System (SSS)
 - Exercising inspector's authority under Additional Protocol (AP)
 - Complementary Access (CA) issues
2. Analysis of State's nuclear program using:
 - State Evaluation Report (SER)
 - Design information
 - Inspection reports
 - Open sources
 - Exports/manufacturing aspects of AP
 - State Article 2 declarations analysis
3. Carry out mock complementary access at a site
 - Create rationale for CA
 - Reactor decommissioning review and discussion with respect to AP and CA issues
 - Managed Access from the State's position with conflict resolution

4. Freedonia Country Profile

The state of Freedonia has the following nuclear facilities and capabilities. It has a nuclear power reactor, Shoreham Nuclear Power Station (SNPS), shut down by concern over evacuation in emergency situations and environmental concerns. It still has a flagship nuclear research centre in

BNL which has research reactors and diverse high tech research. It has a fuel cycle history that includes no uranium resources, mining or mills, no conversion or fuel fabrication, and no enrichment or reprocessing. However, Freedonia does have a high tech industrial infrastructure with a legacy of aerospace industrial activity, centrifuge related technologies, and a large number of aerospace support engineering firms. It has a cutting edge electronics industry including lasers, magnets, and precision manufacturing that would be of value to potential proliferators. Freedonia also has biotech and pharmaceutical work to round out the Freedonia science and technology portfolio. Figure 2 shows the location of BNL and SNPS.

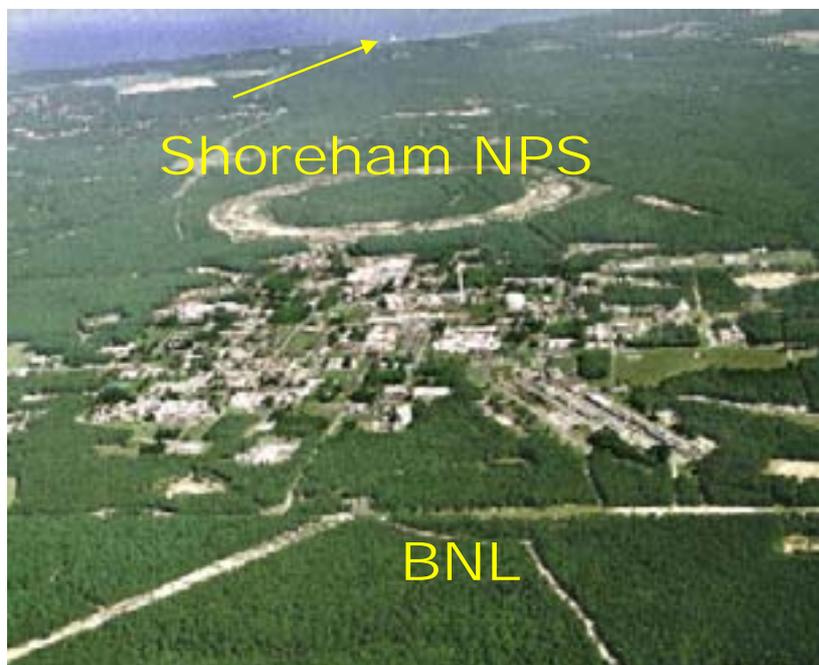


FIG. 2. Brookhaven National Laboratory and Shoreham Nuclear Power Station of Freedonia

While the course will encourage the inspectors to investigate all aspects of the BNL site, BNL will have the student inspectors focus on certain unique facilities on the site so that a fruitful and controlled exercise can be conducted. The inspectors will access the waste management areas with its hot cells, the laser test facilities, and magnet production areas, the shutdown Brookhaven Medical Research Reactor (BMRR) and off-site at the decommissioned Shoreham Nuclear Power Station. These areas will provide the inspectors with possible areas of nuclear activity that will need the exercise of AP tools such as complementary access. The inspectors should have an appreciation of how to do AP declaration analysis, choose complementary access locations, handle managed access, and understand the limits of the rights and responsibilities of an inspector under the AP inspection regime. The IAEA conducted this pilot course with the assistance of the USSP Team on 12-16 June 2006 at IAEA Headquarters in Vienna and 19-23 June 2006 at BNL (Figure 3).



FIG. 3. USSP Team, IAEA Instructors, and Student Inspectors – June 2006

5. Freedomia APCA Exercise Scenarios

For the APCA exercise, the USSP Team working with the IAEA Safeguards Department Section for Safeguards Training decided on choosing three different exercise scenarios. There would be four groups of student inspectors with each group containing three members. Therefore, each member of the group had the opportunity to lead one of the three exercises. The USSP Team arranged the exercises in time and space so that each group would work separate from each other. There was also an example CA performed at the BMRR and the SNPS on the first day of the course to stimulate discussion and warm up the students to doing the three group exercises on the next two days. On the fourth day they would compile their results and write a report to be presented to the entire class and instructors on the fifth and final day. On the final day they would receive feedback on their efforts and see how well they understood the scenarios, if they perceived the possible undeclared nuclear activities that could stem from the research and facilities at BNL, and whether their recommended actions by the IAEA were appropriate.

The example CA on Monday included tours of SNPS and BMRR to raise questions on the proper vehicle to ascertain the decommissioned status of a facility (Figure 4). During the tour of these two sites, the inspectors tried to determine if BMRR and SNPS could be reconstituted and if they could be reconstituted how quickly and how easily. The two facilities were vastly different. The small BMRR is closed down in the beginning stages of decommissioning. BNL removed the fuel and control rods and is planning for decontamination efforts. On the other hand, the power reactor at SNPS is totally decommissioned with its U.S. Nuclear Regulatory Commission (NRC) license as a power reactor removed and the site fit for other activities. SNPS has no fuel and is totally decontaminated with all usable parts of the nuclear steam supply system dismantled and sold off the site or abandoned in place. During the tour and in discussions at BNL after returning to the BNL site, the inspectors, IAEA instructors, and USSP Team discussed the decommissioned status of the two facilities and what inspection tools, CA or Design Information Verification (DIV) would be useful or appropriate for each facility. These discussions helped to put the students in the proper mindset for the week's activities.



FIG. 4. Example CA at SNPS and BMRR

The first group CA scenario was undeclared laser enrichment (Figure 5). The inspectors, who generally have no background in lasers, received lectures on laser enrichment and possible signs of an undeclared laser enrichment program in Vienna the week prior to coming to BNL. BNL offered two facilities for the inspectors to do CA activities in: the Free Electron Laser (FEL) and the Accelerator Test Facility (ATF). The goals for the inspectors at FEL and ATF were to determine if these facilities could be used for laser isotope separation and if they had signs of research in laser isotope separation. The USSP Team decided that the BNL FEL and ATF personnel and the USSP Team personnel acting as the Freedomia State Systems of Accounting for and Control of nuclear material (SSAC) should be cooperative but refuse access to certain areas in ATF. The inspectors would have to be innovative and diplomatic to gain access desired within the limited time available.



FIG. 5. Laser Enrichment Scenario CA at ATF and FEL

The second group CA scenario was a proliferation indicators search (Figure 6). BNL offered the Waste Management Facility (WMF) for the CA activities. The WMF has a small hot cell and a concentration of waste handling capabilities. The inspectors attempted to determine if the WMF had undeclared nuclear materials and undeclared reprocessing activities. The USSP Team decided that the BNL WMF personnel and the USSP Team personnel acting as the Freedonia SSAC should be fairly uncooperative, force the inspectors to wait and deal with bureaucracy, and try to manage inspectors' paths away from sensitive components/systems. The inspectors would have to be diplomatic, use time wisely, and focus on objectives. The time available to the inspectors was limited so the SSAC and operator strategy tried to force the inspectors to waste time and be distracted from seeing activities at the WMF that could be a possible indication of an undeclared activity.



FIG. 6. Proliferation Indicators Scenario CA at WMF

The third group CA scenario was undeclared enrichment focusing on Electromagnetic Isotope Separation (EMIS) and accelerator proliferation concerns (Figure 7). Since the exposition of the clandestine Iraqi EMIS program, the IAEA has been concerned that a proliferator would use EMIS, which the United States used in the Manhattan Project and later abandoned as inefficient, because EMIS would not be a mainstream technique today and, therefore, would not arouse much suspicion from the IAEA. The IAEA wanted its inspectors to be ever vigilant about all possible proliferation pathways. The IAEA has voiced concerns about the use of accelerators as a clandestine mechanism for breeding plutonium [5]. BNL offered the Superconducting Magnet Division's magnet production facility housed in the old Cosmotron accelerator building and its surrounding support structures for CA activities for this scenario. The inspectors searched for signs of magnet production for an EMIS racetrack in the magnet facility. They also searched for signs of producing accelerator magnets for transmutation that could be indicative of undeclared plutonium production. They also wanted to see if import/exports were not reported of sensitive items on the trigger list. The USSP Team decided that the BNL WMF personnel and the USSP Team personnel acting as the Freedonia SSAC should be extremely cooperative but somewhat deceptive with Dr. Murature of the magnet facility lecturing the inspectors expansively and enthusiastically on magnets so as to break the inspectors' concentration from the task at hand. The inspectors had to focus on the mission, not be distracted, and use time wisely. This facility had the most emphasis on the AP's use of managed access by the State and operator to protect information. The USSP Team included John Gilbert who has been involved with

managed access issues in various other treaty regimes such as Intermediate-Range Nuclear Forces (INF) and Chemical Weapons Convention (CWC) and has been a consultant to the IAEA on AP managed access issues [6]. Gilbert gave the inspectors an extremely useful lecture to open the week at BNL to open their eyes to the ways that they could have access control used on them by managed access allowed under the AP plus other means not explicit in the AP that may not be in the spirit of the AP. For example, Gilbert devised ways to control access to keep the inspectors from peering into a white tent on the manufacturing floor of the magnet facility.



FIG. 7. Undeclared Enrichment with EMIS and Accelerators CA at Magnet Facility

6. Course Benefits to the IAEA

The APCA course gives the IAEA the following important benefits. It gives a comprehensive exercise in AP Implementation including declaration and site review, elements of country report and open source review and planning of CA. There is an opportunity to practice CA in a realistic environment at a challenging site such as BNL. The inspectors receive training in managed access and controlled access expertise from experts including classroom lecture on managed access/controlled access and most importantly exercises and the possibility of critique from the USSP Team. The inspectors also get perspective on the difficult aspects of AP implementation such as AP implementation effort and scope, the crucial aspect of CA site selection and the use of CA when other techniques are not appropriate, and the limits of the AP, CA, and the SSS regime. Hence, the goal of this training is to provide the IAEA with inspectors who can make the jump from being the traditional material accountant to being also an investigator able to implement the AP and move IAEA safeguards into the 21st century.

ACKNOWLEDGEMENTS

The authors would like to acknowledge the assistance of BNL staff at the ATF, FEL, WMF, and from the Superconducting Magnet Division as well as our colleagues at the IAEA who worked long and hard with us to insure a successful course.

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The road to safeguards quality: An E-learning tutorial

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Presented by K. Desson and D. Martin

Abstract. To support the implementation of a department-wide Quality Management System (QMS), the IAEA Section for Standardization, Division of Concepts and Planning, Department of Safeguards commissioned the Canadian Safeguards Support Program (CSSP) to develop an interactive computer-based training course (ICBT) on Quality Management to introduce new and existing staff to basic QMS principles. Since January, 2006, a CSSP project manager, instructional designer, and programmer/graphic designer have been working with subject matter experts in the Department of Safeguards Division of Concepts and Planning to research and develop the course. Testing and implementation are planned for later in the year. The course itself, and the step-by-step process used to develop it, are intended to serve as models for a curriculum of QMS and other courses to be delivered to IAEA staff at their desktops by means of the Agency's LAN.

1. Introduction

The IAEA Department of Safeguards is implementing a Quality Management System (QMS) as a means of ensuring soundly-based safeguards conclusions. Guided by a Quality Policy Statement approved by senior management in November, 2004, the Departmental Quality Manager in the Division of Concepts and Planning, Section for Standardization, has since begun to implement a wide range of measures designed to create a QMS based on the requirements of the ISO 9001: 2000 standard.

An early priority has been the development of a training program to introduce staff to the QMS and to teach specific groups how to apply quality management principles to their jobs. Since 2000 training has been provided to Safeguards staff on Quality Management, with the current training curriculum consisting of a one-hour introductory seminar for new inspectors, a 1.5 hour briefing for new staff, a two-day seminar for safeguards managers, and a five-day workshop teaching tools and methodology for staff intending to work on continual process improvement of Agency work processes. In addition, it was determined that an interactive computer-based training course (ICBT) on Quality Management is required to introduce all new and existing staff to basic QMS principles. Offering computer-based training at all desktops is desirable to introduce new staff to QMS concepts as early in their employment as possible and to provide easy refresher training for any staff members who wish to take the course at their desk at their convenience. ICBT complements classroom training as people have different learning styles and preferences and this format may be easier for some to learn from than from conventional classroom training. In addition, ICBT can be used to provide updated information to staff on developments in a more timely manner.

To carry out this work, the Canadian Safeguards Support Program (CSSP) was approached for assistance. Previously, the CSSP had developed several successful computer-based training programs for the Department of Safeguards on topics such as *CANDU Power Station Fundamentals*, *VXI Integrated Fuel Monitor (VIFM) Procedures*, *Interpreting Bundle Counter Radiation Signatures*, and *Core Discharge Monitor Radiation Signatures*. In January, 2006, a CSSP project team consisting of a project manager, instructional designer, and programmer/graphic designer was created to work with subject matter experts in the Department of Safeguards Division of Concepts and Planning.

2. Objectives and Requirements

The main objective for a computer-based Quality Management training course was to explain to new staff how the QMS works and what impacts it will have on how they carry out their duties. As the Department's Task Proposal (SP-1) put it: "Communication is a key element of a QMS and it is essential that staff be aware of the development, implementation and functioning of the Departmental QMS and their role in the functioning of the QMS."

More specifically, after completing the course, the Department's objectives for the course are that participants should be able to:

- Describe what is meant by quality and a quality management system
- Identify the key stakeholders who are interested in the quality of the Department's work
- Describe the implementation of the Department's QMS
- Describe how the QMS impacts on their work
- Know where to find further assistance and information

In addition, the Department gave direction to the CSSP production team regarding required characteristics of the interactive training program. It must:

- Require no more than 2 hours of each learner's time to complete
- Permit the on-line assessment of the participant's understanding of the learning materials
- Incorporate a variety of media and avoid a "page-turner" style of presentation
- Allow some components of the program to be updated by the Department without further intervention from the contractors, such as text screens describing the ever-changing status of QMS implementation within the Department
- Be accessible from IAEA desktop computers via the Agency's local area network (LAN)
- Lead participants through the learning materials systematically, ensuring that they master one set of concepts before moving on to the next set
- Generate a record of the names of staff members who have taken and passed the course, and make this information available in an organized format to course administrators

The Department and production team agreed to pursue the development of a highly interactive, media-rich, centrally administered training program delivered over the IAEA Intranet, as a convenient form of presentation to Safeguards staff.

3. Content Capture

As the Department of Safeguards was already in its second year of QMS implementation at the time the ICBT was commissioned, a number of information resources were readily available to the development team. These included a comprehensive QMS Web site developed by the Department as a reference resource for staff, instructional materials developed for classroom-based courses, a number of informational PowerPoint presentations, conference papers, and a library of books and articles on subjects related to quality management. These documentary sources provided a solid basis from which an initial content outline could be prepared.

The developers were also given access to Department of Safeguards subject matter experts (SMEs), who made themselves available for extended face-to-face and telephonic interviews. The interviews, from which verbatim transcripts were prepared, proved to be particularly helpful in understanding the operation of the QMS within the Department's unique operating environment.

An important feature of the training modules is the video commentary by a well-placed executive that introduces each module. While each commentary is scripted by the Ottawa-based production team in consultation with the individuals who would appear on screen, arrangements were made with the Department of Safeguards training department to use their video equipment and crews to capture the commentaries. This division of labor is believed to be effective and cost efficient.

4. Instructional Design

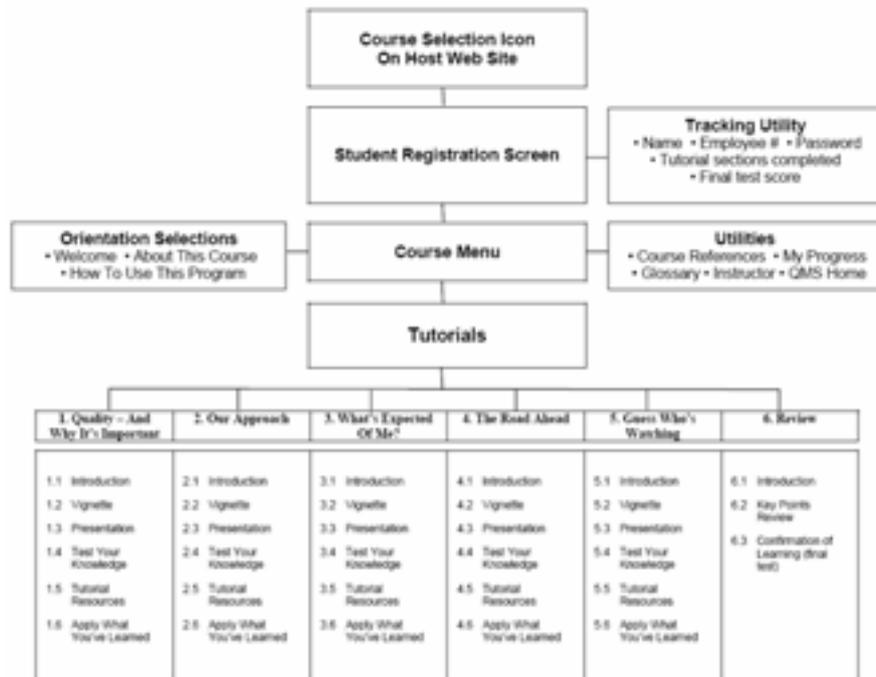
To capture interest and promote learning, the course that has emerged – entitled *The Road to Safeguards Quality: An E-Learning Tutorial* – uses a wide variety of presentation techniques. These include:

- Short video clips featuring concise testimonials and scene-setting commentaries by experienced QMS practitioners throughout the program
- An entertaining “story” module designed to provide a light introduction to key QMS concepts
- Animated vignettes at the beginning of each module to capture key messages in a memorable way
- Frequent quizzes and exercises to test understanding and broaden perspectives
- Continuous online access to reference documents and Web resources
- Ready access to departmental trainers and subject matter experts by means of an e-mail utility
- Tracking of student progress to help participants resume their studies from session to session
- A modular technical design that enables the Department to make course updates easily, as required
- A pass-fail test to conclude the course

Based on e-learning best practices, the tutorials are characterized by:

- Short tutorial duration (15-20 minutes required for the typical student to cover all of the materials in a tutorial)
- Presentations within tutorials limited to a maximum of 12-15 screens, each featuring a main teaching point
- Learning objectives identified for each tutorial in the tutorial Introduction (default) screen
- Key points for each tutorial highlighted in a memorable introductory vignette
- Graphic illustration of teaching points wherever possible
- Frequent use of examples and demonstrations
- Frequent requirement for learners to interact with the program (e.g. by answering a question, launching a demonstration, assembling a diagram, etc.)
- Immediate feedback to questions, quizzes and exercises

The following diagram captures the organization of the ICBT as it finally emerged.



5. Interface Design

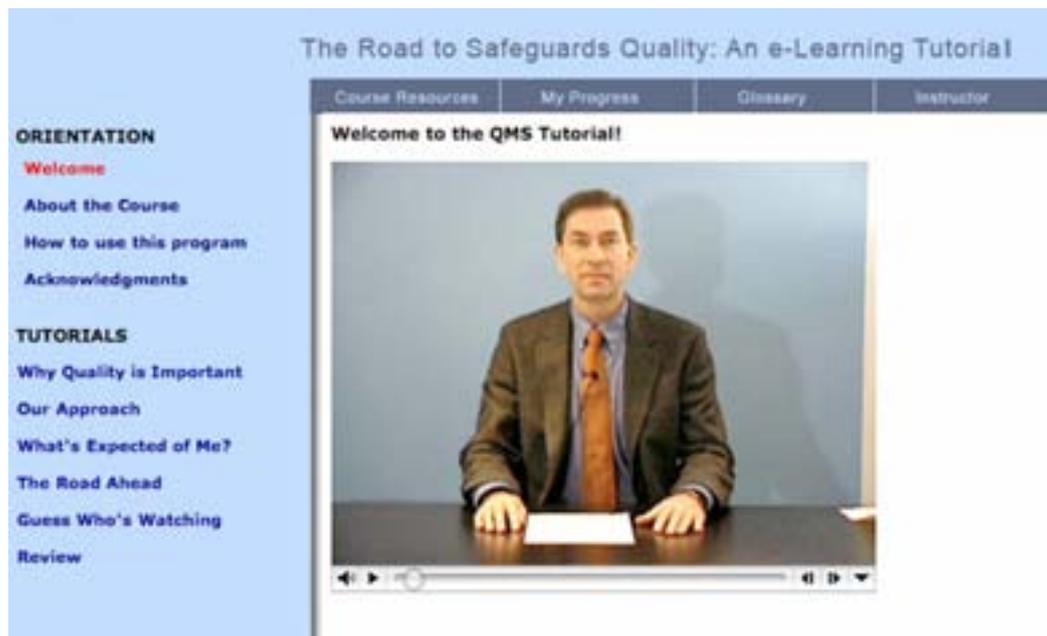
The contractor was required to work within the IAEA's Visual Identity specifications which prescribe such things as fonts for printed materials, a color palette, and the use and treatment of the IAEA logo. Working within the specifications, the graphic designer began by developing a look-and-feel for several types of screens, two of which are illustrated below.

5.1 The Course Menu

The Course Menu consists of:

1. **IAEA Identifier and Course Title Banner**
2. **Utilities** (Course Resources, My Progress, Glossary, Instructor, QMS Home Page)
3. **Orientation Selections** (Welcome, About This Course, How To Use This Program)
4. **Numbered Tutorials** (major topics within the course, with drop-down menus for sub-selections)
5. **Main Presentation Window** (where text and illustrations associated with the current selection are displayed)

The *Welcome* selection appears on screen as the default. The current selection is always highlighted on the menu.

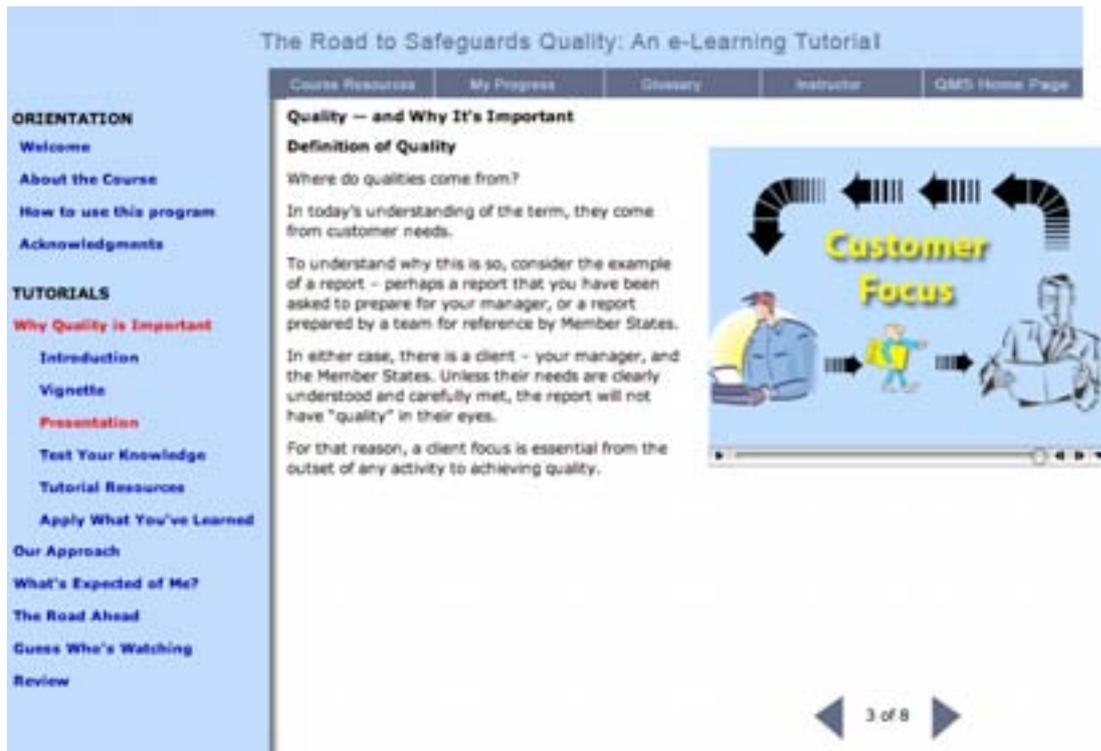


5.2 The Presentation Screen

Presentations are divided into several numbered screens, each covering one teaching point. A presentation screen will normally consist of a text block, [underlined hyperlinks](#) (for definitions, links to reference documents, etc.), and an illustrative “visual” (e.g. a photograph, chart, diagram, animated sequence, etc.)

If there is additional detail that a student might like to know — but is not essential to the teaching point — then a “MORE” button appears. Clicking “MORE” opens a text window that provides the additional detail. A “BACK” button at the end of the text window returns the student to the presentation screen.

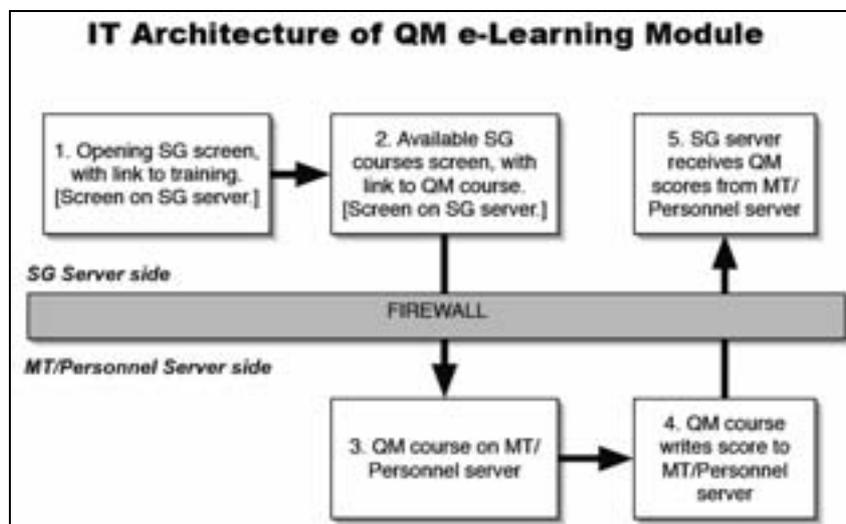
The elements of the *Presentation Screen* are illustrated below.



6. Technical Requirements & Issues

The decision to deliver the course by means of the IAEA Intranet presented several challenges arising from: a departmental Intranet architecture that, for security reasons, includes firewalls that limit outside access to Department of Safeguards materials; and the need to first implement a learning management system for online courses.

The contractor is working with Departmental IT staff to identify ways to overcome firewall barriers without compromising security, to make the program widely available both within the Department of Safeguards and throughout the Agency. The learning management system implemented for this course will support a growing curriculum of on-line courses in coming years.



In addition, to meet the Department's stated requirements, there was a need to implement technical features that would permit the tracking of key user data (e.g. names/passwords of registrants, cumulative figures on course/tutorial usage), as well as the tracking of individual performance such as each user's progress through the course, and final test score to confirm that the minimum standard has been achieved. To protect individual privacy, this kind of information is made available only to authorized course administrators.

7. Effectiveness Verification

End-user evaluations were considered to be important throughout the design, development and implementation cycle to ensure that the finished product is perceived to be attractive, intuitive, functional and useful from the user's point of view. Having course documents, interface designs and prototypes available online provides a means of continuous evaluation and feedback by the Department's team. In addition, efforts are planned to engage end-users at key milestones in the production cycle.

8. Conclusion

After an initial period of heavy usage during which all departmental staff will complete the QMS training module, it is estimated that up to 40 new staff members recruited into the Department each year will complete the *Road to Safeguards Quality* course during their first month of employment. The lessons learned from this seminal course development experience are expected to be applied to other on-line training programs in the years ahead.

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Education on Nuclear Safeguards for European Nuclear Engineering Students

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Key words: nuclear education, training and knowledge management

Abstract

The knowledge retention problem in the nuclear field was acknowledged by the OECD in 2000. ESARDA reacted to that with a strategy to tackle the problem and created a Working Group on Training and Knowledge Management (ESARDA WG TKM). The final objective of the ESARDA WG TKM is the setup of course modules to an internationally recognised reference standard.

This project is in line with the movement of establishing a European curriculum for Nuclear Engineering. Teaching in the Nuclear Safeguards field is indeed strongly influenced by national history so the objective of the course is to provide homogeneous material in safeguards and non proliferation matters at the European level.

This paper reports on the feedback of the course that was held by some of the leading experts in the field of nuclear safeguards in Europe. Its content deals with the general background of safeguards legislation and Treaties, the nuclear fuel cycle, various safeguards techniques, verification technologies and the evolution of safeguards. The audience - 40 university students and 5 young professionals (STUK and JRC) – from 12 different European countries was highly interested and gave positive feedback.

The course has been introduced in the course database of the European Nuclear Education Network on the website <http://www.neptuno-cs.de>. A recognition as academic course of 3 credits under the European Credit Transfer System has been requested to the Belgian Nuclear higher Education Network. In the future this course will be repeated on a regular basis and evaluated, aiming to achieve recognition by the European Nuclear

Education Network (ENEN). With an ENEN-label it is included in the list of optional courses for a European Masters Degree in Nuclear Engineering.

1. Introduction

The declination of the nuclear industry in the last decades of the twentieth century has revealed persecutions in the education of nuclear engineers. European universities did no longer register a minimum number of students for a Master Degree in nuclear engineering. Also the US National Research Council (1990) reported a strong reduction in nuclear engineering students, an extreme high faculty age and shutdown of nuclear research facilities at American universities. The OECD (2000) expressed their major concern about the diminishing and disappearing nuclear knowledge. The International Atomic Energy Agency, the Nuclear Energy Agency of the OECD, the World Nuclear Association of Nuclear Operators and the World Nuclear Association therefore founded the World Nuclear University with 29 nuclear research centres as institutional participants, organizing yearly a nuclear summer course.

2. The European Nuclear Education Network Association

The EC has launched under the 5th Framework Programme a call for addressing this problem and this resulted in the setup of the European Nuclear Engineering Network (ENEN). In parallel to this international project some national satellite networks have been established, such as the Belgian Nuclear higher Education Network BNEN, the Italian Interuniversity Consortium for Research and Technology on Nuclear Energy CIRTEN, and the UK's Nuclear Technology Education Consortium NTEC, the German Education Centrum for Nuclear Technology (TÜV Nord Akademie). According to Van Goethem (2005) the strategy for safeguarding nuclear education and training is based on three pillars: common qualification, mutual recognition, and mobility of scientists and students

The FP5-project ENEN is followed up on the one hand by the NEPTUNO¹ project under the 6th FP and on the other hand with a sustainable European Nuclear Education Network Association. The major objectives of the latter Association is the reinforcement of the three above mentioned pillars and are focused on by the five specific committees: for Teaching and Academic Affairs, for Advanced Courses and Research, for Training and Industrial Projects, for Quality Assurance and for Knowledge Management. The history and the current organisation of this international Association are in more detail described by Giot (2006).

Since 2003 the European Nuclear Education Network Association provides an educational programme for the specialisation in the nuclear field. Students are expected to have already an engineering diploma or equivalent university diploma. Moreover their nationality should be from a country that signed the Non Proliferation Treaty. The complete programme is - although also industry-oriented - taught at European

¹ NEPTUNO represents the Nuclear European Platform for Training and UNiversity Organisations as described on website <http://www.sckcen.be/neptuno/>

universities, profiting from the recognition of the long-established universities, and profiting from the legal aspects of those universities for emitting a diploma, and profiting from the pool of professors that are selected and financed by the universities. The first four students have obtained the Master Degree in Nuclear Engineering in 2005.

It takes minimum one academic year (60 ECTS² credits – corresponding of one final year of university studies with 5 courses of 6 ECTS – 5 courses of 3 ECTS and a thesis of 15 ECTS) for accomplishing these studies. The student can select these courses out of a large variety offered by 24 universities. The database of courses is open for consultation on the website <http://www.neptuno-cs.de> and an analysis of the up to 250 courses in there leads to the conclusion that a safeguards or non-proliferation course is not offered by any university.

3. ESARDA strategy for safeguards education

This shortcoming on education in safeguards was discussed by ESARDA and a strategy to tackle this problem has been defined by the Steering Committee in several steps. As published by Bril (2004) the first ESARDA exercise consisted in the setup of a glossary, followed by technical sheets. Both can be found on the ESARDA website and the latter is still ongoing. The next step was the Course Modules initiative, an idea launched in September 2002. Upon positive evaluation of the demand or interest for these Course Modules, a first small task group was in May 2003 officially formed with Mr. G. Stein, Mr. S. Guardini (replaced by Mr. G. Maenhout in 2004), Mr. K. Van der Meer. This group, called the Training Knowledge Management Working Group actively started to prepare course modules in 2005.

4. The 2005 prototype of a three-days safeguards course

Upon request of the students from the Belgian Nuclear Education Network a first Nuclear Safeguards and Non-Proliferation course has been established by the JRC in collaboration with the SCK.CEN in Ispra from 1st till the 3rd March 2005. The course was attended by 10 university students and 8 young professionals, as shown in Fig. 1. The feedback of the students and the experience at the JRC and the SCKCEN was positive. More details on the course and its feedback is given by Janssens-Maenhout & Poucet (2005) and by van de Meer et al. (2005). The ENEN students made a report on the total content of the safeguards course and the students from the University Ghent worked out a study on the illicit trafficking trend and their evolution from before 1990, between 1990 and 2001, and in the post-era.

² The so-called 3 ECTS = 1 teaching module at university with 20 hours of lecture and 10 hours of exercises, laboratory sessions and seminars. ECTS stands for “European Credit Transfer System”, defined in the Sorbonne-Bologna process for harmonisation of the university courses (need for exchanging students, e.g. under ERASMUS). (Students can follow courses at other universities and it is well-known what their value are).

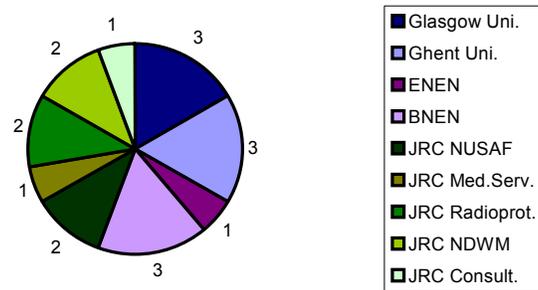


Fig. 1: Distribution of students attending the Safeguards course in Ispra, 1-3 March 2005

5. The first ESARDA course 2006 on Nuclear Safeguards and Non Proliferation

5.1. Content

The first ESARDA course was discussed in content and organization by the ESARDA Training and Knowledge Management Working Group and guaranteed a complete safeguards overview, presented by the major stakeholders (nuclear industry and regulatory authorities) taking into account the presence of the various nationalities in the EC and including the research & development (with involvement of the research centres). The final course schedule for the four days course with theoretical lectures, a class room exercise and some practical visits is given in Fig. 2.

Monday	Tuesday	Wednesday	Thursday visit-scheme is group-dependent
9:00 Introduction (Mr. J. Joly - IRSN)	Basic principles of safeguards (SSAC, NMAC, NRTA, ...) (Mr. B. Burmas - ENG)	9:00 information collection and analysis (Mr. J. Bault - IAEA)	9:00 visit to PERLA laboratory
10:15 Nuclear Fuel Cycle (Ms. C. Jorant - AREVA)	10:15 accountability from statistical point of view (Mr. M. Franklin - JRC)	10:15 coffee break	10:00 exit of controlled area
11:30 Coffee break	11:30 coffee break	10:30 Iraq case study (Mr. J. Bault - IAEA)	10:15 visit to TAME laboratory
11:45 Non proliferation aspects of the nuclear fuel cycle (Mr. J. Joly)	11:45 Monitoring (of CIS, of process, ...) (Mr. P. Funk - IRSN)	11:45 New challenges in security: illicit trafficking (Mr. L. van Dassen - SKI)	11:00 visit to Surveillance labs
13:00 lunch	13:00 lunch	13:00 lunch	12:15 visit to SILAB
14:00 History of Nuclear Safeguards and Non-Proliferation (Mr. T. Jonker, Stockholm Uni)	14:00 Inspections on site (DNV, PIV, ... "surprises") (Mr. P. Schwabach - EURATOM T)	14:30 NDA equipment: neutron/gamma for inspectors (Mr. P. Peerani - JRC)	13:00 lunch
15:15 Coffee break	15:15 Environmental monitoring (Mr. M. Kainowski - CTBTO Prepcom)	16:00 Coffee break	14:30 Some Proliferation Questions (Ms. G. Meerhout - JRC)
16:30 Overview treaties: NPT, AP, CTBT ... (Mr. A. Pouzet - JRC)	16:30 coffee break	16:15 Open discussion topics from journal/newspaper articles/ internet info: questioned	15:15 coffee break
18:45 exercise: how to setup a verification of a certain state (Ms. M. Hunt - IAEA)	18:45 DA & nuclear forensics (Mr. K. Mayer - JRC)	18:00 Closure with wrap-up of safeguards principles	15:30 Closure with feedback exchange
18:00 Proliferation and control: impact for industry (Ms. C. Jorant - AREVA)	18:00 Import/ Export control (Mr. M. Kainowski - CTBTO Prepcom/ Uni Hamburg)		

Fig. 2: Schedule of the first ESARDA Course on Nuclear Safeguards and Non Proliferation in Ispra, 6-9 March 2006.

The first day aimed to give an overview on the different safeguards aspects, from legal point of view and from industry point of view. Mr. Joly, ESARDA president introduced

the course. Ms. Jorant, AREVA director for Non-Proliferation and International Institutions, addressed the fuel cycle and the non-proliferation aspects in there. In particular she illustrated how non proliferation concerns are dealt with within the nuclear industry as major stakeholder participating in the worldwide effort towards a non proliferation culture. Mr. Jönter, Stockholm University, describing the historical evolution of the safeguards system answering whether a nuclear non proliferation system exist today in Europe. Mr. Poucet, Katholieke Universiteit Leuven, gave a seminar on arms control treaties (including the Treaty on Non-Proliferation of Nuclear Weapons (NPT), the Conventions on Biological (BWC) and Chemical Weapons (CWC), the Comprehensive Nuclear Test Ban Treaty (CTBT), the Conventional Forces in Europe Treaty (CFE), the Open Skies Treaty, the Antarctic Treaty, the Treaty of Tlatelolco, the Intermediate Nuclear Forces Treaty (INF) and the Strategic Arms Reduction Treaty (START)) and verification systems. The first day was closed with the setup of a class room exercise by Ms. Hunt: “How to setup a verification of a certain region?”.

The second day focused on the basic principles and logic of nuclear material accountancy and control (NMAC), inspections, monitoring and import/export control. Mr. Burrows addressed the material management principles and in particular what is different about nuclear material management, and defined clear components of the NMAC system. This was completed with a statistical point of view, given by Mr. Franklin on the auditing of nuclear material accountancy. Mr. Funk addressed the containment and surveillance aspects, while Mr. Schwalbach reported on Euratom’s on-site inspection strategy. The destructive analyses as carried out in the analytical laboratories to determine accurately in the samples the concentration of isotopes or to identify the presence of some isotopes on some swipes was addressed by Mr. Mayer, with some examples of nuclear forensics. Mr. Kalinowski completed this with more details on environmental sampling and with the explanation how to perform import/ export control.

The tools of inspectors for collecting data (by exploring open sources or by measuring) and for analyzing it have been described on the third day of lectures. Mr. Baute clearly indicated the difficulty to collect and analyse the data. He reported life on the challenges and lessons learned from nuclear inspections in Iraq that his team from the International Atomic Energy Agency carried out in collaboration with the United Nations Monitoring, Verification, and Inspection Commission. To collect independent data, it is needed to carry out on some samples non destructive assay (NDA), applying neutron counting and gamma spectrometry. The functioning of the latter NDA inspectors tools has been explained by Mr. Peerani. Finally Mr. van Dassen provided some real examples from his collaborations in Russia on nuclear safeguards and illustrated the practical difficulties in the fight against illicit trafficking. The lecture sessions was then closed with the discussion of the exercise results with the different group of students.

A fourth day was organized to give some practical feeling with some visits to laboratories: the Performance Laboratory (PERLA) with an extensive collection of well-characterised nuclear reference materials and non destructive analyzing techniques, the Tank Measurements Laboratory (TAME) for total inventory calibrations, densitometry and solution monitoring, Seal and Identification Techniques Laboratory (SILAB), for safeguarding with authenticated seals all nuclear material (such as fuel assemblies) in storage places or containers or transport casks, the Surveillance and Information Retrieval

Laboratory (SIRLAB) with 2D/3D laser surveillance system and 3D image reconstruction tool for remote verification.

5.2. Participation and feedback

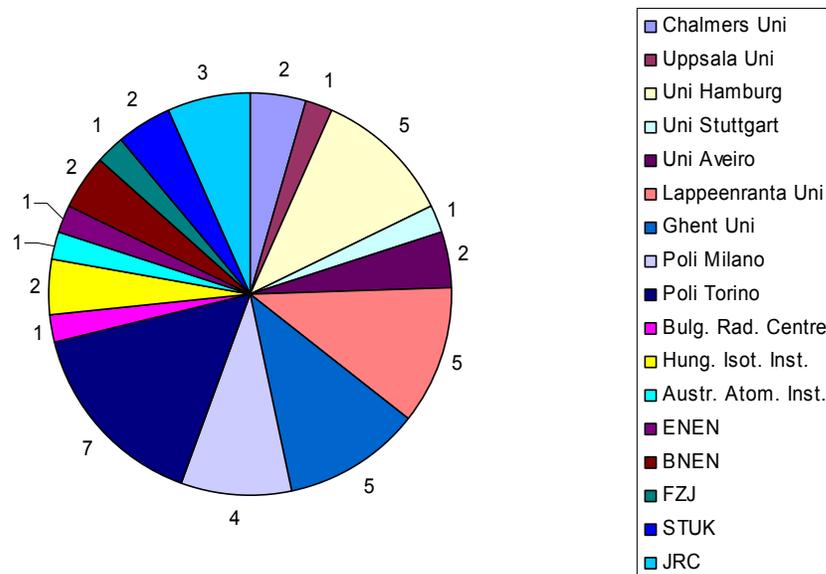


Fig. 3: Distribution of students attending the first ESARDA Safeguards course in Ispra, 6-9 March 2005

The course was attended by 45 participants, of which 40 students from various universities, spread over 10 different European countries. Fig. 3 presents the distribution of the 45 course attendees with their affiliation. The course was highly appreciated by all participants with positive feedback on the content of the lectures, the exercise and the practical demonstrations during the visits. The lecturers had to reply many questions of the interested audience.

As feedback the students suggested to spread the course schedule over five days, including more exercises, also hands-on exercises in the labs. Many students have been surprised about the many different actors in the nuclear safeguards world and the “slang” they use. It is desired to address this in more detail. Although in general the students prefer the lectures on more technical topics, the coupling between the technical issues and the politics with factual illustrations opened significantly their perception.

6. Conclusions

The first ESARDA course was a success, shown by the numerous participants and the positive feedback. The course will be yearly repeated (tentatively in the first week of March) and recognition of this course with 3ECTS in an academic curriculum with the BNEN (ENEN) label is requested to the appropriate committees. Future repetitions, slightly modified to cope with the suggestions, will be open to both university students and professionals from industry with parallel sessions. The support of the lecturers and

their organizations will be further needed. It is also the aim to issue a publication on the course material, contributed by each lecturer as author.

The ESARDA Training and Knowledge Management Working Group is engaged to steadily enhance the course in content and organization, taking into account students' suggestions. But there is still a long way to go for a sustainable ESARDA safeguards course.

7. References

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Giot (2006), Enseignement et formation en sciences et technologies nucléaires: le défi des compétences, *Revue Générale Nucléaire*, **1**, Jan.-Fév. 2006, pp.56-62

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Education on Nuclear Safeguards for European Nuclear Engineering Students

ESARDA Working Group Training and Knowledge Management: G. Janssens-Maenhout, A. Hamilton, T. Jönter, G. Stein, E. Martikka, A. Håkansson, K. van der Meer, B. Autrusson, R. Howsley
Lecturers ESARDA course 6-9 March 2006: J. Baute, B.A. Burrows, M. Franklin, P. Funk, M. Hunt, J. Joly, C. Jorant, M. Kalinowski, P. Peerani, A.E. Poucet, K. Mayer, P. Schwalbach, L. van Dassen, IPSC, Joint Research Centre Ispra: L.-V. Bril, J. Gonçalves

JRC course, Ispra March 2006

- Two days of lectures: Fuel cycle, treaties, strategy, methods, tools.
- One day visit to JRC labs NDA, M/V, surveillance, sealing, satellite imagery
- Evaluation: Case study, reports, presentation

Suggestions and Conclusions

- Increase course to 5 days allowing more time for exercises.
- Include an overview of the safeguards community with explanations of abbreviations used.
- Repeat course annually as part of BNEN curriculum.
- 8 students now involved in writing evaluation essay.

47 students participating from:

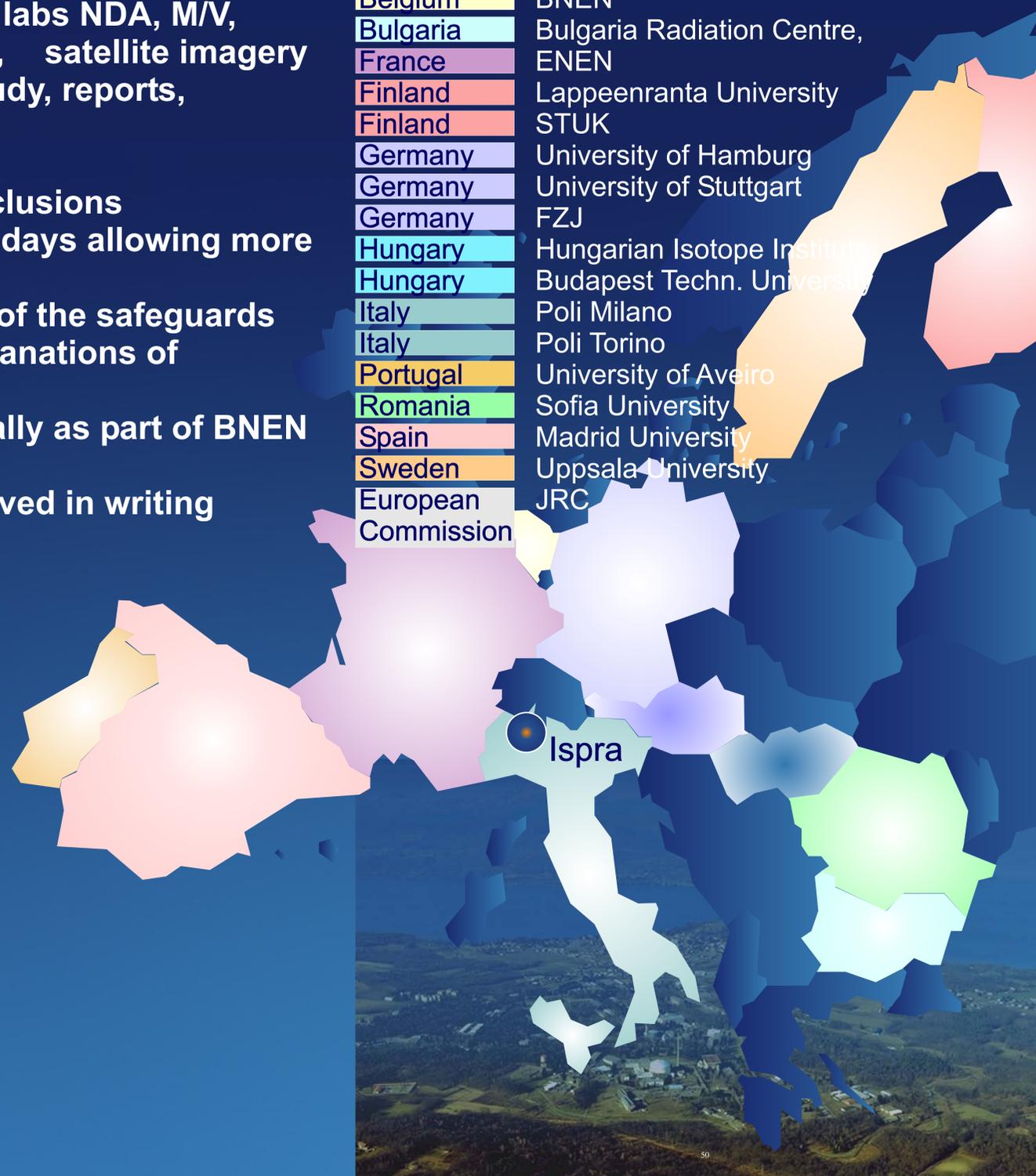
Austria	Austrian Atomic Institute
Belgium	Ghent University
Belgium	BNEN
Bulgaria	Bulgaria Radiation Centre,
France	ENEN
Finland	Lappeenranta University
Finland	STUK
Germany	University of Hamburg
Germany	University of Stuttgart
Germany	FZJ
Hungary	Hungarian Isotope Institute
Hungary	Budapest Techn. University
Italy	Poli Milano
Italy	Poli Torino
Portugal	University of Aveiro
Romania	Sofia University
Spain	Madrid University
Sweden	Uppsala University
European Commission	JRC

Lectures of First ESARDA Course Ispra 6-9 March 2006

- Introduction
- Nuclear Fuel Cycle
- Non-proliferation aspects of the Nuclear Fuel Cycle
- History of Nuclear Safeguards and Non-Proliferation
- Overview treaties: NPT, AP, CTBT
- Proliferation and control: Impact for industry
- Basic principles of safeguards: SSAC, NMAC, NRTA
- Accountancy from statistical point of view
- Monitoring of C/S processes
- Inspections on site: DIV, PIV
- Environmental Monitoring
- DA and nuclear forensics
- Import/Export control
- Information collection and analysis
- Iraq case study
- New challenges in security: Illicit trafficking
- NDA equipment: Neutron/gamma for inspectors

Additional to the lectures

- Exercise on setting up a verification of a state
- Visit to PERLA laboratory
- Visit to TAME laboratory
- Visit to Surveillance laboratories
- Visit to SILAB
- Proliferation questions answered
- Open discussion on media topics
- Closure with wrap up of safeguards principles
- Feedback from students



Pool of 2006 Course Lecturers

- J. Baute* joined the IAEA in 1994 and became director of Iraq's Nuclear Verification Office. Presently he is director of the IAEA Safeguards Information Technology Directorate.
- B. Burrows* joined the British Nuclear Group in 1975 for nuclear material management and is currently the BNFL Group Manager for International Safeguards
- M. Franklin* has been working in the JRC safeguards program since 1978 and is specialist in mathematical statistics
- P. Funk* is since more than 10 years involved in the French and International safeguards and is leading the Containment/surveillance lab at the Institute for Radioprotection and Nuclear Security
- M. Hunt* has been Nuclear Safeguards inspector of IAEA for the Commonwealth of Independent States, and is presently IAEA training coordinator
- C. Jorant* is presently director of Non-Proliferation and International Institutions in the International and Marketing Department of *Companie Générale des Matières Nucleaires (AREVA)*
- T. Jonter* is heading the Department of Economic History at the Stockholm University, leading educational programmes on Nuclear Non Proliferation at different universities in former Soviet Union
- M. Kalinowski* is director of the Carl-Friedrich von Weizsäcker Center for Science and Peace Research at the University of Hamburg. He joined the International Data Center of the Provisional Technical Secretariat of the Preparatory Commission Comp. nucl. Test Ban Treaty Organization

- K. Mayer* is responsible for analytical methods for nuclear material measurements at the JRC Karlsruhe with experience on actinide reference materials
- G. Maenhout* joined in 2001 the nuclear safeguards unit and is leading the activities on process monitoring of nuclear facilities.
- P. Peerani* leads the physical modeling (e.g. Monte Carlo) for nuclear measurements (NDA, solution monitoring) at JRC Ispra with experience as analytical inspector.
- A. Poucet* has been head of the Non Proliferation and Nuclear Safeguards Unit at the JRC Ispra, and is presently head of the Traceability and Vulnerability Assessment Unit.
- P. Schwallbach* joined the EC as EURATOM inspector in 1992 and is heading the logistic support for nuclear material verification
- L. van Dassen* joined in 2001 the Swedish Nuclear Power Inspectorate where he is presently Deputy Head of the International Cooperation Programme, leading non-proliferation assistance to former Soviet Union

Course Internet Site

ESARDA WG TKM: http://esarda2.jrc.it/internal_activities/WC-MC/Web-Courses/index.html Joint

Other links:
Nuclear Safeguards Unit: <http://npns.jrc.it/frameset.html>
Joint Research Centre: <http://www.jrc.cec.eu.int>

**2007 COURSE
Nuclear Safeguards
and Non Proliferation**



Ispra, Italy, March 5-9, 2007

**Organised by the
European Safeguards Research and Development Association**

**Hosted by the
Joint Research Centre Ispra, Italy**

Origin of the course

The knowledge retention problem in the nuclear field was acknowledged by the OECD in 2000. The United Nations study on disarmament and non-proliferation education (2002) made detailed recommendations for urgently required improvements. ESARDA, the European Safeguards Research and Development Association reacted to these shortcomings with a strategy to tackle the problem and created a Working Group on Training and Knowledge Management (ESARDA WG TKM). The final objective of the ESARDA WG TKM is the setup of academic course modules to an internationally recognised reference standard.

This project is in line with the movement of establishing a European curriculum for Nuclear Engineering. Teaching in the Nuclear Safeguards field is indeed strongly influenced by national history so the objective of the course is to provide homogeneous material in Nuclear Safeguards and Non-Proliferation matters at the European and international level.

Learning objectives

This compact course is open to masters degree students, in particular nuclear engineering students, but also to young professionals and International Relations/ law students. It aims at complementing nuclear engineering studies by including nuclear safeguards in the academic curriculum.

The basic aim of the course is to stimulate students' interests in safeguards. The course addresses aspects of the efforts to create a global nuclear nonproliferation system and how this system works in practice: the Treaty on Nonproliferation of Nuclear Weapons (NPT), safeguards technology, and export control. Also regional settings, such as Euratom Treaty, are presented and discussed. The course deals in particular with technical aspects and application of safeguards; i.e. how to implement the safeguards principles and methodology within the different nuclear facilities. Therefore the course will create an overview on inspections techniques, ranging from neutron/ gamma detectors, to design information verification, to environmental sampling, etc.

Course content

Introduction: The evolution of the Non Proliferation Treaty -regime, safeguards, international control regimes in theory and practice, and present trends in the nuclear nonproliferation efforts.

What is safeguarded: Definition of nuclear material that is subject to nuclear safeguards and related safeguards goals (significant quantity, timeliness and detection probabilities)

Where is it found: Description of the nuclear fuel cycle from mining to final repository, focussing on enrichment in the front-end and reprocessing in the back-end

Which legal protection means exist: Overview on international and regional Non-Proliferation Treaties and established Institutions and Organisations

What is the methodology to verify: Nuclear material accountability principles and statistics of auditing

How are inspections performed: Overview on inspector tools and their use to verify the nuclear activities as declared under the safeguards agreements (Non Destructive Assay, Monitoring, Containment/ Surveillance); additional safeguards measures under the Additional Protocol (complementary access, satellite imagery, environmental sampling) and how they are applied in field (storage facility, process facility, enrichment facility, research institute, spent fuel transfer)

How to control Import/ Export: Guidelines of the Nuclear Suppliers Group, trigger list and dual-use list. Means to combat illicit trafficking, inclusive nuclear forensics

What additional information offers: Collection of open source data and demonstration of some case studies (Iraq, 1993)

Practical organisation

The course features a full five-days program with 75 minutes lectures by experts in the field of nuclear safeguards, visits to five safeguards laboratories and a classroom exercise.

The course material, consisting of a complete set of presentations and literature will be provided to the participants. It is recommended that the students prepare themselves with the reading material on the website

For this limited enrolment course early registration is recommended. The registration form on <http://esarda2.jrc.it/> under the link to the Working Group of Training and Knowledge Management has to be completely compiled. University students can apply for accommodation free of charge, but only a limited number of places per university are available.

There is no course fee

All participants are encouraged to make a poster on a given topic accompanied with 4 page explanatory paper. The best posters will be selected for presentation at the ESARDA Symposium (Aix en Provence, May 22 -24, 2007)

Upon request the students can be quoted for this course with an additional test. In 2006 the course was recognised for 2ECTS by the Belgian Nuclear Higher Education Network.

Venue: JRC Ispra, Building 36, Amphitheatre

Schedule: from Monday, March 5^o, 2007 at 9:00 to Friday, March 9^o, 2007 until 18:00

Preferred communication mode: email: greet.maenhout@jrc.it

Experiences in Germany with the implementation of the additional protocol

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Germany

Like in other countries of the European Union (EU), experience has been gained in Germany with the implementation of the Additional Protocol, which had been signed in September 1998 and, after the relevant national law had been published in February 2000, the Additional Protocol entered into force in the EU on April 30th, 2004.

The political conditions in Germany are set by the EURATOM Treaty established in 1958, in which the sovereignty on nuclear material was transferred to the European Atomic Energy Community (EURATOM). This means, e.g., that all nuclear material in the EU is the property of the EU, represented by the European Commission. The authority with respect to these issues lies, by direct European law, with the EU, and, therefore, Germany has no State System of Accounting for and Control of Nuclear Material (SSAC). Another political factor, since 1998, has been the German phase-out policy on nuclear electricity production. This means, for example, that there are no state-funded nuclear R&D activities to be reported under the Additional Protocol.

In Germany, some preparations for the implementation of the Additional Protocol had started, in cooperation with the nuclear research capacity and the nuclear industry, already in 1996, when the Additional Protocol was still under negotiation.

First of all, the actual number of facilities, LOFs and CAM-holders had to be investigated, as it was already clear at that time that the lists of EURATOM and the IAEA were not reflecting the reality any more. In the past, EURATOM had made very limited use of the possibilities to request an exemption for nuclear material used in non-nuclear applications. The IAEA in turn had not consequently followed up the status of nuclear installations once they were reported as closed down and all nuclear material was removed. The lists included, for instance, chinaware factories and glass or bulb manufacturers, and, therefore, the second task was to investigate into the type and inventory of material that was handled, and to identify those installations which were relevant for the Additional Protocol.

A third task was to investigate which companies from other industry branches were relevant for the Additional Protocol under Annexes I and II. These companies (not being familiar with safeguards) had to be informed about the whole background and about their legal obligations.

A specific problem was the site definition for the former nuclear research centres: Most of the institutes were “de-nuclearised” in the 1990s after commercial reprocessing, fast breeder and high-temperature reactor projects had been abandoned in Germany, so that the question arose which institutes were relevant for the Additional Protocol. The issue was studied in 1998/99 using the example of the Jülich Research Centre [1], where at that time only 5% of the activities were related to energy-relevant nuclear science and technology.

The phase before the entry into force of the Additional Protocol was the identification and definition of the sites, provision of information for the operators, preparation of the expanded declarations, and the application of the AP-related software provided by IAEA and EURATOM. These activities were carried out in close co-operation with EURATOM because Germany transferred most of the obligations assigned to the state in the Additional Protocol to EURATOM and the German operators report their AP declarations directly to EURATOM.

As listed in the declaration of December 31st, 2005, there are 51 sites in Germany, which include 91 Material Balance Areas (MBA). At 11 of these sites nuclear facilities are under decommissioning, and one site consists of a hall with scrap. In addition, there is a former mine which is being leached out and where the Uranium that is gained is a result of the cleaning process of the liquid residues.

In addition, there are 6 manufacturers of Annex I-goods.

The task to update the lists of nuclear installations is still in progress. The books outside the declaration under the Additional Protocol further include 29 LOFs (17 of which have no nuclear relevance) and another 28 non-nuclear LOFs which either have no nuclear material in their books or which even may not exist any more. From an additional 192 installations which are part of the Catch-All-MBA (“CAM”) and where material is in non-nuclear use, at least 20 are not existing any more. Whereas EURATOM had verified the empty status of a former LOF or facility already in previous years, the IAEA had been hesitant to join these inspections. Instead of making “full use of the Community’s system of safeguards in accordance with the Agreement” by accepting the results of EURATOM’s final inspections, the IAEA’s interest in these locations was raised after the entering into force of the AP in 2004. Now, the IAEA’s requests are causing considerable effort and extra costs.

At some point, there had been even the idea brought into the discussion that all these places should be included in the declaration of the Additional Protocol. However, this idea was not further pursued, as the Additional Protocol is clearly directed to nuclear activities and is not supposed to include, e.g., a hospital which uses a shielding of depleted Uranium for its Cobalt cobalt-60 source.

Any attempt to have access to buildings of former LOFs or to the areas where such buildings had once been standing, has no clear legal basis in the German law. And even in those few cases, where the operator is still holding the same place, there is no obligation to grant access. As an additional problem even in such a place, the number of people who are familiar with its “history” is decreasing.

Since the beginning of 2005, there have been (until the end of July 2006) 16 Complementary Accesses at a total of 12 sites. In addition, there were extended inspection activities at other facilities.

The implementation of the Additional Protocol in Germany gives a differentiated picture. In most cases, the inspectors know the legal basis of their activities. But there have also been cases of a very individual interpretation.

Some IAEA staff members translate, e.g., the definition of the term “decommissioned“ in Article 18(c) from “...residual structures and equipment essential for its use ...“ into “...residual structures and **all** equipment essential for its use have been removed or rendered inoperable ...“. Such a definition increases the number of sites unnecessarily.

Furthermore, it has happened that, in order to avoid a formal “Complementary Access“, a “visit“ has been asked for. Such “visit“ has no legal basis, and, in addition, there is no obligation afterwards to provide information on the results of a visit to EURATOM and to the State.

Another problem is the high demand of some IAEA inspectors for photos. Photography has always been a safeguards tool for specific cases by comparing the reality with the “should-be status“ documented in a picture. Examples are the comparison of a weld seam of a specific fuel item for identification or, in another case, of the complex piping of a specific facility with their respective photos.

However, it seems that there are no transparent criteria for the conditions under which photos are deemed necessary. There are examples that, upon request of the IAEA inspector, pictures have been taken that have no apparent value for the subsequent inspection. Some of them show components of a facility which are common in all facilities of such type, which an inspector should be familiar with from similar facilities. Others show situations (e.g., boxes next to a wall) that can change at any time because of the operational procedures.

Claiming the use of photography for “documentation of a visual observation“ may be a reasonable approach for complex situations. But due to the interference with other aspects, it is not an appropriate method to show that the inspector has been there: This is sufficiently explained in the texts of the statement and of the inspection report, which may even include measurement and analytical results, where applicable.

Some observations and experiences with the application of the Additional Protocol were briefly addressed in this paper as a contribution to the learning process on all sides, i.e., within the IAEA, EURATOM, and the member states.

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Preparation for implementation of the safeguards system in the Republic of Tajikistan

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Tajikistan is not a nuclear state, though mass media wrote and writes a lot about uranium deposits, their exploration, development and production in Tajikistan. During many years already are not operating the available uranium processing plants because of lack of raw material.

But, in tailings and waste fields of Tajikistan has been accumulated a considerable amount of radioactive wastes. Till 1990 on the territory of the Northern Tajikistan in the result of industrial exploration of uranium mines and processing of raw material have been created tailing pits of radioactive wastes, which are huge according to their capacity – more than 50 millions of tons and activity – about 6 thousand curies.

They are allocated in the area of more than a thousand hectares. Only one of the operating tailings (the Degmai-settlement), which occupies the territory of 90 hectares and where allocated 19,8 million of tons of radioactive wastes with activity of 4218 curie, has radiation background, which reaches 700-900 microcurie per hour. Most of “tails” do not have proper covering and are very dangerous for the environment [1].

Another, uncovered radioactive tailing pit “Waste of base ores” is located in Taboshar-city. It has the following properties:

- area of 4 hectares;
- amount of 2.03×10^6 tons;
- activity of 253 curie;
- radiation background – 300-500 microcurie per hour.

The danger of this object is that it has the pyramidal shape, raise above relief and is located in the mudflow basin. In the heavy and enduring rains period there has place wash-out of radioactive material from the body of tailing pit as well as erosion of “wastes” and agricultural fields and pastures are being contaminated. Except this, a strong wind carries away the particles of radioactive material to other areas, too.

Thus both of radioactive tailing pits (“Digmay” and “Waste of base ores”) pose most considerable threats for the population and the environment of the Northern Tajikistan. These tailing pits cause many problems in the medium-term and long-term perspective for both our state and transboundary states. Most of “tails” do not have sufficient cover, which negatively reflects on the contamination risk factors of the region and presents a big ecological problem.

It is well-known that during the Soviet period the radiation safety service kept most of negative data in secret from society. Today we are constructing a democratic state, and, along with other institutes, we are reforming the radiation safety service of the country.

Tajikistan became the Member State of IAEA in 2001. According to the Order of the Government of the Republic of Tajikistan, the Academy of Sciences became the contact point of Tajikistan for cooperation with the IAEA. In January of 2003 at the framework of the Academy of Sciences was established the Nuclear and Radiation Safety Agency, which on the basis of Article 6 of the Law of the Republic of Tajikistan “About Radiation Safety” is the Regulatory Authority of the Republic of Tajikistan on Problems of ensuring radiation safety in the Republic of Tajikistan.

Our Nuclear and Radiation Safety Agency successfully cooperates with the IAEA Safeguards Department, where the particular place in this cooperation have problems of creation of the legislative base on radiation safety. Today in Tajikistan we have four laws and a number of regulations on provision of radiation safety adopted:

1. Law of the Republic of Tajikistan “About radiation safety”, which was adopted on August 1, 2003 having No. 42;
2. Law of the Republic of Tajikistan “About Utilization of Atomic Energy” from 9 December 2004 having No. 69;
3. Law of the Republic of Tajikistan “About licensing of special particular types of activities”;
4. Law of the Republic of Tajikistan “About provision of sanitary and epidemiological safety of population” from December 31 2000 having No. 522.

In accordance with the Decision of the Government of the Republic of Tajikistan from December 3, 2004 No. 482 “About ratification of the Regulation on State regulation in the field of ensuring radiation safety” with the purpose of collective solution of problems and coordination of the activity of the appropriate state authorized bodies in the field of radiation safety was created the Interagency Committee and its membership was approved by the Decision of the Government of the Republic of Tajikistan from December 2, 2005 No. 471.

By the Decision of the Government of the Republic of Tajikistan have been approved:

1. Regulation “About State regulation in the field of Ensuring the Radiation Safety” from December 3, 2004 No. 482;
2. Regulation “About licensing peculiarities of separate types of activities” from September 1, 2005 No.337;
3. Regulation “About Interagency Committee on Ensuring the Radiation Safety” from December 2, 2005 No. 471.

The Nuclear and Radiation Safety Agency (Regulatory Authority) successfully cooperates with the IAEA Safeguard Department. There have been a number of expert missions of the IAEA Safeguard Department to Tajikistan. We visited together with experts:

- Tailing pits and plant on synthesis of uranyl-uranate;
- Republican waste repository site;
- Inoperative research reactor of the Physical and technical Institute of the Academy of Sciences of the Republic of Tajikistan;
- Establishments of the Ministry of Health; Ministry of education and Ministry of Extreme Situations.

In 2003 our specialists together with experts from the IAEA Safeguard Department liquidated the threat of radiation contamination, which was caused by the helicopter crash in the mountains of Northern Tajikistan.

The IAEA Safeguard Department provided us assistance in equipping the Ministry of Extreme Situations with the most necessary equipment and devices (dosimeters, radiometers, and etc.).

We continue working on preparation of legislative documents on radiation safety. Today we are preparing the projects of the following regulations:

- State system of accounting and control of nuclear material;
- About individual doses of professionals from ionizing radiation sources;
- About requirements on ensuring radiation safety;
- Norms of radiation safety;
- About transportation of radioactive wastes;
- And etc.

These documents have been developed taking into account the exploitation experience and international requirements. In the development of the draft documents participated IAEA experts.

The Republic of Tajikistan signed and ratified The Non-proliferation Treaty in 1995 before her membership to IAEA. Our republic has signed the following universal is the member of the following universal treaties:

1. Nuclear Non-Proliferation Treaty (1995);
2. Convention on the Prohibition of Military or Any Other Hostile Use of Environmental Modification Techniques (1978);
3. Convention on Physical Protection of Nuclear material (1996);
4. Comprehensive Nuclear Test Ban Treaty (1998);

5. Agreement between the Republic of Tajikistan and the IAEA about application of safeguards in connection with the Non-Proliferation Treaty (2003);
6. Protocol Additional to Agreement between the Republic of Tajikistan and the IAEA about application of safeguards in connection with the Non-Proliferation Treaty (2003);
7. Protocol amending the Agreement between the Republic of Tajikistan and the IAEA about application of safeguards in connection with the Non-Proliferation Treaty (2006).

We hope that our successful and close cooperation with IAEA and recognition of international agreements and treaties by our Republic will allow:

- ✓ the international society to recognize us as reliable partners;
- ✓ us to improve the Republican system of radiation protection of population up to the level of developed states.

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Experience of the preparations for implementing integrated safeguards in Kazakhstan

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Abstract. For the last years the Kazakhstan State system for nuclear material accounting and control was appreciably strengthened and at present it can be the foundation for implementing of the integrated safeguards in Kazakhstan. The effectiveness of the system had been examined during the spent fuel packing process on the shut down facility BN-350. Kazakhstan signed Additional Protocol and carry out preliminary work to prepare initial declaration.

1. Introduction

The state system for accounting and control of nuclear materials in the Republic of Kazakhstan consists of two levels:

- State level: The Kazakhstan Atomic Energy Committee (KAEC) is engaged in collection of industry inventories, preparation of the reports to the IAEA and storage of accompanying and other essential data, as well as maintenance of database on exported and imported materials and a database on material inventories at the facilities.
- Facility level: Registration and processing of data started from key measurement points and creating a database of nuclear materials inventories.

Nuclear materials accounting at the various facilities is distinguished from each other due to variety of the materials, performing functions and material transfer. Each facility has a separate department for nuclear materials accounting. Materials are subject to overall monitoring on the enterprise's territory beginning from the moment of the material receipt. The accounting system includes maintenance of database of nuclear materials by all the key measurement points, reporting to the KAEC, and annual reviews of system efficiency by means of physical inventory takings. All the operations related to the management and accounting of nuclear materials are carried out in compliance with standard requirements established by the regulatory body. All facilities are subject to IAEA safeguards, the material is verified, and furthermore, the IAEA organizes inspections. All the facilities are equipped with computer systems, which include technical and organizational aspects. The database has complete information about all the items of nuclear materials and their characteristics: quantity, isotopic content, and location.

At present all the facilities have been licensed. In accordance with the requirements for the facilities involved in nuclear material management, the structure of the systems for accounting and control, facility regulations currently in force and procedures on nuclear material handling have been analyzed. It allowed for both state body and facilities to revise the existing system for accounting and control of nuclear materials. The result of this activity was improvement of the present system and the regulatory base, the development of the work

plan for systematic analyses and correction of the system as a whole. The donor-states rendered invaluable assistance in this improvement.

2. Examination of the system

The effectiveness of the system had been examined in a number of activities realized after shutdown of BN-350 facility.

2.1. Repacking of spent fuel

After decision on the reactor shutdown had been reached in 1999, it was decided to repack the spent fuel for following transportation and long-term storage. Arrangements were intended for continuous monitoring by the IAEA's inspections and facility staff, that required development of additional regulatory measures and instructions.

The "Technological regulation for packing of spent fuel of the BN-350 reactor" was developed; it contains detailed description of the personnel actions during the package process and measures on accounting and control of the nuclear materials. In addition the instruction for facility operators and IAEA inspectors' cooperation was developed, since all work was carried out under IAEA's permanent control. While packing all the fuel assemblies have been measured. Measured data and information on changing the storage location have been registered in the database. The record cards for each new fuel batch were filled up. The maps of the cooling pond and database were corrected.

The special forms containing information on old and new items were developed in order to present the information to IAEA's inspectors on the work undertaken.

All the movements were registered in the reports as re-batching. This procedure was applied to a reactor for the first time. Usually re-batching is used in the facilities with bulk form material.

2.2. Fresh fuel transportation

The transportation of the part of the unused nuclear fuel (fuel assemblies) from the stopped BN-350 reactor to the fuel fabrication plant was the second test of the system effectiveness.

In accordance with long-term schedule of the Republic of Kazakhstan, which goal is decommissioning of the BN-350 reactor, it was decided to move the unused fresh fuel to the Ulba Metallurgical Plant JSC (UMP) in Ust-Kamenogorsk where it was dismantled and down-blended into LEU. In accordance with Agreement on Grant between "Nuclear Threat Initiative" Fund (USA) and "Institute of Non-proliferation" Association (afterward "Non-proliferation Support Centre", dated on February 2, 2002, the schedule of the safe transportation and down-blending of the "fresh" reactor fuel had been elaborated.

In accordance with the developed and approved "Program for transportation of unirradiated fuel from BN-350" transportation of fresh fuel from Aktau to Ust-Kamenogorsk (route RSE "MAEC"-JSC UMP) was implemented in 2002.

In respect to safeguards, the particularity of this event was a necessity to elaborate measures for accounting and control during shipping and receiving. Additional regulations were

developed to provide for integrity of the nuclear materials, use of tamper indicating devices, and check of nuclear material quantity.

2.3. Down-blending of the HEU

The processing of the fuel assemblies from BN-350 at the Ulba facility was the third important action allowing for testing the effectiveness of our system.

The separate area of nuclear material storage facility equipped by surveillance and containment devices was allotted for stocking and temporary storage of the fuel assemblies. The received assemblies were previously sorted by their enrichment level.

The accounting and control at the new storage area is presented at Fig.1.

The new instructions allowing for continuous control over the nuclear material movement were developed for accounting and control of nuclear materials during processing activities. These instructions were presented in the frame of the licensing of this activity.

In order to transfer the fuel elements to the processing facility they had been freed by using fuel assembly cutting installation and sorted. Then fuel elements had been put in special container, loaded in car and transferred to the processing area. Fuel elements have been disassembled and then processed pursuant to technological procedures. Entire process had been implemented under the IAEA safeguards coverage. It was a good lesson to improve accounting system at the facility. The video surveillance system was installed at the processing area by IAEA specialists.

The procedure of accounting and control of nuclear materials during down-blending activities is given on Fig 2. Control of the weight, uranium concentration, and content of the uranium had been carried out at every stage of the process.

During all the aforementioned activities the modern tamper indicating devices (seals) were used and the instructions for devices' use were developed. In our case there were seals. All the seals had individual numbers, the seal flow passing through nuclear facilities was strictly accounted by the records allowing for determination of current place of a seal. Also requirement for destruction of a seal had been established to eliminate a possibility of seal repeated use.

Physical inventory verification of nuclear material had been carried out after each change (scheduled verifications carry out annually no later 12 months after regular PIV). Verifications during processing were executed monthly.

3. Additional Protocol State

Additional protocol to the Agreement between the Republic of Kazakhstan and the International Atomic Energy Agency for the Application of Safeguards in connection with the Treaty on the Non-Proliferation of Nuclear Weapons was signed on February 6, 2004. At present in accordance with the internal rules the Additional Protocol is under ratification process.

Kazakhstan carries out preliminary work to prepare initial declaration. All the facilities submitted the preliminary declarations on the Additional Protocol to government authority.

For that each facility received copy of software designed of IAEA for presentation of preliminary declarations on the Additional Protocol. Now these declarations are analyzed and preliminary Kazakhstani report is prepared.

One of the important elements of the Additional protocol is the implementation of complementary access for IAEA inspectors. This issue entails the necessity to develop additional procedures.

It is of the utmost importance that the trainings shall be carried out for the staff of the facilities, which will be under the provisions of the Additional protocol. Therefore training and methodical assistance will be needed.

4. Conclusion

The Kazakhstan carried out the extensive activities on improvement of the state system for accounting and control of nuclear materials and the work on the implementation of the Additional protocol. And this allows for applying the integrated safeguards in Kazakhstan in the nearest future.

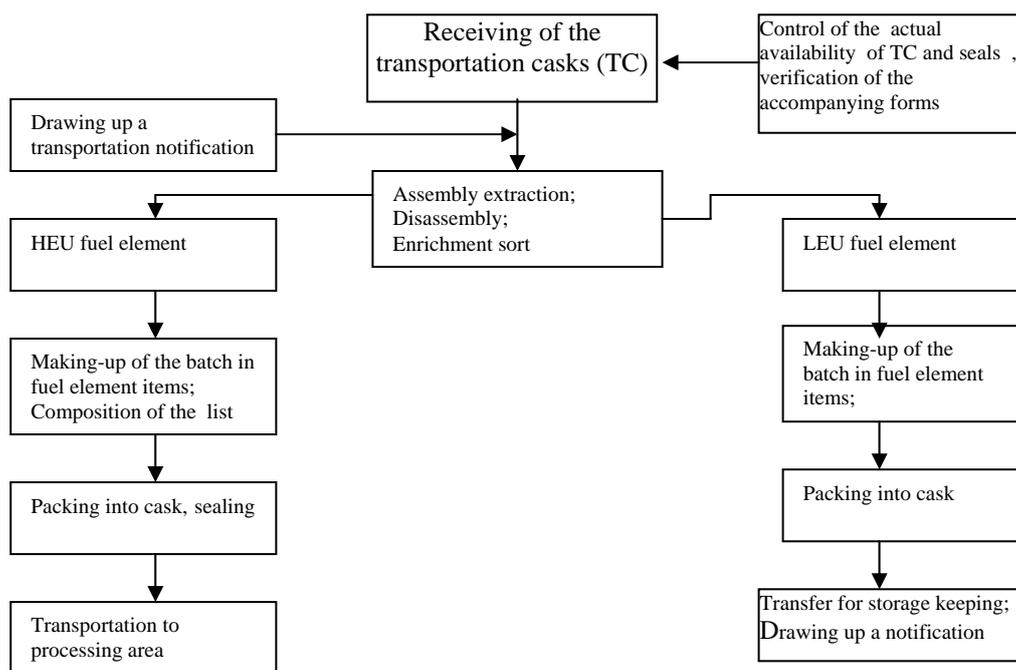


FIG. 1. The accounting and control at the new storage area.

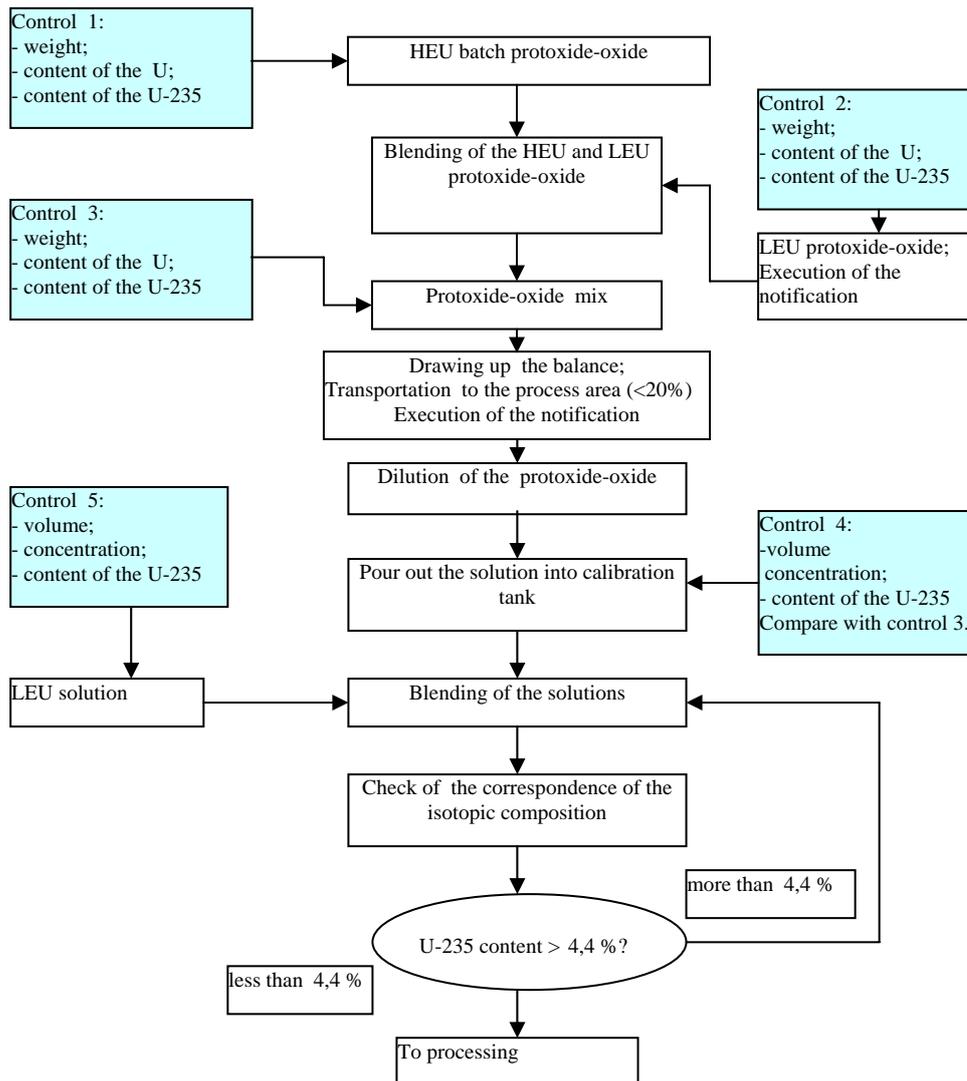


FIG. 2. The procedure of accounting and control of nuclear materials during down-blending activities.

Experience of safeguards implementation at Rokkasho Enrichment Plant (REP) and an innovative safeguards approach for new plant

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Presented by D. Langlands

Abstract. Implementation of safeguards at the Rokkasho Enrichment Plant (REP) commenced in 1992, and is based upon the Hexapartite Safeguards Project (HSP) concept. Since that time, there has been ongoing cooperation between the IAEA, the Japan Safeguards Office (JSGO) and Japan Nuclear Fuel Limited (JNFL) towards continuous improvement and development of procedures and methodology. From a safeguards viewpoint, verification of uranium enrichment and the quantity produced (flow and inventory) and confirmation that there are no undeclared activities are essential. From the viewpoint of process control and safety measures, uranium quantity and enrichment control are essential. Future systems may, as a result of cost constraints, have to be optimized to perform these functions jointly.

This paper describes the experience of safeguards implementation at the REP to date, and the current initiatives for improvement in the UF₆ cylinder movement and transfer verification, for the expanded content of limited frequency unannounced access (LFUA) inspections and for a group sealing approach for large numbers of UF₆ cylinders in static storage. Relative to a new plant, the paper also explains a new safeguards approach that utilizes an installed non-destructive assay (NDA) system, together with a remote monitoring capability.

1. Introduction

The Rokkasho Enrichment Plant (REP) has operated since 1992 under safeguards, based upon the Hexapartite Safeguards Project (HSP) concept. The safeguards goals, defined in the HSP, can be achieved through the use of safeguards measures by the inspectorate that would enable the detection (in a timely manner and with a high degree of confidence) of the diversion of a significant quantity of uranium, including the production of a significant quantity of uranium at an enrichment level higher than that declared. The limited frequency unannounced access (LFUA) to cascade areas has been implemented, and recently expanded to plant areas outside the cascade areas, and environmental sampling has been employed since 1995.

However, further improvements are necessary, especially in areas of continuity of knowledge (CoK) relative to the physical inventory verification (PIV) under the condition of in-operation physical inventory taking and the effective flow verification for UF₆ cylinder movements. Therefore, the Japan Safeguards Office (JSGO), Japan Nuclear Fuel Limited (JNFL) and the IAEA have discussed improvements to the safeguards approach for the existing REP, and also examined possibilities for a new safeguards approach at the new plant yet to be designed and constructed. The main considerations were:

- (a) Ensuring the safety of facility operations and personnel;
- (b) Meeting international safeguards requirements;

- (c) Achieving efficient and economical production operations; and
- (d) Exploring the potential joint use of operator installed equipment.

2. Inspection activities

The IAEA safeguards criteria and facility attachment (FA) for the REP fully incorporate the HSP concept. The current inspection activities at the REP are summarized in Table 1.

Table 1. Summary of inspection activities at REP.

	Inspection Activities	Frequency
Interim Inspection	<ol style="list-style-type: none"> 1. Examination of records and reports. 2. Flow verification by NDA and weighing: <ol style="list-style-type: none"> a) Feed cylinders (before connected to the process); b) Product and tails cylinders (after removed from process, before shipment); and c) Heel cylinder verification. 3. Application of seal and replacement on UF₆ cylinders. 	Currently approximately once every two months (formerly once per month)
PIV	<ol style="list-style-type: none"> 1. Pre-PIV (before ‘switchover’): <ol style="list-style-type: none"> a) NDA and weighing of feed cylinders as process inventory; and b) Examination of operation records related to physical inventory taking (PIT). 2. PIV (at the time of ‘switchover’): <ol style="list-style-type: none"> a) Observation of the switching activity; b) Examination of records and reports; c) NDA and weighing of feed, product, tails and heel cylinders for storage inventory; d) Seal replacement of UF₆ cylinders; and e) Examination of operation records related to PIT. 3. Post-PIV (after ‘switchover’): <ol style="list-style-type: none"> a) NDA and weighing of selected product, tails and purge cylinders as process inventory; and b) Examination of operation records related to PIT. 	Once per year
LFUA	<ol style="list-style-type: none"> 1. Visual verification of cascade area. 2. NDA measurement of cascade header pipes. 3. Taking of environmental samples. 	REP: approximately 13 times per year
Other	<ol style="list-style-type: none"> 1. Flow verification by sample taking: <ol style="list-style-type: none"> a) Feed cylinders (only other than natural U); and b) Product and tails cylinders. 	Approximately 6 times per year

3. Issues for safeguards improvement at the REP

3.1. Insufficient continuity of knowledge for in-operation PIV

A unique point for material accounting of the enrichment plant is the method of physical inventory taking (PIV) performed while the plant is in operation, without the clean out operation by switching of feed and withdrawal lines simultaneously. The switching diagram for the process is shown in Figure 1. Physical inventory is determined by the following procedure:

- (a) Feed cylinders and cold traps in operation are switched simultaneously to standby one at a time.
- (b) Quantities of uranium in standby feed cylinders and cold traps are determined prior to the time of switching. Quantities of uranium in the switched-over feed cylinder and cold traps are determined after the time of switching.
- (c) Quantities of uranium in cylinders are measured before attachment to, and after detachment from, the process. Uranium in cold traps is transferred to extract cylinders and their quantities are measured. Quantities of remaining UF_6 gas in cold traps are calculated by temperature, pressure and volume.

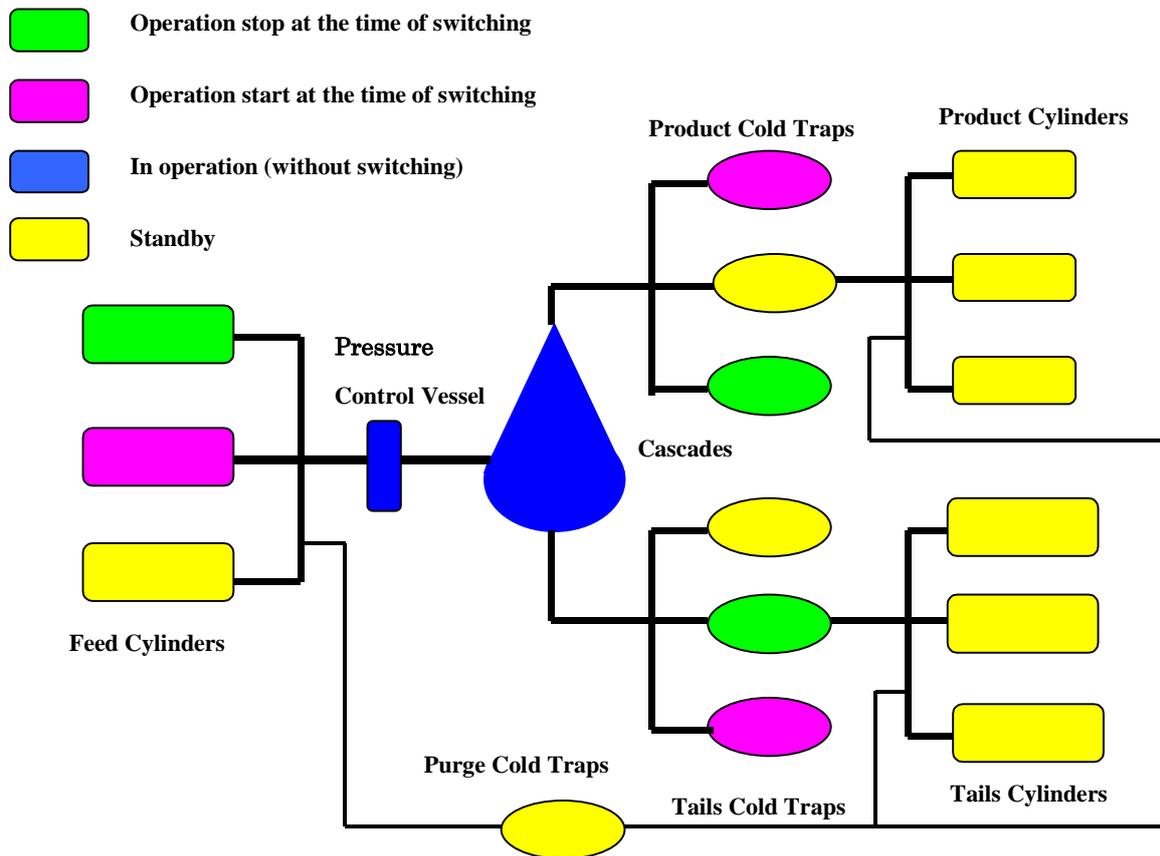


FIG. 1. Switching chart of operation lines [1].

- (d) Quantities of UF₆ gas in the pressure control vessel at the time of switching are calculated by temperature, pressure and volume.
- (e) After detachment of cylinders, samples are taken from product cylinders for enrichment and uranium concentration measurements. Enrichment and uranium concentration of tails cylinders are estimated from the operational date.

In order to verify the physical inventory under the in-operation PIT, a complete PIV is performed consisting of a pre-PIV, a PIV and a post-PIV for process inventory verification, and encompassing a time period of more than one month from the pre-PIV until completion of the post-PIV activities. Therefore maintaining the CoK is quite difficult between the pre-PIV and the PIV, and between the moment of 'switchover' and the time of verification of the cylinders selected, taken off-line and disconnected; this sometimes occurs days after the 'switchover'.

3.2. Effective UF₆ cylinder flow verification (lack of over-production verification)

Flow verification is performed on UF₆ cylinders at the time of the scheduled interim inspection and the PIV, based upon the operator's declaration. A number of UF₆ cylinders, however, remain connected to the process and are inside autoclaves during the interim inspection (and during the PIV). Hence some of the UF₆ cylinders are inaccessible during this period (for safety reasons). Therefore, improvement in flow verification is necessary to increase the inspection confidence level.

4. Improvements to safeguards at the existing REP

JNFL, JSGO and the IAEA discussed the development of the following improvements at an enrichment task force meeting.

4.1. Maintaining continuity of knowledge of material during PIV

A pre-PIV can be eliminated and only randomly selected cylinders need to be 'switched over' for verification against a declaration made at that time -- i.e. not all of the cylinders from the 'previous' material balance period (MBP). Cylinders will be randomly selected and removed from autoclaves for verification, by destructive analysis (DA) and/or non-destructive assay (NDA) and weighing. The material selected, which is contained in de-sublimers (also known as cold traps), has to be transferred from this containment by process pipes to UF₆ cylinders (which are inside autoclaves) for the purpose of verification; both the selected cold trap and the receiving UF₆ cylinders must be verified on completion of the process operation and subsequent disconnection of the cylinder from that process. The cold trap must be verified for emptiness and the receiving cylinder(s) for quantity. CoK on the transfer of selected materials is maintained by the use of temporary containment and surveillance (C/S) on associated closed process valves to eliminate the possibility of substitution or diversion. At the time of random selection of in-process material for verification, valves on interlinking pipes will be sealed in the closed position on all associated pipes, using IAEA paper seals as appropriate.

4.2. Improvement of flow verification

The UF₆ cylinder transfer route between UF₆ storage and process areas can be monitored by suitable equipment as an independent verification for UF₆ cylinder transfers during the absence of the inspector. Other potential transfer routes are verified as being closed to transfer (or provided with a surveillance system).

A C/S system composed of surveillance cameras with motion sensor and a timer-based trigger, together with a cylinder ID verifier system, is shown in Figure 2. This system, less the laser reader, has been installed on a trial basis at the REP. The cylinder ID laser reader is under development. A high definition camera is also under consideration in a backup role.

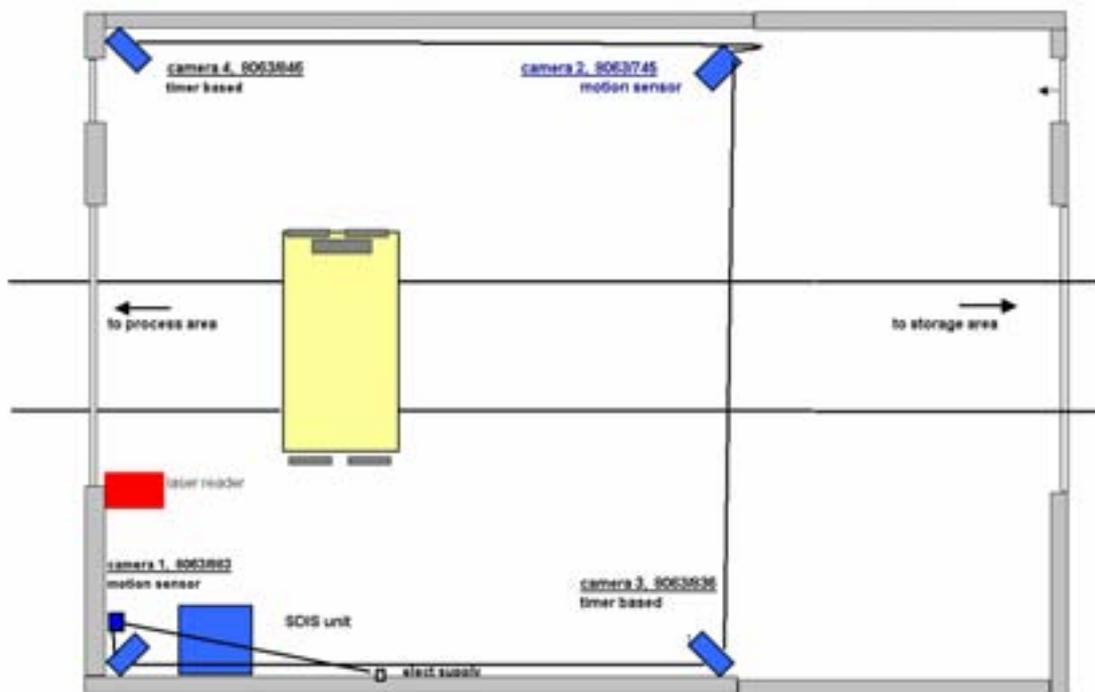


FIG. 2. Surveillance system at REP for UF₆ cylinder transfers.

The inspectorates can verify all UF₆ cylinder transfers between the storage and processes through the surveillance system information and the advance operator's declaration, which gives cylinder ID, date of UF₆ cylinder transfer and type of UF₆ cylinder.

4.3. Further improvements

JNFL, JSGO and the IAEA have discussed further improvements for the safeguarding of the REP, including the following:

- (a) Possibility of short notice random inspection (SNRI) instead of scheduled interim inspection; and
- (b) Use of group seals for stored tails UF₆ cylinders to replace individual seals for cylinder, to reduce the inspection efforts for seals replacement; a trial installation is in place in one cylinder bay.

5. Use of operator equipment for inspection purposes

5.1. Operator equipment available at the REP

From the viewpoint of process control and safety measures, uranium quantity and enrichment control are essential. The plant control and measurement system can be utilized with appropriate system authentication.

Table 2 describes the plant control system for consideration for inspection use by signal splitting.

Table 2. The Plant Control System at REP [2].

Process Control Items	Method	Use for SG
1. UF ₆ weight control in the cylinder	Load-cell at cylinder autoclave	questionable
2. Enrichment control (product)	Inline mass, UF ₆ gas flow meter and pressure gauge	questionable
3. Temperature restrictions	Thermometer	no
4. UF ₆ gas pressure restriction	Pressure gauge	no

5.2. A study of authentication

JNFL has studied authentication measures for the existing plant control equipment, such as load-cell and UF₆ gas flow meter. The findings include:

- (a) Authentication of the load-cell signal is almost impossible due to the load-cell specific electric circuit; tamper-proof measures to the load-cell in the autoclave are almost impossible due to the autoclave structure.
- (b) Authentication of the flow meter is also quite difficult, especially the verification of the orifice diameter which has been installed in the process pipeline. Additionally, a flow meter has no capability to distinguish between UF₆ gas and other gases.
- (c) The number of meters required to cover all routes may be prohibitive in terms of cost, reliability and maintenance.

It is concluded that the use of operator's equipment by signal splitting is not appropriate for inspection purposes.

6. Innovative safeguards approach for a new plant

JNFL has studied a new safeguards approach for the new REP, and discussed an outline of this with JSGO and the IAEA. The IAEA also independently discussed the NDA methodology internally and with Los Alamos National Laboratory (LANL), USA. JNFL again took into consideration the following points:

- (a) Ensuring the safety of facility operations and personnel;
- (b) Achievement of the international safeguards requirements; and
- (c) Efficient and economical production operations.

6.1. System requirements and conceptual design

An installed NDA system is ideal for use both for safeguards implementation and plant control purposes, because the system provides real-time uranium quantity and enrichment information. Being a new installation in a new plant, signal splitting can be engineered under controlled conditions by inspectorates. If the system is commonly used for both purposes, the cost for the systems and installations may be appropriately shared between the operator and the inspectorate.

In order to provide effective and efficient safeguards, the system should have the following capabilities:

- (a) The remote monitoring and verification capabilities for the uranium quantity and enrichment of the whole process; and
- (b) The capability to verify direction of UF_6 cylinder movement connecting or disconnecting by pattern recognition, and to distinguish empty cylinder or otherwise status.

The conceptual designs of the NDA systems are shown in Figure 3 and Figure 4.

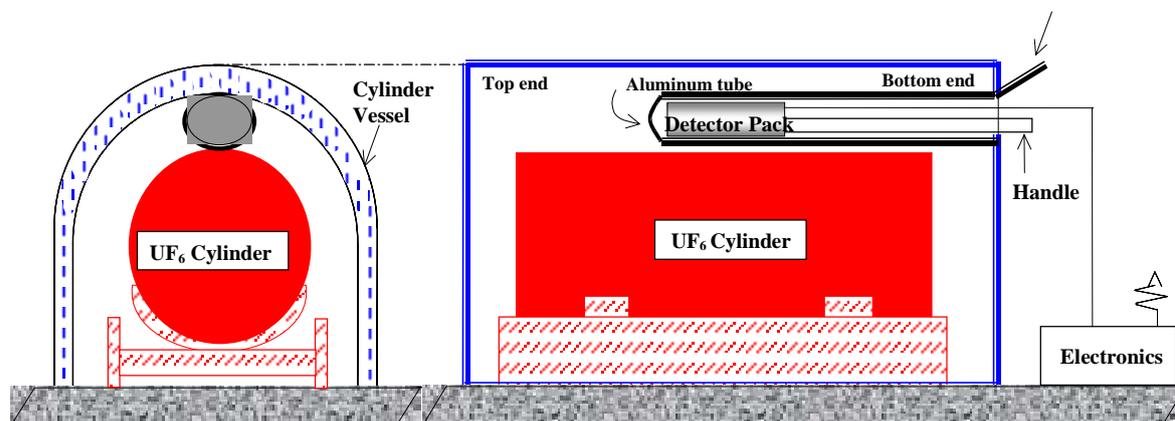


FIG. 3. Conceptual design of UF_6 cylinder autoclave NDA system.

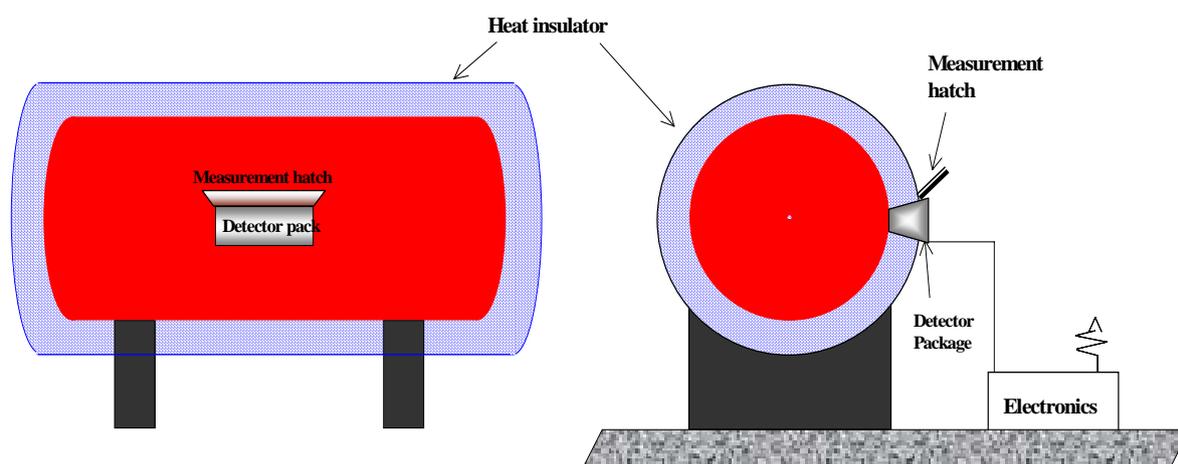


FIG. 4. Conceptual design of cold trap NDA system.

The above NDA system will provide almost the entire process inventory mass and enrichment, and information of connecting or disconnecting of UF₆ cylinders. Together with the C/S system installed between the UF₆ storage and process area, this will identify the mass, enrichment and location of all cylinders. The conceptual design of the safeguards system from the C/S viewpoint is shown in Figure 5.

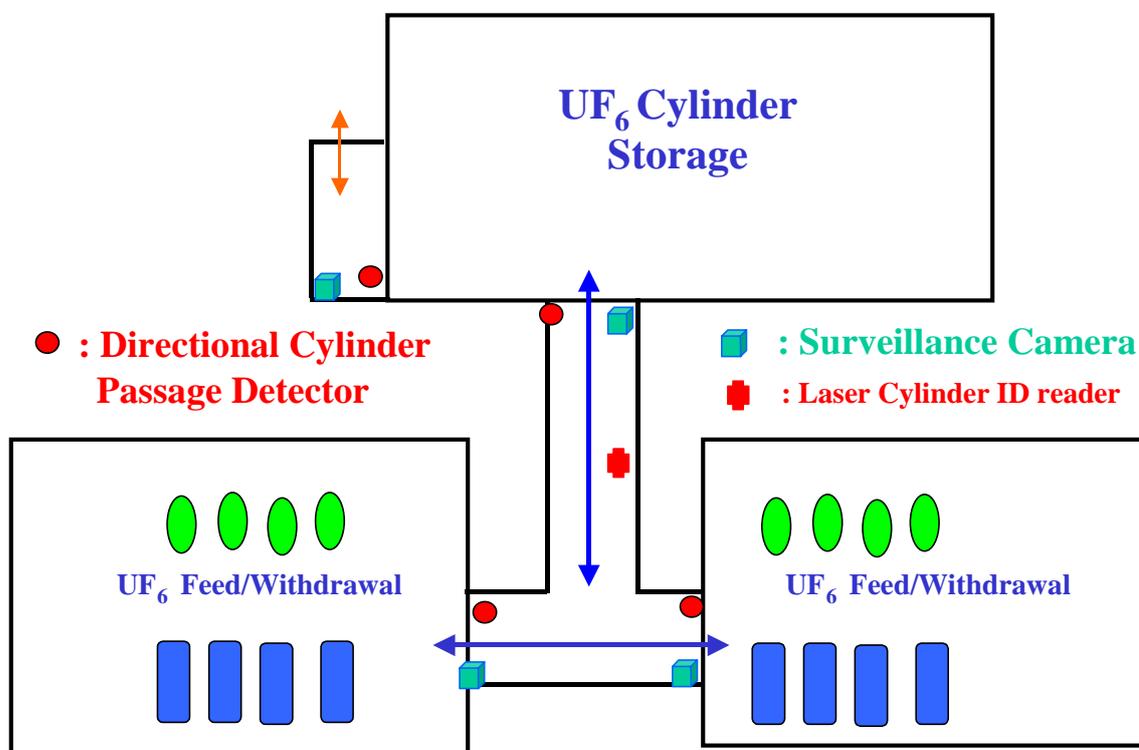


FIG. 5. Conceptual design of the safeguards system.

6.2. Requirement of NDA detectors and options

The cylinder autoclave and cold trap NDA detector would be used under hard temperature conditions:

- (a) For enrichment verification: gamma ray detector;
- (b) For quantity verification: neutron detector; and
- (c) For cylinder transfer direction verification: neutron detector.

Currently available or near-term future options for NDA detectors are shown in Table 3.

Table 3. NDA detector options.

	gamma	neutron
Cylinder Autoclave	NaI	^3He
	HPXe	^3He
	Hybrid HPXe	
Cylinder Passage Detector	/	^3He

HPXe (high-pressure xenon and Hybrid HPXe [3]) would be ideal for the assay, both of which are under development at LANL.

7. Conclusions

Under the above safeguards system the following inspection activities would be performed by remote inspection mode:

- (a) Process inventory verification at each workstation (autoclave etc.);
- (b) Cylinder transfer verification (within facility, including connect or disconnect to/from autoclave); and
- (c) Verification of feed and production mass with enrichment.

The following inspection activities would be required at the facility:

- (a) PIV for verification of storage items, and process inventory including chemical traps (other than UF_6 in autoclaves and cold traps);
- (b) LFUA;
- (c) Maintenance and normalization of the NDA system; and
- (d) UF_6 cylinder seal replacement.

The new safeguards approach should provide the following advantages:

- (a) Enhanced effectiveness and efficiency of safeguards.
- (b) Frequency of LFUA could be reduced.
- (c) DA sampling from UF_6 cylinders (feed, product and tails) could be eliminated, because the remote verification scheme would cover all cylinder transfers and provide real-time process inventory with enrichment. Therefore any unusual operation could be detected, thereby eliminating the possibility of protracted diversion.
- (d) Inspection effort could be reduced to about 50 person days of inspection.

8. Further studies

The following topics should be considered for further study:

- (a) Evaluation/examination of neutron ‘cross talk’ effects from neighbouring autoclaves and cold traps, and a means to confirm non-presence of ‘introduced’ sources; and
- (b) Further development of HPXe hybrid and HPXe.

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JNFL MOX fuel fabrication plant (J-MOX): Plant overview and safeguards considerations

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Abstract. Construction of J-MOX has been treated as an official project since April 2005 when licensing of J-MOX was submitted to the Government of Japan. Then, formalities and discussion with IAEA have also started. For better understanding of newly planned J-MOX, site plan, construction schedule, feed & product, operating conditions and process description are presented. J-MOX is now being designed as three MBAs structure and material accountancy method is planned based upon specific features of J-MOX. In arguing safeguards systems with IAEA, J-MOX plant characteristics should be considered with paying attention to RRP safeguards approach and current international and domestic safeguards environments.

1. Introduction

In April 2005, Japan Nuclear Fuel Limited (JNFL) submitted the Government of Japan an application for the permission of MOX fuel fabrication business for JNFL MOX Fuel Fabrication Plant (J-MOX). This implies that construction of J-MOX can be treated as an official project in Japan. Therefore, J-MOX safeguards formalities and discussion with the Agency have also started.

By now, Working Group Meetings and Design Information Examination (DIE) Sessions have been held for J-MOX among IAEA, Japan Safeguards Office (JSGO) and JNFL. JNFL explained details of J-MOX process design and proposed material accountancy method through the meetings and sessions so that IAEA can design practical safeguards systems.

In this paper, safeguards relevant events, plant overview and material accountancy are presented for better understanding of J-MOX. And some safeguards considerations, which are derived from specific features of J-MOX and environments surrounding J-MOX, are also presented for indicating the direction of future safeguards argument for J-MOX with IAEA.

2. Safeguards relevant events

In April 2005, JNFL submitted the Government of Japan an application for the permission of MOX fuel fabrication business for J-MOX. Since then, safeguards formalities and discussion with IAEA and JSGO have also started as listed below;

2005	May	Preliminary design information of J-MOX was sent to IAEA
	June	J-MOX plant overview was presented to IAEA at Plenary Meeting and establishment of Working Group Meeting was agreed
	June	1st Working Group Meeting

	August - September	1st DIE Session and Technical Meeting
	November	2nd DIE Session and Technical Meeting
2006	April	3rd DIE Session and Technical Meeting
	June	2nd Working Group Meeting and Technical Meeting
	July	4th DIE Session

Note) This information is as of August 2006.

3. Plant overview

3.1 Site plan

J-MOX is composed of three buildings, Fuel Fabrication Building, Utilities Building and Administration Building. They are built just adjacent to Rokkasho Reprocessing Plant (RRP). Fuel Fabrication Building has site area of approx. 80m x 80m. There are three floors under the ground and two floors on the ground. Fuel Fabrication Building is connected with RRP by an underground trench to transport MOX powder reprocessed in RRP to J-MOX from RRP. Bird's-eye view of planned J-MOX site is shown in FIG. 1.



FIG. 1. J-MOX site.

3.2 Construction schedule

The current J-MOX construction schedule is shown in FIG. 2.

As inspector's systems used for safeguards purpose should be developed, installed and tested in line with the J-MOX construction schedule, JNFL, JSGO and IAEA agreed some milestones toward completion of the inspector's systems. The first milestone is the date when JSGO, JNFL and IAEA are to agree rough equipment specifications and installation place of the inspector's systems.

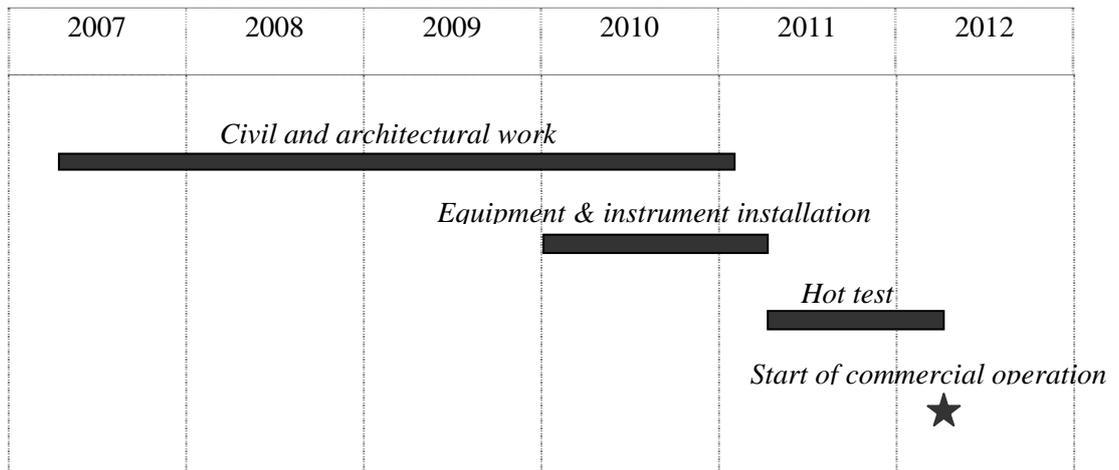


FIG. 2. J-MOX construction schedule.

3.3 Feed and product

Feed materials to J-MOX are MOX powder from RRP, UO₂ powder from a re-conversion facility and UO₂ fuel rods from LEU fuel fabrication facilities. The feed MOX powder is co-denitrated in RRP by using micro-wave heating technology, containing 50% plutonium and 50% low enrichment uranium. The feed UO₂ powder is depleted uranium and used for dilution of MOX powder to adjust Pu content¹ to required fuel specifications. The feed UO₂ fuel rods are assembled with MOX fuel rods for BWR.

Product material is MOX fuel assembly for LWR (PWR and BWR). Each MOX fuel assembly consists of MOX fuel rods with different Pu contents (and U fuel rods in case of BWR MOX fuel assembly) and each MOX fuel rod consists of MOX pellets all of which are the same Pu content.

3.4 Operating conditions

Maximum design capacity is 130 t-HM/Y. J-MOX is to be operated within this capacity on condition that all amount of plutonium annually reprocessed in RRP are consumed through a series of production campaign in J-MOX.

Basically normal operation mode is two shifts per day and five operating days per week.

3.5 Process description [1]

J-MOX process is composed of Feed Powder Receiving Process, Powder Preparation Process, Pellet Fabrication Process, Fuel Rod Fabrication Process, Fuel Assembly Fabrication Process, Packing & Shipping Process, Scrap Treatment Line, Analytical Laboratory and Solid Waste Storage as shown in FIG. 3.

(1) Feed Powder Receiving Process

A MOX canister that contains three MOX powder cans filled with feed MOX powder is transferred from U/Pu Mixed Oxide Storage Building of RRP and temporarily stored. UO₂ powder cans are transported from a re-conversion facility and stored.

(2) Powder Preparation Process

Necessary amounts of the feed MOX powder and the feed UO₂ powder are dosed by weighing and blended through two dilution steps to meet the Pu contents required in MOX fuel specifications. The MIMAS (MICronized MASTer Blending) process developed by Belgonucleaire and adopted in MELOX plant is used in this process. Final Pu contents of the blended MOX powder are in the range

¹ Pu content = Pu/(Pu+U)

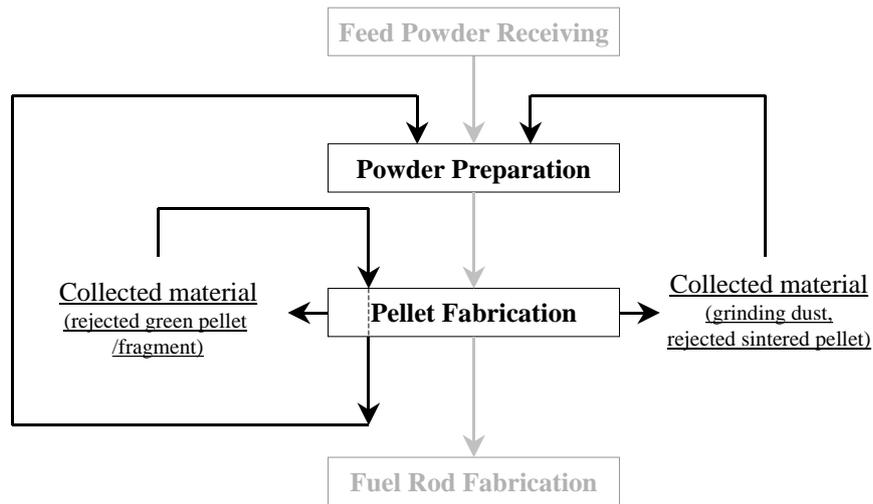


FIG. 4. Recycle flow of collected materials.

(6) Fuel Rod Fabrication Process

Product pellets are inserted into a cladding tube and the rod is welded with an end plug.

(7) Fuel Assembly Fabrication Process

Fuel rods are bundled with grid-frames/spacer and other parts to form a MOX fuel assembly.

(8) Packing and Shipping Process

The MOX fuel assembly is packed into a transportation container and shipped out by track.

(9) Analytical Laboratory

Samples taken from Powder Preparation Process, Pellet Fabrication Process and Fuel Rod Fabrication Process are analyzed for the purpose of process control, criticality control, quality control and material accountability.

(10) Solid Waste Storage

Solid wastes generated from each process are loaded into a container (carton box and drum etc.). Dose rate and contamination level of the container are measured. Then, they are repacked into a crate, stored and transferred to RRP.

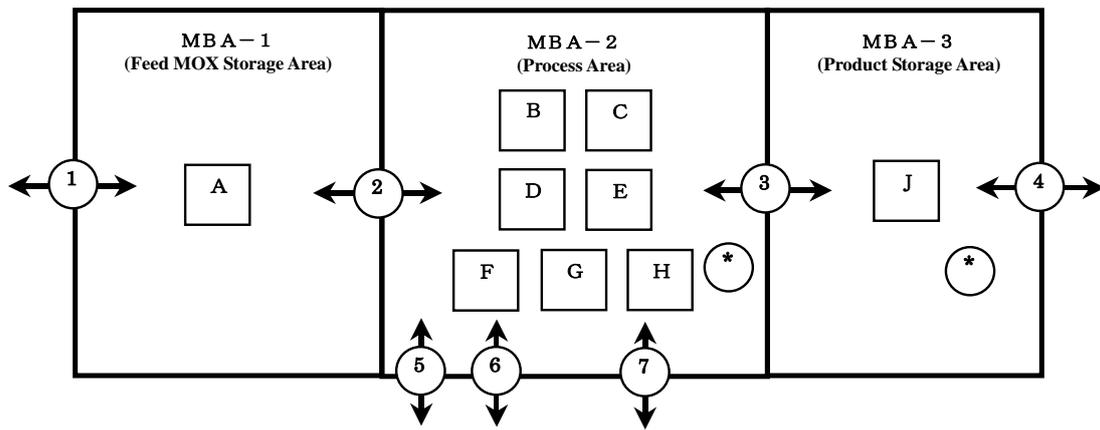
4. Material accountability

4.1 MBA/KMP structure

JNFL is now designing J-MOX as three MBAs structure for the purpose of material accountability as shown in FIG. 5. This is current JNFL proposal that needs to be discussed with IAEA.

4.2 Material accountability principles

Basic principles of J-MOX material accountability are as follows;



Flow Key Measurement Point (FKMP)

- 1 : Receipt and Shipment of Feed MOX Powder
- 2 : Transfer of Feed MOX Powder between MBA-1 and MBA-2
- 3 : Transfer of MOX Fuel Assembly between MBA-2 and MBA-3
- 4 : Shipment and Receipt of MOX Fuel Assembly
- 5 : Receipt and Shipment of UO₂ Powder, LEU Fuel Rod, U Alloy
- 6 : Shipment and Receipt of Solid Waste, Sample, Small Quantity of Nuclear Material
- 7 : Transfer to Retained Waste, Retransfer from Retained Waste
- * : Nuclear Loss, Category Change

Inventory Key Measurement Point (IKMP)

- A : MOX Canister Temporary Storage Area
- B : UO₂ Powder Storage Area
- C : Powder Preparation Process
- D : Scrap Treatment and Test Line
- E : Pellet Fabrication Process
- F : Fuel Rod Fabrication Process
- G : Analytical Laboratory
- H : Solid Waste Storage Area
- J : Fuel Assembly Storage Area

FIG. 5. MBA/KMP structure of J-MOX.

- a. Shipper/receiver difference (SRD) of feed MOX powder is not evaluated between RRP and J-MOX. And SRD of feed UO₂ powder is also not evaluated because shipper's data is checked by JNFL independently.
- b. All fabrication processes are divided into elemental material accounting units, for example one glovebox or one storage pit, in which nuclear material of inventory and flow is accounted for.
- c. In the most cases, amount of nuclear material is obtained from weight measurement and estimated concentration based upon shipper's accountancy data (RRP data for MOX powder and re-conversion facility's data for UO₂ powder) from Feed Powder Receiving to Pellet Fabrication Process.
- d. Destructive assay sampling points for the purpose of material accountancy will be limited to product pellet after grinding, recyclable scrap after treatment and collected materials after treatment.
- e. GUAM (Glovebox Unattended Assay and Monitoring system) will be installed to measure Pu amount of inventory (so-called "holdup") inside glovebox.
- f. PSMC (Plutonium Scrap Multiplicity Counter) will be installed to measure Pu amount of recyclable scrap.
- g. WPAS (Waste Package Assay System) will be installed to measure Pu amount of solid waste.
- h. Centralized data management computer system adopts an on-line network over the plant and collects data for process control, material accountancy, criticality control and quality control. Therefore, operator can easily access to daily inventory and flow data.

5. Safeguards considerations

In discussing J-MOX safeguards systems with IAEA, requirements resulting from the following aspects should be considered;

5.1 J-MOX plant characteristics

(1) Large throughput

Annual Pu throughput of J-MOX will reach to about 7t. The large throughput will cause some difficulties in terms of material accountancy and safeguards that existing small throughput facilities have not faced so far.

(2) Large capacity of storage areas distributed over the plant

In order to keep flexibility of operation and necessary buffer volume between processes, several storage areas for MOX powder, pellet, scrap, fuel rod and fuel assembly are placed over the plant. These storage areas have a large capacity and there might be quite a few items at Interim Inventory Verification (IIV) and/or Physical Inventory Verification (PIV). Considering operator and inspector resources required for the verification, appropriate Containment and Surveillance (C/S) systems should be installed to reduce re-measurement efforts as much as possible.

(3) Automated process operation

J-MOX fabrication process will be fully automated and remotely operated. Therefore, when some inspector's systems are installed in J-MOX, the systems should be unattended and automated ones. This is also a requirement in terms of reducing radiation exposure to the operators and inspectors.

(4) Handling MOX powder inside glovebox

All equipment that handles MOX powder is installed inside glovebox. A possibility of having a holdup inside the equipment and/or glovebox cannot be ignored even though the equipment is machined so as not to keep nuclear materials inside nor to spread nuclear material outside. Therefore, a fixed system that can measure inventory inside a glovebox with reasonable errors should be considered from an operator's material accountancy point of view and an inspector's verification point of view.

5.2 Expanding RRP safeguards approaches

(1) Similar systems to similar strata

J-MOX is built just adjacent to RRP and connected with RRP by an underground trench. And same strata such as MOX powder are handled in both RRP and J-MOX. Therefore, some of safeguards systems applied to RRP will be expanded to J-MOX. For example, iPCAS might be used for verification of MOX canister which is transferred to J-MOX from RRP. Similar concepts of process parameter monitoring system such as PIMS (Plutonium Inventory Measurement System) that the inspectorates are using for monitoring of MOX powder movement in RRP might be considered for Powder Preparation Process.

(2) Utilizing OSL (On-Site Laboratory in RRP)

Basic agreement was made among JSGO, JNFL and IAEA that OSL should be utilized for analysis of inspector's samples taken from J-MOX. Technical issues regarding transportation and handling procedure of inspector's sample should be discussed and clarified.

5.3 International and domestic safeguards environment

(1) Increasing transparency of operation

Increasing transparency of operation is getting more and more important these days for a large scale plant which processes nuclear material in a bulk form. Therefore, such measures and systems that can accomplish the function should be developed for J-MOX.

(2) Safeguards experiences gained at an existing MOX fuel fabrication plant in Japan

Safeguards measures and systems that have been already demonstrated and proven at an existing MOX fuel fabrication plant in Japan should be utilized to the extent possible and practical.

(3) Integrated safeguards approach and measures

Implementation of integrated safeguards was approved by IAEA in June 2004 and safeguards regime in Japan has been changed stepwise since September 2005. IAEA and JSFO are now trying to establish integrated safeguards approaches for facilities that handle un-irradiated direct-use material (e.g. fresh MOX and HEU). Therefore, J-MOX will also be subject to the integrated safeguards and a trend of development and study of the integrated safeguards approaches should be carefully watched for J-MOX application.

6. Future view

JNFL is now concentrating on making IAEA understand J-MOX design so that IAEA can establish safeguards approach of J-MOX as soon as possible. DIE Session was held several times by now and it is the time to start safeguards approach discussion among JSFO, JNFL and IAEA. Seeking for practical and non-intrusive safeguards systems in line with J-MOX construction, JNFL is ready to keep close and co-operative relationship with IAEA.

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Tools and techniques for safeguards training and knowledge management

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Presented by N. Herber and D. Martin

Abstract. This paper identifies some of the challenges faced by the International Atomic Energy Agency in preserving and transferring knowledge in the safeguards field. The IAEA Section for Safeguards Training relies heavily on experienced safeguards inspectors from the Operations Divisions and on outside experts to deliver training modules. However, with a smaller number of experienced inspectors and an increasing workload, the availability of inspectors to undertake training-related assignments is proportionally reduced. Due to their rigorous work and travel schedules, new inspectors often find it difficult to attend traditional instructor-led, classroom- and laboratory-based training. To assist the IAEA to address these challenges, the Canadian Safeguards Support Program has adopted a set of innovative tools and techniques geared to the topics of training, knowledge management, and exchange of information.

1. Introduction

The Canadian Nuclear Safety Commission (CNSC), an independent agency of the Government of Canada, regulates the use of nuclear energy and materials in Canada, and ensures that Canada's international commitments on the peaceful use of nuclear energy are respected. An important dimension of its work is support for the International Atomic Energy Agency's (IAEA's) program of international safeguards. This work is carried out through the Canadian Safeguards Support Program (CSSP).

This paper identifies some of the challenges faced by the IAEA in preserving and transferring knowledge in the safeguards field. It describes some of the CSSP activities geared to the topics of training, knowledge management, and exchange of information.

2. Training Challenges for the IAEA

In a paper presented in 2001 on training challenges facing the IAEA [1], Colin Carroll and Mike Stein of Sonalysts Inc. and Thomas Killeen of the IAEA's Section for Safeguards Training, explored the training implications of changes in the job requirements for IAEA safeguards inspectors, which are taking place at the same time the Agency is experiencing high personnel turnover among inspectors.

As part of its ongoing training methodology, the IAEA Section for Safeguards Training (TTR) continuously monitors and evaluates the training implications of changes in the job requirements for IAEA safeguards inspectors. The introduction of the Additional Protocol and new safeguards devices and technologies has significantly increased the amount of training that inspectors now need. Coincidentally, many of the most experienced inspectors have reached retirement age, or the number of years set by the rotation policy, resulting in a high turnover of personnel. Their replacements include individuals with a wide variety of work experiences and educational backgrounds, and from a broad range of nationalities representing the diversity of the IAEA's Member States.

TTR relies heavily on experienced safeguards inspectors from the Operations Divisions and on outside experts sponsored by Member State Support Programs to deliver training modules, both at IAEA Headquarters in Vienna and at nuclear facilities in Member States. However, with a smaller number of experienced inspectors and an increasing workload, the availability of inspectors to undertake training-related assignments is proportionally reduced. As a result, the availability of qualified instructors has been decreasing at a time when the need for training of entry-level inspectors is increasing. To further complicate matters, reliance on traditional instructor-led, classroom- and laboratory-based training often makes participation in formal training courses difficult for inspectors because of their rigorous work and travel schedules.

In summary, the major training challenges faced by the IAEA are:

- a steep learning curve for new inspectors who have much less experience in the nuclear industry than their predecessors
- pressure to reduce training time, while maintaining training effectiveness
- a loss of many experienced inspectors (a major source of qualified instructors and a resource for "corporate memory") due to retirement and rotation policy

To assist the IAEA to address these challenges, the CSSP has adopted a strategy to create solutions that use the following tools and techniques:

- courses based on sound instructional design principles
- highly-visual presentations featuring multimedia elements
- self-contained, instructor-independent training kits
- simple, just-in-time, "guerilla" techniques for knowledge capture and transfer

This paper describes these tools and techniques, gives some real-world examples, and explains how these tools and techniques have helped to meet the challenges faced by the IAEA.

3. Instructional Design Principles

There are two main principles used in our instructional design. The first is to prepare a full, professional-level lesson plan which specifies all the teaching points, the timing of the presentation,

and indications of which slides or support materials are to be used with a particular teaching point (FIG. 1).

VIFM Workshop: Master Lesson Plan Module 1: CANDU Nuclear Generating Station (NGS) Overview		Revised April 25, 2005	Page 9
<p>16.</p> 	<ul style="list-style-type: none"> The fuelling machine rotating magazine has 12 positions (Darlington has 11). This set of photos show a CANDU 6 fuelling machine partially disassembled for repairs. This is a separator-method fuelling machine. 	0:31	
<p>17.</p> 	<p>CANDU 600 Fuelling Scheme</p> <ul style="list-style-type: none"> In a CANDU 600 reactor, the fuelling machine travels horizontally to transport fresh fuel from the loading area to the reactor face and to take spent fuel back to the spent fuel area. The fuel loading area and the spent fuel bay are separated from the reactor core by large shielding doors The shielding doors may be open or closed during fuelling. 	0:32	
<p>18.</p> 	<ul style="list-style-type: none"> The spent fuel port is connected via elevators and conveyors to the spent fuel bay Defective fuel is canned rather than being sent directly to the spent fuel bay. The spent fuel storage bay is outside the reactor building air lock Spent fuel is kept underwater. 	0:35	

FIG. 1. The VIFM Workshop master lesson plan shows the support material in the first column and the teaching points in the second column. The third column gives suggested timings for each topic.

The second instructional design principle addresses the requirement for follow-up and reference materials after a classroom course is completed. This is done by giving participants a self-study CD-ROM containing course material re-packaged and re-designed for easy access on their own laptop or desktop computers (FIG. 2). In some cases, reviewing a self-study CD-ROM is a prerequisite for incoming students. This ensures all students are starting from a level playing field and allows the instructor to quickly review or completely skip introductory-level material.



FIG. 2. Reference CD-ROMs allow course participants to preview or review the material they covered in class. The CDs can also serve as a “refresher” for inspectors in the field.

4. Highly Visual Presentations

The tools used to create highly visual presentations (in PowerPoint, on CD/DVD-ROMs, and through intranet or extranet web delivery) include:

- video
- graphics
- computer-generated animations
- interactive teaching software

These tools are used in various combinations to create presentations that support instructors in the classroom, and to build CD/DVD-ROM-based and web-based instructional modules that support self-study and reference. These tools produce a more vivid learning experience and in so doing enhance the rate of assimilation of knowledge and its retention.

Where video is difficult to obtain (such as in the reactor vault of an operating nuclear generating station), highly realistic computer-generated animations can show inspectors details that they would otherwise be unable to see (FIG. 3). If these animations are made interactive, they allow the inspector to “operate” the equipment in question and see exactly what effects different operations may have.

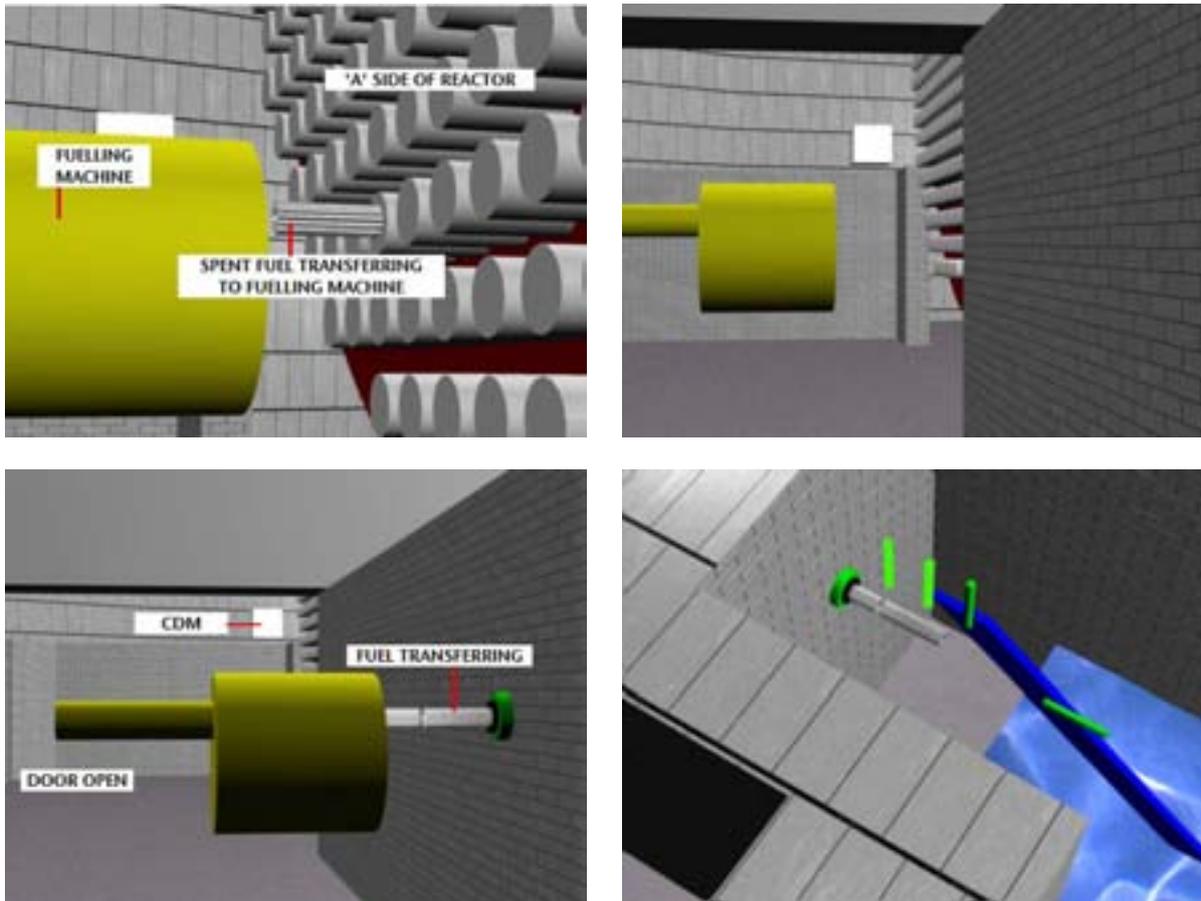


FIG. 3. This highly realistic computer-generated animation allows inspectors to look inside an operating reactor from the comfort and safety of a classroom or their office. These four frames from the animation shows the path followed by spent fuel removed from a CANDU reactor.

5. Self-Contained Training Kits

The CSSP has sponsored the production of self-contained training kits that are prepared by a team of highly skilled professionals: instructional designers, subject matter experts, multi-media and web designers. These kits ensure consistent quality of the training material, reduce instructor's preparation time, and serve as a repository for the corporate knowledge. Figure 4 shows such a kit. This kit contains:

- an instruction guide (with tips on giving better presentations)
- professional-level lesson plans for classroom delivery and for field exercises
- participant hand-outs
- evaluation sheets
- an instructor's CD-ROM containing the multimedia presentations ("PowerPoint on steroids") and all course materials in electronic form
- a participants' CD-ROM containing all relevant course content



FIG. 4. A self-contained training kit created for the new Digital Cerenkov Viewing Device course.

6. Guerilla Knowledge Capture

Knowledge management systems must deal with at least three classes of knowledge: task-based, job-based, and philosophy-based knowledge. Several papers presented at INMM conferences [2][3][4] have detailed the corporate framework and methodologies to implement knowledge management systems. However, implementing such systems requires a finite amount of time, and during that ramp-up time important assets can be lost. Employees with key skills and knowledge may leave the organization, retire, or transfer to other groups that give them little opportunity to participate in training activities.

Adopting the philosophy that something is better than nothing, guerilla knowledge gatherers use technology and techniques that can harvest information before it escapes the corporate domain. Although the harvest may not be perfect, it is information-rich and can provide otherwise unobtainable source material for later integration with corporate knowledge assets. These techniques were first described in papers presented at INMM conferences [5][6].

One technique the CSSP has used for guerilla knowledge capture is videotaping presentations and lectures given by soon-to-be-departing experienced staff. The material is captured using simple home equipment such as digital video cameras and the tapes are stored for future reference or “mined” immediately for nuggets of knowledge. Similar video capture can be carried out to gather training information from technical experts whenever there is an opportunity.

A web-based solution called a “wiki” is used to facilitate knowledge exchange. This tool is a web site that can be edited by any authorized user. With careful design, its ease-of-use promotes rapid exchange of information for preparation of joint papers and training material for a course, or facilitates discussion after a technical workshop. For example, a wiki has successfully been set up as a follow-on communication tool for the n-VISION project of SGIT (FIG. 5).

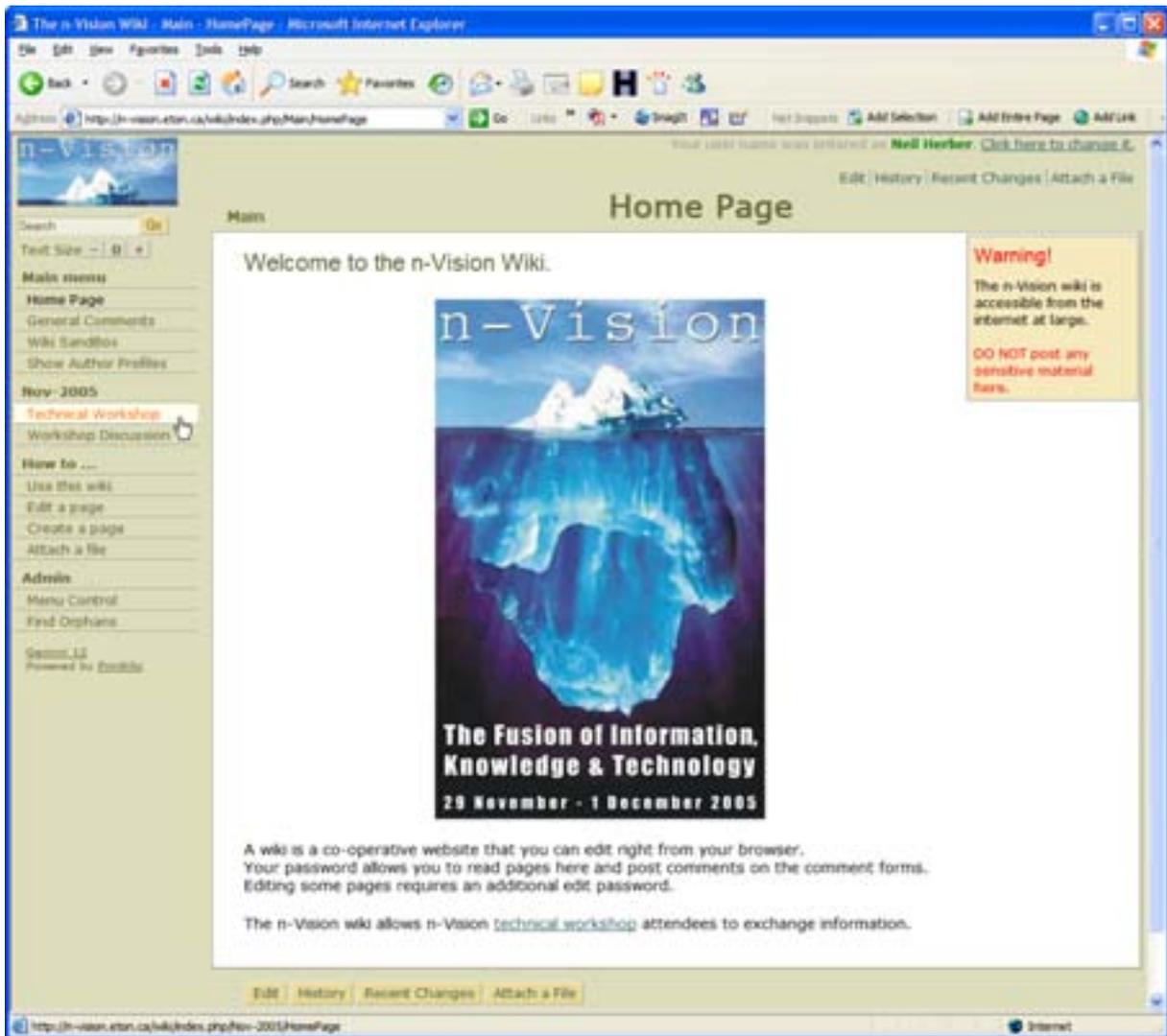


FIG. 5. A web-based wiki facilitates knowledge sharing amongst expert groups. This example is used as a follow-up forum for the n-Vision meeting held at the IAEA in February 2006.

The CSSP and IAEA have also used wikis as a means for course developers to collaborate remotely. Authors, animators, subject matter experts, and IAEA training staff can instantly share files and update web pages to track course development progress. Because the wiki resides on a password-protected server on the internet, team members can keep up to date from any location that has web access, providing more timely contributions and a “living snapshot” of the state of the instructional material.

7. Some Examples

Using these multi-media tools and techniques, the CSSP has developed several successful training programs for the Department of Safeguards on topics such as:

- CANDU Power Station Fundamentals
- VXI Integrated Fuel Monitor (VIFM) Inspection Procedures
- Interpreting Bundle Counter Radiation Signatures

- Interpreting Core Discharge Monitor (CDM) Radiation Signatures

In the self-study training module on VIFM Bundle Counters, an animation, controlled by the user, illustrates fuel movement. The use of animation allows inspectors to see inside parts of the nuclear generating station that are normally inaccessible during operation. The interactivity of the VCR-like controls engages the user and facilitates learning and retention of knowledge. In fact, when this animation was introduced, it reduced the classroom time spent describing bundle counter operation by at least an hour (out of a 3-hour module) for a net savings of 33% in classroom time. The students also had a better grasp of the operation when they operated the animation themselves.

8. Conclusion

Inspector training is acknowledged to be a key element in the effectiveness of IAEA operations. The strategies articulated in this paper show promise as a means to ensure a high level of inspection knowledge and skills in all IAEA inspection activities. They provide a way to address IAEA training challenges arising out of high personnel turnover (both training staff and inspectors), changes in inspector's job requirements, and the complex multi-national characteristic of the inspector population.

The CSSP strategy for supplying assistance to the IAEA for training and knowledge management has resulted in shorter learning times, more flexibility in learning (since inspectors can schedule self-study sessions), and effective capture of knowledge and expertise. The strategy will be applied when responding to future requests for assistance from the IAEA.

Various examples of computer-based training packages including simulations of radiation monitoring equipment will be demonstrated at the poster session.

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Safeguards system in the Republic of Belarus

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Abstract. At the time of signing of the Treaty on the Non-Proliferation of Nuclear Weapons (1993), the Republic of Belarus did not have any state system of nuclear material accountancy and control. Accounting for and control were performed by departmental systems which did not involve computer data processing, generation of reports, independent measurements, data coding, etc. They were not able to perform accounting for and control in real time. In this paper the existing system of nuclear material accounting for and control is briefly discussed as well as system of physical protection of nuclear materials, export/import measures and measures to combat illicit trafficking.

1. Introduction

The Republic of Belarus has been a Contracting Party to the Treaty on the Non-Proliferation of Nuclear Weapons since 1993 and in 1995 signed Agreement between the Republic of Belarus and the International Atomic Energy Agency for the application of safeguards. Through Regulatory Resolution *On Measures to Fulfil Provisions of the Treaty on the Non-Proliferation of Nuclear Weapons* issued by the Council of Ministers of the Republic of Belarus in 1993, Department for Supervision of Industrial and Nuclear Safety (Promatomnadzor) of the Ministry for Emergencies was designated as the national competent authority in the field of nuclear and radiation safety, responsible for the establishment and maintenance of the State System of Accounting for and Control of Nuclear Materials (SSAC).

2. System of accounting for and control of nuclear materials

Measures of accounting for and control are applied to all categories of nuclear materials that are located on the territory of the Republic of Belarus and used in peaceful nuclear activities. They are applied starting from the moment that Belarus takes on the responsibility for an imported material.

PROMATOMNADZOR is responsible for:

- fulfilment of the Agreement with the IAEA on the application of safeguards;
- establishment and implementation of the state system of accounting for and control of nuclear materials, including reporting safeguards information to the IAEA;
- establishment of requirements for physical protection of nuclear materials;
- control of domestic movements and transit of nuclear materials;
- licensing and state supervision of activities involving use of nuclear materials;
- issuing permits on movement of nuclear materials across the borders of Belarus.

Promatomnadzor is also a central contact point in case of a loss, unauthorised use or removal of nuclear materials and carry out expert function in the system of export-import control.

In the Republic of Belarus there are two material balance areas (BY-A), “Sosny” Academic Scientific and Technical Centre and “Ekores” (BY-E), the state-owned enterprise. These MBA’s have two critical assemblies, a “dry” storage of nuclear materials, a “wet” storage of spent nuclear fuel, research laboratories, and a neutron generator with subcritical assembly. Material accountancy and control conducted at the installations are based on their design features and the character of material used.

The installations make use of the following types of nuclear material:

- high enriched uranium over 20% enrichment by U-235;
- enriched uranium up to 20% enrichment by U-235;
- natural and depleted uranium and their compounds;
- small quantities of material representing parts or mixtures of the above-mentioned materials, including scrap;
- small quantities of plutonium and thorium.

The system accounts for all the nuclear material meeting the criteria defined in the Safeguards Agreement and includes 2 levels: on-site accounting and control and state accounting and control exercised by Promatomnadzor. The SSAC is based on the following documents:

- Structure of the state system of accounting for and control of nuclear materials in the Republic of Belarus;
- Procedures for exercising supervision of accounting for and control of nuclear materials in the Republic of Belarus;
- Provisional rules governing operation of the state system of accounting for and control of nuclear materials in the Republic of Belarus;
- Requirements on the organisation and conduction of accounting for and control of nuclear materials in the process of their use, storage and transportation in storage facilities, research and experimental reactors, critical and subcritical assemblies, research laboratories and research installations;
- Requirements on the submission of accounting documentation to the national supervisory authority;
- Regulations for implementation of the IAEA Safeguards.

These documents define basic rules and procedures concerning the conduct of accounting for and control of nuclear materials as well as preparation and submission of reports to the state supervisory authority and the IAEA.

2.1. Organisation and functioning of the system of accounting for and control

Responsibility for the system of accounting and control in MBA’s lies with enterprises holding permits (licenses) for carrying out activities involving use, storage, transportation of fissionable nuclear materials, i.e. the operators.

According to the national regulations and standards, a system of accounting for and control set up at the facility level shall report to the state on how it exercises an internal control over uses, quantities and flows of nuclear material available at installations. Facilities must have material accountancy staff responsible for: keeping accounting/control documents; keeping records of shipped/received materials; recording changes in quantities arising from material burnout and reproduction; control of material transfers between KMP’s within an MBA;

physical inventory checks within an MBA followed by balance-striking and determination of material unaccounted for (MUF); analysis of accounting data and reporting them to the supervisory authority with the observance of prescribed time limits and forms.

In conformity with the above requirements, the system of accounting and control at the facility level is characterised by the following:

- personal responsibility is introduced for the organisation and functioning of the system at the facility;
- structure of MBA's, KMP's, inventory listings and material flows are clearly defined;
- nuclear material is split into categories;
- changes in physical inventory are reflected in the book reflecting material transfers into and out of an MBA;
- accounting and operating records are maintained for MBA's and KMP's;
- accounting reports are generated for each MBA and submitted to the national supervisory authority;
- physical inventory is taken, balance striking is performed, and MUF is assessed;
- adequate containment and surveillance measures are taken;
- if necessary, automated systems are used for collection, storage and presentation of material-related information at the facility.

Promatomnadzor exercises the following functions:

- determines, in co-operation with the IAEA, the starting point for application of safeguards to nuclear materials;
- organises work in connection with the termination of the application of safeguards in cases foreseen by the Safeguards Agreement;
- organises work in connection with exemptions from and re-application of safeguards;
- accompanies IAEA inspectors;
- inspects accounting and control system at a facility;
- notifies enterprises about the IAEA inspections;
- receives from enterprises, evaluates, prepares and sends accounting documents to the IAEA in accordance with the Safeguards Agreement;
- organises approval of candidatures of IAEA inspectors;
- acts as a contact point receiving information from the IAEA and remittance of this information to governmental bodies and other entities.

Supervision of the adequacy of accounting for and control measures in respect of nuclear materials is carried out by the Promatomnadzor's through making inspections of enterprises and facilities. The requirements on and periodicity of inspections are established depending on the material category, type and design of installation, and containment/surveillance measures used.

3. Export/import control and prevention of non-authorized use of nuclear materials

Export and import control of nuclear materials and dual-use commodities is an important component of the safeguards system.

The Law of the Republic of Belarus «On Exports Control» defines the legal basis for activities of state bodies, legal and natural persons of the Republic of Belarus in the field of export

control and regulates relations arising in connection with the movement of objects subject to export control across the customs border of the Republic of Belarus and their subsequent use. The objects subject to export control (specific goods) include such items «as goods, technologies and services connected with nuclear fuel cycle and production of nuclear materials which can be used for production of nuclear weapons and nuclear explosive systems» as well as «dual-use commodities».

The Law sets up, in particular, such requirements as:

- licensing of imports/exports of objects subject to control;
- making pre- and post-licensing inspections of objects subject to export control;
- provision of state guarantees for the use of dual purpose commodities for the declared purposes;
- interaction with international organisations and export control bodies of other states;

In order to implement the Law «On Export Control», in January 1998 the Council of Ministers adopted a Decree *N° 27* entitled «On the Improvement of the State Control over the Movements of Specific Goods (Works, Services) across the Customs Border of the Republic of Belarus». The decree brings into force two regulations which are aimed at the implementation of the Law:

- Regulation on the Order of Licensing of Export (Import) of Specific Goods (Works, Services, Nuclear Materials) and
- Regulation on the Order of Official Registration of Obligations for the Use of Exported (Imported) Specific Goods (Works, Services, Nuclear Materials) for Declared Purposes and Organisation of Control over the Fulfilment of Such Obligations.

The Decree defines functions of ministries and other authorized bodies in the field of export control. The Ministry of Foreign Affairs (earlier the Ministry of External Economic Relations) is assigned to be a body empowered to issue licenses for export of specific goods (works, services,) as well as to co-ordinate activities of all agencies and institutions involved in the export control work. The decree establishes the procedure and size of payments to the state budget for licensing and other services.

Information support of the export/import control system is provided by ORACLE-controlled computer databases. The system includes the Ministry of Foreign Affairs, which issue export licences, the Customs Committee, the Academy of Sciences and other agencies.

State Customs Committee is responsible for detection of unauthorised imports and exports of nuclear and other radioactive materials at the customs border of the Republic of Belarus. They report each case to other responsible authorities.

Situated in a high-risk trafficking area, Belarus considers the task of combating illicit trafficking essential to ensure state security, public health and environmental protection. Belarus has always been a strong supporter of the activities aimed at stopping illicit trafficking carried out at the international level. Thus Belarus has been participating in the IAEA Illicit Trafficking Database. At the national level there have been also a series of activities underway to prevent, intercept and respond to illicit trafficking, such as strengthening regulatory control, detection of illegal cross-border movements, provision of training opportunities for the relevant personnel.

4. Physical Protection of Nuclear Materials

In 1993 Belarus became a contracting party to the Convention on the Physical Protection of Nuclear Materials. Through Regulatory Resolution No. 338 On Measures for Physical Protection of Nuclear Materials issued by the Council of Ministers of the Republic of Belarus in 1993, the Department for Supervision of Industrial and Nuclear Safety was appointed as the authority responsible for ensuring physical protection of nuclear materials and facilities and was tasked to develop and approve relevant normative documents. In 1994 Promatomnadzor issued the Order On Ensuring Physical Protection of Nuclear Materials during Use, Storage and Transportation that was in line with then relevant IAEA recommendations. The Order lays down functions and responsibilities of the state bodies concerned and operators, categorization of nuclear material and requirements to the physical protection during its use, storage and transportation. In accordance with the Order Promatomnadzor in co-operation with all the state bodies concerned drafts and revises orders and regulations on physical protection. Promatomnadzor is also a central point of contact on issues of physical protection of nuclear material in accordance with the requirements of the Convention on Physical Protection.

Promatomnadzor also exercises the following functions in the field of physical protection of nuclear materials and installations:

- co-operates with the IAEA and regulatory authorities of other countries on physical protection issues in case of an international transfer;
- co-ordinates activities aimed at implementing the Convention on the Physical Protection as well as activities of organizational and technical nature related to the operation of the system of physical protection;
- in co-operation with an operator maintains the state system of nuclear materials accounting and control.

In the Republic of Belarus all activities involving nuclear materials are to be licensed. As it is elsewhere the responsibility for establishing and maintaining a system of physical protection rests with an operator. Ensuring physical protection is one of the obligatory requirements to be met in order to get a license for design and operation of nuclear installations and storage of nuclear materials. The procedure of issuing licenses is regulated by the Council of Ministers. In the Republic of Belarus Promatomnadzor is responsible for issuing licenses for activities involving use, storage and transportation of nuclear materials. Before granting a license an obligatory expert evaluation is conducted and only if its results are positive a license is issued. It should be mentioned that activities related to the operation of a system of physical protection are not licensed since they form an obligatory requirement for a license to be issued for the activities involving nuclear materials.

Promatomnadzor exercises its functions in close co-operation with the Ministry of Internal Affairs, the State Security Committee, the Ministry of Transport and the State Customs Committee, in particular, in drafting relevant orders and regulations and combating illicit trafficking of nuclear materials.

Promatomnadzor, the Ministry of Interior and the State Security Committee exercise control over implementation of the physical protection of nuclear materials and installations in Belarus. The check-up of the system is to be conducted by a special commission consisting of representatives of Promatomnadzor, the Ministry of Internal Affairs, the State Security Committee and an operator at least once a year. Such checks are also conducted in each case

of considerable constructional or functional change of the system as they are aimed at assessing effectiveness of the physical protection measures and defining necessary upgrades to optimize the system in relation to concrete situations at an installation. In addition, an operator is to control effectiveness of the physical protection measures on a regular basis.

The State Security Committee is also responsible for granting authorizations to staff of an operator to enable them to have access to nuclear materials, exercising control over ensuring security of information on nuclear materials and system of physical protection.

As the Ministry of Internal Affairs ensures security it licenses all relevant activities.

In case of transportation of nuclear materials a permit of Promatomnadzor is needed. The Ministry of Internal Affairs is responsible for ensuring physical protection during transportation. In case of an international transfer the responsibility for ensuring physical protection is regulated by an agreement between the states concerned.

In Belarus all nuclear materials, including HEU, are located at the Joint Institute of Energy and Nuclear Research “Sosny” under the National Academy of Sciences. Up to 1996 the system of physical protection was based on components set up and put into operation in the 60’s as the facility was built. Of course, with time technical components were partially upgraded and some new ones were introduced, such as perimeter barriers, perimeter sensors, alarm systems, lightning systems, intrusion sensors, entry control, access control, central alarm station, etc. That system covered all important areas and installations on site. It should be mentioned that there are installations with significant quantities of radioactive materials in the territory of the facility. From the viewpoint of effectiveness and economy of funds their system of physical protection was integrated into the system of physical protection of nuclear materials.

The system of physical protection that is currently in place at the “Sosny” facility includes elements of the “old” system, and the “new” one installed in 1996 as a result of a multilateral co-operative effort between Belarus, Sweden, Japan and USA. The main technical components of the system are:

- detection system including magnetic, microwave and infrared sensors;
- video-surveillance system;
- system of access delay including electronic blocking devices;
- system of authorized access including magnetic cards;
- system of computerized control over all components and communication system.

The system is maintained by a group consisting of technical personnel and duty operators who monitor facilities 24 hours a day. Instructions and technical procedures were worked out for the operation of all major components of the system. During the operation of the system certain changes were made to the construction of some components to take into account local peculiarities. Thus, for example, the mechanical problem of a turnstile used at the entrance point of one of the most sensitive buildings was rectified, some electronic nodes were replaced (in the first instance, those for power supply, resistance to external effects, for example, with a view to minimizing false alarms). After the 11th of September 2001 Government of the Republic of Belarus undertook measures to revise and strengthen the system of physical protection of nuclear materials. Thus, physical protection of the most sensitive buildings was upgraded.

Without going into details, it can be mentioned that the system of physical protection makes it possible to maintain automatic control of access to the most sensitive zones and the facility as a whole, automatic detection and testing of all components, automatic generation of alarm signal in case of an incident and writing of video signal.

Experience of Thailand on conclusion of additional protocol

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Abstract. Thailand started Atoms for Peace programme since 1954 and ratified to International Atomic Energy Agency Comprehensive Safeguards Agreement in 1974 two year after sign the Treaty on the Non-Proliferation of Nuclear Weapon (NPT). On 22 September 2005 Thailand signed the Additional Protocol with IAEA and in the stage to implement the obligation under the Protocol.

1. Introduction

Thailand started Atoms for Peace programme since 1954 by appointing the “Thai Atomic Energy Commission” which was renamed as the “Thai Atomic Energy Commission for Peace” in 1956 and assigned to be responsible for and carry out all matters related to atomic energy for the peaceful purposes. After the United Nations had adopted the International Atomic Energy Agency Statute on 26 October 1956 and had founded the International Atomic Energy Agency. The Thai Government ratified the statute of IAEA on 15 October 1957 and consequently was 58th on the list of IAEA member states. On 26 April 1961, the Thai Government proclaimed the enactment of the Atomic Energy for Peace Act B.E. 2504(1961) for regulating the peaceful uses of atomic energy in Thailand. Under this Act the Office of Atomic Energy for Peace(OAEP) have been established and was renamed as “Office of Atoms for Peace(OAP)” on 3 October 2002. On 27 October 1962 the Thai research reactor was activated to criticality for the first time in Thai history. Thailand ‘s basic policy for peaceful utilization of atomic energy are to compliance with Nuclear Non-Proliferation, safety and security of nuclear facilities and radioactive materials.

2. Additional Protocol

The Agreement between the Government of the Kingdom of Thailand and the International Atomic Energy Agency for the application of Safeguards in connection with the Treaty on the Non-Proliferation of Nuclear Weapon have been in force on May 1974. Over the past 40 year Thailand’s nuclear activity in comply with the peaceful use of nuclear technology through the Office of Atoms for Peace under the Atomic Energy for Peace Act. Under a comprehensive safeguards agreement based on INFCIRC/153, OAEP has established a system of accounting for and control of all nuclear material subject to safeguards and maintained material accountancy records.

The initiation to conclude the Additional Protocol started on 2002 by setting up the national committee which include the representative from various function of government and private sector such as representative from Ministry of foreign Affairs, Ministry of Commerce, Ministry of Industry, Customs Office, Office of the Attorney General and the Federation of Thai Industries. The committee did set up the work plans for consideration the Protocol which are:

- a. Management arrangement for translation of Model INFCIRC/540 into the national language.

- b. General consideration all the content of the model including review the national legislation.
- c. Learning period on the technical aspect of the model for activities (nuclear fuel cycle-related R&D), buildings, nuclear-related manufacturing, mines/mills, source material locations and exports/imports, nuclear waste, complementary access and managed access.
- d. Outreach activities for impact issues on the research institutes and industrial sector through the Federation of Thai Industries.
- e. Consultation period with the other countries such as Australia and Japan.
- f. Decision making by Thai Atomic Energy for Peace Commission.
- g. Report for approval to sign by Thai Cabinet.

3. Conclusion

On 22 September 2005 Thailand signed the Additional Protocol with IAEA and in the stage to implement the obligation under the Protocol by review the Law and regulation, setting the various measures for the provision of Information under the declaration in Article 2 as well as strengthen state authority.

REFERENCES

- [1] Atomic Energy for Peace Act 2504(1961).
- [2] Treaty on the Non-Proliferation of Nuclear Weapons, INFCIRC/140. IAEA.
- [3] The Agreement between the Government of the Kingdom of Thailand and the International Atomic Energy Agency for the application of Safeguards in connection with the Treaty on the Non-Proliferation of Nuclear Weapon, INFCIRC/241.
- [4] Protocol Additional to the Agreement between the Government of the Kingdom of Thailand and the International Atomic Energy Agency for the application of Safeguards in connection with the Treaty on the Non-Proliferation of Nuclear Weapon.

Development of a safeguards database

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Abstract. An effort has been made to computerize, the information regarding IAEA safeguards inspections in Pakistan, the data about designated inspectors, material inventory at safeguarded facilities and details of safeguards agreements with the Agency. The database has been developed using ACCESS. The information such as searching of designated inspectors of a particular nationality, material inventory of a certain kind of nuclear material at a safeguarded facility or the information on any safeguards agreement can be easily traced using this database which provides flexible search options. All the information is managed from single database file. Within the file, the safeguards data is divided into separate tables. Various forms have been created to edit/add data directly to the tables. Various queries can be made to present data in a certain format.

1. Introduction to Database

An attempt has been made to computerize the information regarding safeguards inspections conducted at our facilities, the IAEA inspectors designated for Pakistan, material inventory at our safeguarded facilities and our safeguards agreements with the Agency. The information on these components such as searching of inspectors of a particular nationality ever designated to Pakistan, the material inventory of a certain kind of nuclear material present at any of the safeguarded facility, or the information on the type of safeguards agreement covering any facility can be easily traced using the database that provides flexible search options.

The Relational Database Management System ACCESS has been used for the development of the database. All the information is managed from a single database file. Within the file, the safeguards data is divided into separate tables, one for each type of information. Relationships are defined between the tables to bring the data from multiple tables together in a query, form, or report.

Various forms have been created to edit/add data indirectly to the tables. To find and retrieve the data that is frequently required, including data from multiple tables, various queries have been made. Reports have been created using these queries, to present data in a certain format.

In order to create application to navigate around the Safeguards Database, the Switchboard Manager has been used that automatically creates switchboard Forms that helps to navigate around the database. The switchboard Form has buttons that can be clicked to open various forms and reports or open other switchboards that can further open additional forms and reports. The Safeguards Database also employs various macros and event procedures to automate common tasks. In order to secure the Safeguards Database, the simplest method, that is to set a password for opening the Database, has been used.

2. Components of Safeguards Database (DB):

There are three components, which the Safeguards Database deals with; namely the inspections, the material inventory and the Safeguards Agreements.



Clicking the Safeguards Database icon on the desktop will prompt for user's password to confirm authorization to access Safeguards Database (DB). Being a valid user, the window as shown in Figure 1 will appear:



FIG. 1. Main Switchboard.

3. Inspections:

Our safeguarded facilities are subject to inspections pursuant to various Safeguards Agreements and Subsidiary Arrangements concluded with the International Atomic Energy Agency (IAEA). The inspections to the facilities are carried out by the IAEA inspectors designated for Pakistan. Beside inspection data, the Database maintains the records of designated inspectors especially the current list of designated inspectors for Pakistan according to our quota requirements of 14.

Clicking the “Inspections” button on the Main Switchboard Window (Figure 1) will open the “Inspections Window” (Figure 2) This “Inspections Window” gives options whether to update, view and search any inspection related activities.



FIG. 2. Inspections Window.

3.1 Update

It is the only category where we enter/edit data regarding the inspections conducted at our safeguarded facilities, the information about designated and de-designated inspectors and the facilities under IAEA safeguards. Clicking “Update” button on the “Inspections Window” (Figure 2) will further open the “Update Window” (Figure 3) as shown below:



FIG. 3. Update Window.

3.2 Update Inspection

To enter details of any inspection conducted by the Agency inspectors at our safeguarded facilities, clicking the inspection button on the above “Update Window” (Figure 3) will open the “Inspection Form” (figure 4) as shown below:



FIG. 4. Inspection Form.

In the above form, the “Facility Code” of the facility where the inspection has been carried out, is to be selected from drop down list of the combo box. The drop down list will automatically show the facility codes alongwith the names of all the facilities under safeguards.

The text box for the “Date of Inspection” in the above “Inspection Form” (Figure 4) will show the current system date by default for the new inspection entry, which can be changed to actual inspection date. The duration of the inspection is to be manually entered in terms of number of days in the text box provided for it. The combo box provided for the results of the inspection will show “Awaited”, by default for the new inspection entry.

However, it can be changed either to “No Departure” from the drop down list or by typing any other remarks depending on the actual results received from the Agency later on.

Lastly, the inspectors’ identification numbers (Inspector ID) for those Agency inspectors who have carried out the inspection are recorded. These are the system generated unique numbers for each of the designated inspectors. The drop down list will show the “Inspector ID” alongwith their names. This list will always contain only those inspectors’ names that constitute the current list of designated inspectors normally not more than 14, according to our present quota requirements. This list updates automatically for the current designated and de-designated inspectors. The list can be navigated with the help of scroll bar button provided for this purpose.

3.3 Update List of Inspectors

At present, we have a quota of 14 IAEA inspectors designated for Pakistan that can conduct inspections at our safeguarded facilities. According to the inspectors designation procedure set forth in paragraph 1 of the Annex to the Agency document GC(V)/INF/39 (Inspector document) IAEA may propose names of their inspectors to be designated for Pakistan. Pakistan Atomic Energy Commission (PAEC), after clearance from various Government Agencies/Ministries, may approve the name(s) of proposed safeguards inspectors. When a new inspector is designated for Pakistan, the following form in the Safeguards Database is completed for that inspector except the “Date of De-designation”:

The “Date of De-designation” is filled up for the inspectors whom the Agency de-designates. The record of the de-designated inspector can be easily found through records navigation buttons provided on the form.



FIG. 5. List of Inspectors Form.

3.4 View

Clicking the “View” button on the “Inspections Window” (Figure 2) will open the “View Window”(Figure 6) as shown below:

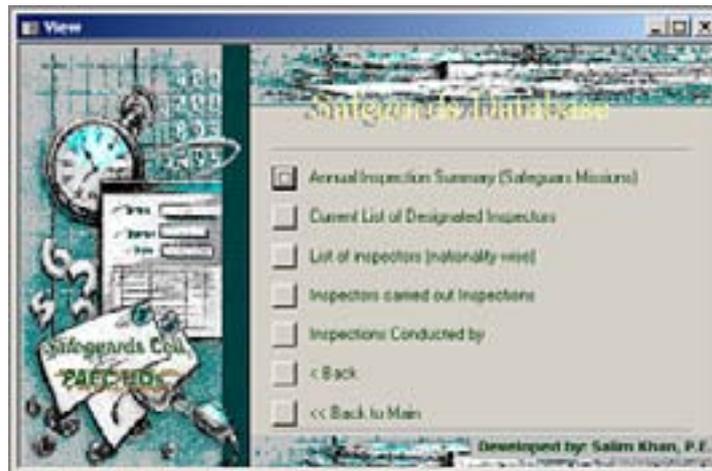


FIG. 6. View Window.

In this category, we extract data stored in the database in the forms of various reports frequently needed by us. The annual inspection summary gives information about the safeguards missions carried out in a particular year. Current list gives the names of the inspectors according to their natinality. The list gives maximum 14 names as agreed with the Agency. However, it will automatically adjust the maximum number of inspectors if there is any change in agreed quota with the Agency in future. The information of inspections conducted by a particular inspector at a selected facility can also be obtained.

3.5 Search

Clicking the “Search” button on the “Inspections Window” (Figure 2) will open the “Search Window” as shown below:

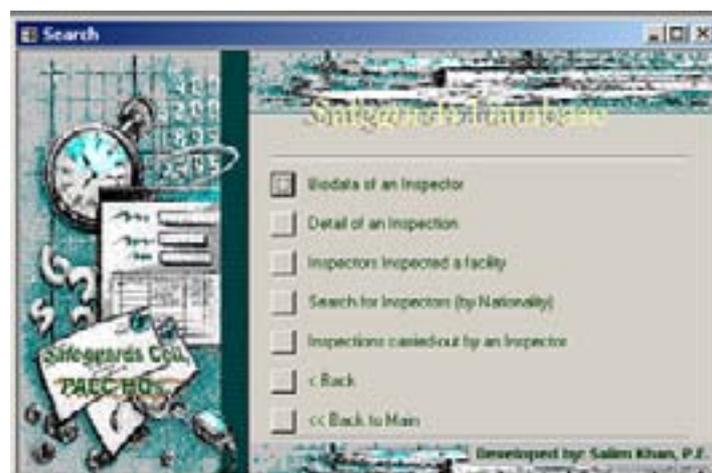


FIG. 7. Search Window.

4. Material Inventory:

The Next Area in the Safeguards Database is the Material Inventory for safeguarded nuclear or non-nuclear materials. Clicking the “Material Inventory” button on the “Main Switchboard Window” (Figure 1) opens the “Inventory Window” as shown in Figure 8.



FIG. 8. Inventory Window.

4.1 Edit MBR/Accounting Report:

PAEC submits Material Accounting Reports for its safeguarded facilities to the IAEA at regular intervals in accordance with the terms of various Safeguards Agreements and Subsidiary Arrangements/Facility Attachments concluded/agreed with the Agency. The material inventory in Safeguards Database is updated at the time of submission of these Material Accounting Reports to the Agency.

In order to update the Material Inventory in the Safeguards Database for any of the safeguarded facility, click the “Edit MBR/Accounting Report” button on the “Inventory Window” (Figure 8). This will open “MBR Window” as shown below in Figure 9:



FIG. 9. MBR Window.

In order to edit the MBR/Accounting Report of any of our facilities, we select its facility code and “inventory type” in the above window. This selection displays MBR of the facility in the lower portion of the window. All of the items under safeguards in this facility can be navigated through navigation button. Any change in the inventory of items in this facility can be entered. After many entries inventory changes the list becomes too lengthy. In such a case, if desired, we double click the lower most button in the above window to (algebraically) sum up the inventory with the remarks as “Updated”.

If there is any new type of material to be placed under safeguards or if any material that may not remain under Safeguards at any of our safeguarded facilities, we may either click the

square provided in the upper right corner of the above window or click the “Introduction of new material inventory” button in the “Inventory Window” of Figure 8. This opens the “New Inventory Window” as shown below:

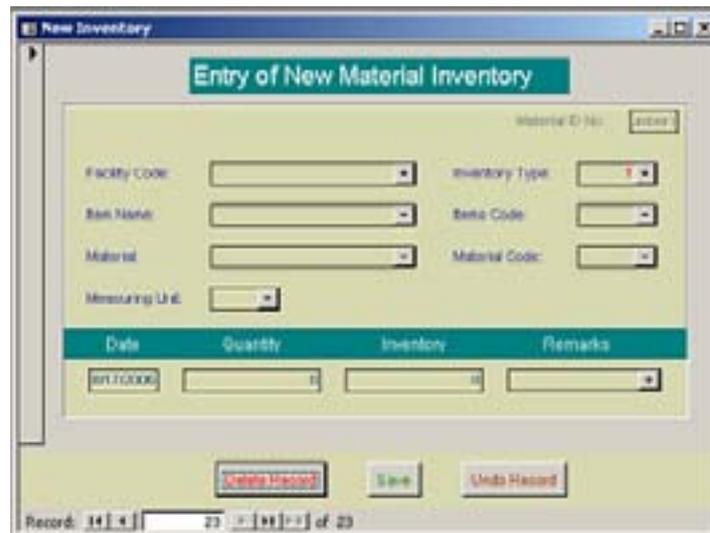


FIG. 10. New Inventory Window.

Now if we want to introduce new material inventory we enter its record in the above form. Similarly, if any material that may not remain under safeguards at any of our safeguarded facilities we navigate to find its record in the above window and then press the “Delete record” button in the above window.

5. Safeguards Agreements

The last component of the Safeguards Database deals with the Safeguards Agreements. Pakistan has concluded various bilateral and trilateral Agreements with the International Atomic Energy Agency (IAEA) and has some Subsidiary Arrangements/Facility Attachments enforced for the implementation of safeguards to our facilities. In order to deal with such information, in the Safeguards Database, click the “Safeguards Agreement” button on the “Main Switchboard Window” (Figure 1). This will further open the “Agreements Window” as in Figure 11 below:

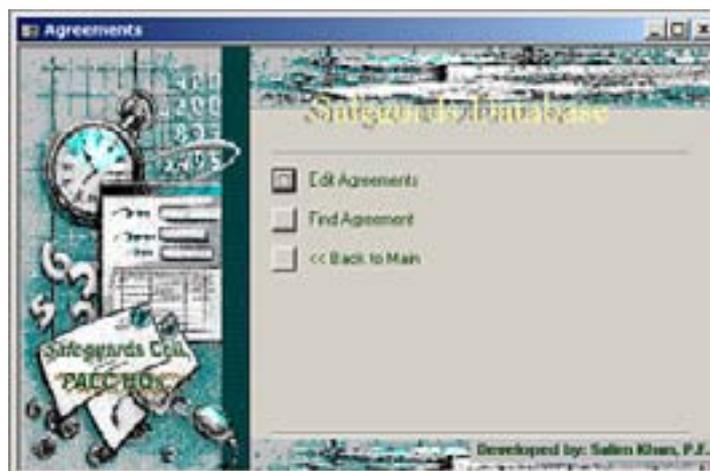


FIG. 11. Agreements Window.

A real-time system of accounting and control of nuclear materials and radioactive sources

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Abstract. The present study describes a real time system of accounting and control of nuclear materials and radioactive sources in a nuclear facility. The system's approach is based on the recording, accounting and verification of every item containing such materials during any process, or that moves in/or/out the facility at specific key measurement points. The system is designed to operate in coordination with the physical protection system of the facility. A database is created for chronological recording and retrieving of all transactions involving the material movements and inventory at any time. The investigated system is capable of controlling the flow of materials in the facility and generating real time reporting. This work may be of benefit to apply in different types of nuclear and/or radiation facilities, and in the field of combating illicit trafficking of nuclear materials or radioactive sources.

1. Introduction

Non-fossil fuelled energy systems such as nuclear fission reactors or thermo-nuclear fusion reactors use nuclear material(s) [NMs] (Thorium, Uranium, Plutonium,..., etc.) or other radioactive materials (Tritium, Lithium,....., etc.). In many other fields of peaceful applications of atomic energy (industrial, medical, agricultural,...., etc.), radiation facilities use various kinds of radioactive source(s) [RSs] (⁶⁰Cobalt, ¹³⁷Cesium,...., etc.). These materials are considered of high strategic value, high financial value and extremely dangerous [1-2].

The global security of NMs and RSs may necessitate the implementation of suitable approaches of nuclear safeguards [SG], and the application of accounting and control measures –which would be implemented– in coordination with physical protection [PP] measures for such materials at any place, in any time. This may be done by applying these measures whether materials are in inventory (storage), in operation, in manufacturing processes or in transport. Recent studies in the field of accounting and control of NMs (and/or RSs) and the PP of NMs and facilities have shown that SG and PP are two sides of the same coin for purposes of nuclear security. In principle, these are essential elements of national responsibility of the concerned State which should be implemented through appropriate authorized national regulatory systems [3-8].

On the one hand, the accounting and control system in a nuclear (or radiation) facility may deal with accounting, verification (including measurements) and control of NMs and/or RSs. An effective system should be capable of ascertaining quantitatively what, where and how much material is present; and how much material may be missing from the facility at any

given time. Also, the system should ascertain that the quantity, type, category and location of materials in the facility are conformal to the national nuclear regulations [3-6].

On the other hand, a reliable PP System [PPS] is needed in a facility for the purposes of securing, controlling and monitoring the NMs and RSs; and for the security of the facility itself. An effective PPS in such a facility should incorporate certain functions, namely; detection of any malevolent event that may be performed against the facility, delay of a supposed aggressor and a suitable response. These functions should be performed by the PPS in an orderly manner and within a length of time that should be less than the time required for the aggressor to complete his aggressive task in the facility [7-9].

The present study describes a developed real time system of accounting and control of NMs and RSs in a nuclear facility working as a research and development [R&D] laboratory. The system operates in coordination with the PPS of the facility.

2. Approach

In general, the purpose of the accounting and control of NMs (and RSs) is to establish the quantities of such materials within defined areas; and the changes in these quantities that may take place within defined periods of time. This necessitates the accurate, timely and full book keeping and recording of ALL and EACH category of the materials present in the facility. The defined area(s) in a facility are Material Balance Area(s) [MBAs] and Key Measurement Point(s) [KMPs]. The MBAs are areas into/or/out of which all material transfers can be performed at KMPs –which are- control locations where inventory or flow of materials can be determined [3-6].

The essential elements for implementing the accounting and control system necessitates that the facility operator should [3-6]:

- (a) Count, identify, keep and/or measure the NMs and RSs in the defined areas;
- (a) Prepare the records of ALL transactions involving the materials and;
- (b) Prepare the accounting reports of these transactions; and submit regulatory reports to the concerned national regulatory authority.

The methodology of the accounting and control system in the present work is based on a “Computerized Real Time Mode”. The system is designed to operate in coordination with the PPS of the facility in order to allow cross-checking of the inventory changes of NMs and RSs; hence enhancing the control and PP of materials in the facility [3,6,9,15].

For the purpose of verification of NMs and RSs in the facility, internationally acceptable verification techniques would be employed. Practically, Non Destructive Assay [NDA] measurements would be performed in order to attain the goal of the real time system of accounting and control. The NDA techniques -since many years- play an extremely important role in performing rapid and precise detection and identification measurements of NMs and RSs [10-12].

The basic verification measure in this study is the REAL TIME ACCOUNTANCY of NMs and RSs. This implies the application of real time accounting of materials and performing -in the same time- measurements to verify qualitatively and/or quantitatively the amounts of such materials. For this purpose, the facility operator would perform physical verification of inventory changes and/or inventory of materials, i.e. to perform [3-6,10-12]:

- (a) Item counting and tag checking (containers of uranium powder, uranium pellets,....,etc.);

- (b) Weighing, and/or estimating volume (if necessary);
- (c) Identifying the material by attribute measurement in order to detect missing items. This may be done using low or medium level γ -ray spectrometry (e.g. NaI (TI) detector, or CdZnTe detector, or;
- (d) Performing measurements - if needed - with a high level NDA technique; such as high-resolution γ -ray spectrometry using Germanium detector, in order to verify ^{235}U enrichment, or isotopic composition of the assayed material, or; using neutron counting to verify ^{235}U mass content.

3. System design

The employed measuring system for the verification of NMs and RSs is a classical γ -ray spectrometer. Its basic components are; NaI (TI) Scintillation Detector mounted on a Photomultiplier Tube, a Pre-amplifier, an Amplifier, a Multi-channel Pulse Height Analyzer, a stabilized High Voltage Power Supply, a suitable computer (PC – Pentium III or compatible with 20 GB hard disc) and a Printer [10-12].

A computer program is developed for constructing the real time accounting and control system which can deal with information concerning NMs and RSs as well. This is done by the aid of a software technique. The program uses a Database Management System [DBMS] which enables it to store, retrieve or modify data in the database [DB] as required by the operator. In order to access data from the DBMS, the program makes use of three “Oracle” tools [13-15].

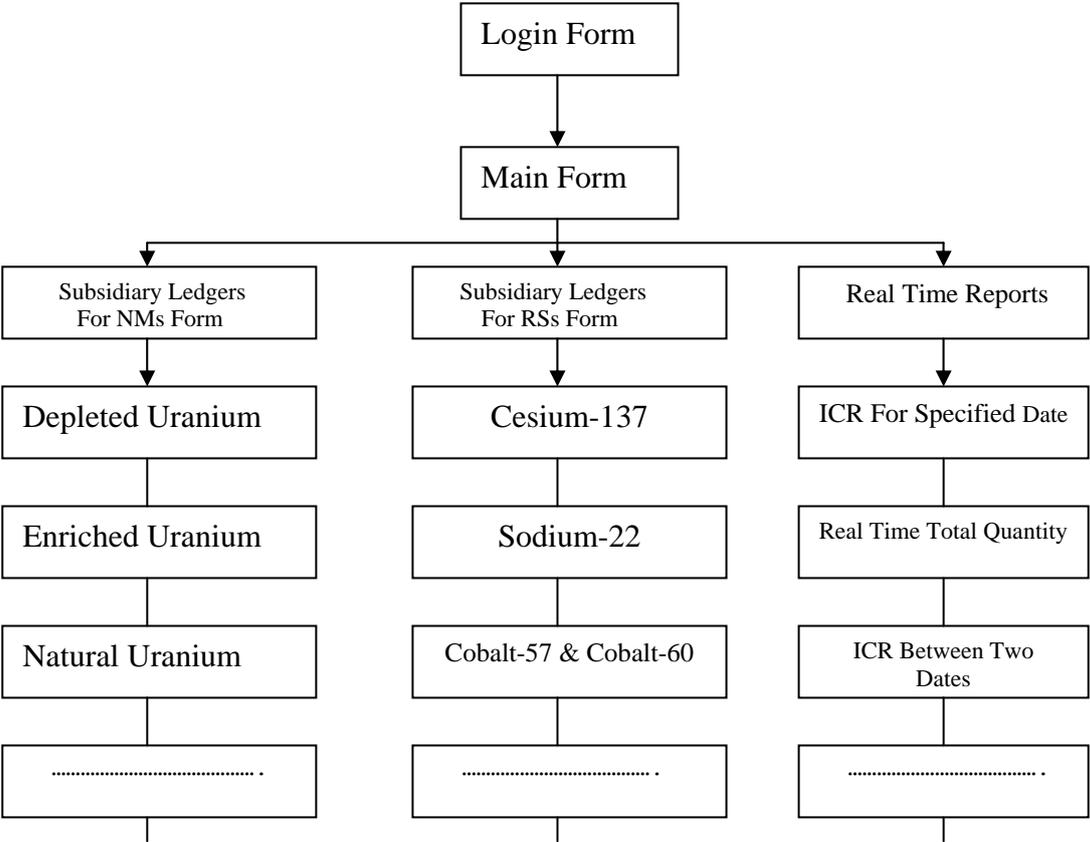
The first tool is an “Oracle Interface” which can be employed by the use of a Sequential Query Language [SQL] for communicating with the “Oracle Server” to access data, and/or a Procedural Language [PL] in coordination with the SQL. The PL/SQL is an extension to the SQL with design features of programming languages. The second tool is an “Oracle Form Developer” which is a productive development environment for building scalable database applications. The Third tool is an “Oracle Report Producer” for building reports as designed in the program. The flow diagram of the program is presented in Fig. 1. The program routine for building database forms, reports and tables includes subroutines (codes) for building the specific database of this work, for the creation of essential tables containing the data; and for creating the main forms and reports. Each one in the menu of these would contain the specific “Data Input” to be entered on real time mode for each movement or transaction of each kind of NMs and RSs in the facility [13-15].

4. Implementation and results

The real time system of accounting and control of NMs and RSs was implemented in a nuclear R&D laboratory. The system operates in coordination with the PPS of the facility. The details of its PPS have been reported elsewhere [9]. In the system database, the “Main form” contains different formats and reports designed to present the retrieved information in Subsidiary Ledger(s) [SL(s)] for each type of NM or RS. Also, it indicates “Menus” of SLs of NMs, RSs and that of different reports.

Performing immediate verification of the NMs or RSs is an obligation of the real time accountancy system when starting any operation or transfer of the material. Results of such verification -whether qualitative or quantitative- should be recorded by the operator according to the system procedure. The submitted information for each movement, operation or

transaction, i.e. in/or/out the facility should specify the date, time, place, location and type of operation; and the name, quantity, category and description of the material.



NB: ICR, Inventory Change Report

FIG. 1. Flow diagram of the real time accounting and control system of nuclear materials and radioactive sources.

The retrieval of reports from the menu of the “Reports Main Form” includes various reports containing information about ALL kinds of materials, ALL operations performed on the materials in the facility and ALL inventory changes in their corresponding dates. Some of such reports are Inventory Change Reports [ICR] in a specific date or in a period between two dates, for all materials or for a selected material. Other reports would give the total quantity of materials at a specific date for all materials or, for a selected material as required by the operator. Examples of reports such as; “Real Time Total Quantity For All Materials”, “ICR For Selected Material With Specified Date” and “ICR For Selected Material Between Two Dates” in the facility are presented in Tables I, II and III respectively.

It is important to remark that the information contained in the presented figure(s) or table(s) are ONLY examples indicating “Unreal Numbers”. The verification assays of NMs and RSs have been performed for the purpose of validating the performance of the real time accounting and control system of the present work. The verified materials are Standard NMs

– under the International Atomic Energy Agency Safeguards [IAEA SG] – at the Egyptian Atomic Energy Authority (MBA: ET-Z, KMP: E).

Table I. An example of “Real Time Total Quantity For All Materials” report^a.

Report run on : November 27, 2005 5:45 PM			
Facility Name : Nuclear R&D Laboratory			
Material Code	Material Name	Total Quantity	Fissile Quantity
			Units
D	Depleted uranium	537.150	
N	Natural uranium	200.300	
E	Enriched uranium	40.347	1.699
U	Unified uranium	291.930	
T	Thorium	56.420	
C3	Cesium-137	3	
C1	Cobalt-60	2	
B1	Barium-133	3	
C4	Cadmium-109	2	
S1	Sodium-22	2	

^a Information indicated in the Table(s) are “Unreal Values” used only for validating the system.

5. Conclusion

This work presents a developed system of accounting and control of NMs and RSs in a nuclear R&D facility. The system is based on a Computerized Real Time Mode. It is integrated to the facility PPS in order to allow for cross-checking of the material inventory changes. The system’s approach is based on recording, accounting and verifying of every item containing such materials during any process, or that moving in/or/out the facility at specific KMPs. A database is created for chronological recording and retrieving of all the material transactions involving the material movements, inventory changes and inventory at any time in the facility.

It may be concluded that the present system is capable of controlling the inventory and flow of NMs and RSs in the nuclear facility and generating real time information and reports. Further development in the system would be needed in order that it may be possible to generate “SG State Reports” conformal to “IAEA SG Formats”. Also, it may be of benefit to apply this work in different nuclear or radiation facilities; and in the field of combating illicit trafficking of NMs and RSs.

Table II. An example of "ICR For Selected Material With Specified Date" report.

Transfer Date	Material Name	Pervious Quantity	Previous Quantity (Fissile)	In Quantity	out Quantity	Fissile Quantity	Current Quantity	Total Quantity (Fissile)	Units	MDC	Isotope Code	Location
13- JUN-05	Depleted uranium	69.055	5.654				74.709		gram	qwer		physics lab
13- JUN-05	Depleted uranium	74.709		3.654			71.005		gram	qwer		accelerator lab
13- JUN-05	Depleted uranium	71.055		4.124			66.931		gram	qwer		safeguards lab
13- JUN-05	Depleted uranium	66.931		6.765			60.166		gram	qwer		physics lab
13- JUN-05	Depleted uranium	60.166	15 .123				75.289		gram	qwer		accelerator lab
13- JUN-05	Depleted uranium	75.289		20.654			54.635		gram	qwer		physics lab

NB: MDC, Material Description Code.

Table III. An example of “ICR For Selected Material Between Two Dates” report.

Report run on											
February 4, 2006 9:05 AM											
Facility Name	: Nuclear R&D Laboratory										
Transfer Date	Material Name	Pervious Quantity	Previous Quantity (Fissile)	In Quantity	out Quantity	Fissile Quantity	Current Quantity	Current Quantity (Fissile)	Units MDC	Isotope Code	Location
04-APR-05	Depleted uranium	78.055		7.001			71.054		gram	qwer	safeguards lab
04-APR-05	Depleted uranium	71.054		3.001			68.053		gram	qwer	safeguards lab
05-APR-05	Depleted uranium	68.053	1.002				69.055		gram	qwer	safeguards lab
13-JUN-05	Depleted uranium	69.005	5.654				74.709		gram	qwer	safeguards lab
13-JUN-05	Depleted uranium	74.709		3.654			71.055		gram	qwer	safeguards lab
13-JUN-05	Depleted uranium	71.055		4.124			66.931		gram	qwer	safeguards lab
13-JUN-05	Depleted uranium	66.931		6.765			60.166		gram	qwer	safeguards lab
13-JUN-05	Depleted uranium	60.166	15.123				75.289		gram	qwer	safeguards lab
13-JUN-05	Depleted uranium	75.289		20.654			54.635		gram	qwer	physical lab

NB: MDC, Material Description Code.

ACKNOWLEDGEMENT

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Proposed national actions for safeguards improvements in the United Republic of Tanzania

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Abstract. IAEA safeguards alone cannot ensure the physical protection of nuclear and other radioactive material of nuclear facilities from terrorists. It is the responsibility of every country to undertake all necessary safety and security measures and to adequate control of such material and facilities. Moreover, the discovery of clandestine nuclear weapons programme in some countries despite an existing comprehensive safeguards agreement demonstrated that an effective verifications regime must also focus on possible undeclared material and activities. For the Agency to be able to fulfill its verification responsibilities in a credible manner, the additional protocol must become the standard for all countries that are party to the treaty on the Non-Proliferation Nuclear Weapons. The aim of additional protocol is to provide assurance about both declared and possible undeclared activities. The IAEA safeguards system provides the state with a means to demonstrate transparency in its nuclear activities and that it is complying with its non-proliferation undertakings. Under the Treaty on the Non-Proliferation of Nuclear Weapons, governments around the world including United Republic of Tanzania have committed to three common objectives: Preventing the proliferation of nuclear weapons; pursuing nuclear disarmament; and promoting the peaceful uses of nuclear energy. It is widely recognized that establishing and maintaining effective national controls of nuclear material and nuclear activities is not only a legal obligation under the NPT, but is also in the national interest of each state. In this regard the United Republic of Tanzania (URT) has designated Tanzania Atomic Energy Commission (TAEC) to exercises oversight and control over any nuclear materials and activities on our territory. Our system of Accounting for and Control of Nuclear Material (SSAC) aims to the deterrence and detection of any theft or misuse of nuclear material thereby contributing to the security of nuclear material and combating illicit trafficking. From 1996 to 2002 in the URT there were 13 incidents of illicit trafficking reported by police. Some of the prevailing problems were identified as: Lack of public awareness to assist reveal the secrets movement and selling process; and lack of proper training system and equipment for border control officials, customs and law enforcement officers. This paper aims to discuss the proposed national actions for safeguards improvements in the country. The proposed actions focus on: prevention, detection and response of any illicit or non-peaceful use of nuclear or other radioactive materials in the country; building up nuclear security awareness and provision of training; consistence of national nuclear legislation and regulations subject to a state's obligations under the NPT and its safeguards agreement and additional protocol; upgrading national accounting and SSAC capabilities; upgrading detection capabilities at borders; and international cooperation.

Introduction

United Republic of Tanzania is one of some countries in the world, which do not intends to manufacture or acquire nuclear weapons or nuclear explosive devices or seek or to receive assistance in the manufacture of nuclear weapons or nuclear explosive devices. This is proved by the country voluntarily became party to the NPT and the decision to conclude a safeguards agreement with the IAEA and sign and ratify the Additional Protocol, which entered into force on 07/02/2005. Our existing legislation is sufficient to implement the Additional Protocol. In this year we have submitted to the IAEA a hard copy of declarations pursuant to the Additional Protocol under articles 3.a, 3.c, 3.d, and 3.e. [5]. The aim of this paper is to summarize the proposed national actions for safeguards improvements in the United Republic of Tanzania.

Prevention detection and response of any illicit or non-peaceful use of nuclear or other radioactive materials in the country

Illicit trafficking of radioactive material in URT has been an issue of concern since the first seizures in the 1990s. Most of confirmed cases have a criminal dimension, even if they were not for known terrorist purposes. The attack of USA embassy in Dar Es Salaam in 1998 dramatically emphasized the requirement for the enhanced control and security of nuclear and other radioactive material in the country. In response to control illicit trafficking in the country, TAEC has been given authority by Atomic Energy Act of 2003 to combat illicit trafficking and the inadvertent movement of radioactive material in the country. Some of the TAEC's activities are: physical protection of

nuclear material and nuclear installations; nuclear material accountancy, detection and response to illicit nuclear trafficking; the security and safety of radioactive sources and emergency response measures. TAEC cooperates with other organization such as police, customs and border inspectors in joint efforts to prevent incidents of illicit trafficking and inadvertent movements in the country, by providing training and technical assistance. In this context the TAEC has already organized three training courses co-sponsored by IAEA, USDOE and Interpol on the inadvertent movement and illicit trafficking of radioactive material. Although many has been done to combat illicit trafficking in the country, the following areas should be taken into consideration by TAEC:-

- To install fixed radiation portal monitors (RPMs) at check points such as those at road and rail border crossings, airports or seaports to detect the presence of gamma and neutron radiation and alert the officers to the presence of radioactive material;
- To provide personal radiation detectors (PRDs) to law enforcement officers on duty. PRDs are suitable for use by first responders to a radiation alarm because of their small size and they do not require extensive training to operate;
- To issue hand held radionuclide identification devices to every point that use RPMs, hand held RIDs would be used to verify an alarm triggered by RPMs localize the source and identify the radionuclide;
- To sign a memorandum of understanding with Tanzania postal corporation (TPC) that aims at ensuring the safe and secure transport of acceptable radioactive material through the mail and detection of illicit radioactive material, including nuclear material in the mail stream;
- To develop joint training programs and awareness campaigns with TPC on application of the strict measures to regulate the mailing of radioactive materials and to ensure the measures on a detection of illicit trafficking involving radioactive material are in place. Up to date no confirmed information about the illicit trafficking of radioactive material in public mail in the country. It must be noted that the transport of conventional explosive has taken place in public mail and has led to health hazard and death. The combination of radioactive material and conventional explosive in a letter or parcel should be identified to represents a potential risk to the public and postal workers as well.
- Advertisement on the TV, popular newspapers, TAEC's website and education to the public on how to reveal the secrets of movements of illicit trafficking and selling process.

Building nuclear Security awareness and provision of training

The September event in USA focused many countries in the world attention on the importance of preventing terrorist or other criminal misuse of nuclear material. In this context, TAEC is on progress to formulate a national security plan in cooperation with law enforcement organs such as police and other government authorities. The plan is aiming at strengthen the physical protection of nuclear material and other radioactive material in use, storage or transport and of nuclear radioactive facilities against nuclear terrorism; the security of radioactive materials in non – nuclear applications; the national capability for detection and unauthorized movement of radioactive materials; the national radiological emergency response capabilities and the country's multilateral / bilateral linkages with international organizations / countries with nuclear security programme.

On the side of training programme on building nuclear security awareness in the country, TAEC is advised to cover on the following: -

- The common causes of loss of control of radioactive materials;
- The importance of effective nuclear regulatory authority;
- Importance of having effective national radiation protection services and adequate nation nuclear legislation and regulations;
- Commitment by the management to safety and security;
- Danger of inadequate security during storage, transport and use of radioactive material;

- Legal action on any deliberate avoidance of regulatory requirements, theft and malevolent use; and
- The importance of media for transmitting information while recognizing the need for and maintaining confidentiality of sensitive information.

Consistence of national nuclear legislation and regulations subject to a state's obligations and its safeguards agreement and additional protocol

It should be recognized that the public safety and security could be threatened by some form of nuclear terrorism. The recent highly organized terrorist attack in Tanzania, Kenya and United states and in some other countries in the world has demonstrated the need for increased security and safety of nuclear material, domestically and internationally. Moreover the discovery of clandestine nuclear weapons programme in some countries despite an existing comprehensive safeguards agreement demonstrated that the Additional Protocol (AP) must become the standard for all countries that are part to the treaty on the Non – Proliferation of Nuclear Weapons (NPT) and the requirements of AP should be incorporated to every state's nuclear legislation [4,6]. By recognizing the threat picture, the Atomic Energy Act of 2003 has empowered TAEC to exercises oversight and control over any nuclear materials and activities in our territory to ensure that peaceful applications of nuclear energy contributes effectively to the country's economic growth. The recent regulations specify only minimum requirements for safety and security of radioactive material [2]. In order to make sure an effective SSAC in our nation, the procedures on how to deter and detect of any theft or misuse of nuclear material or illegal transport of radioactive, which may cross our borders, should be incorporated in the recent atomic energy regulations.

Upgrading national accounting and SSAC capabilities

IAEA activities under its safeguards programme serve to reinforce member state's efforts to combat nuclear terrorism. Under IAEA INFCIRC/153 agreements require states to have effective state systems of Accountancy and Control (SSAC) for nuclear material, and the IAEA provides assistance to member states in ensuring the technical effectiveness of these systems [3]. Such nuclear material accountancy and its verification by the IAEA contribute substantially to the deterrence and detection of any theft of nuclear material. It should be noted that each state with a comprehensive agreement is required to establish and maintain a state system of Accounting for and control of nuclear material (SSAC). SSACs are the state authority, office or persons formally designated to keep track of nuclear material and activities, and to interact with national or international entities such as the IAEA on safeguards implementation matters. It should be recognized that the IAEA has established several advisory services so that to help member states improvement of their national physical oversight systems. Through the international SSAC advisory service (ISSAS), the URT can get advice on how to strengthening it's SSAC. ISSAS missions compare the procedures and practices in member states with the obligations specified under safeguards agreements, with international consensus guidelines and against equivalent practices in other countries. Also IAEA encourage development of a DBT (Design basic threat) for each of its member states to strengthen effectiveness of their physical protection systems and offers workshops to help them define and implement a DBT and the measures needed to protect the states nuclear facilities. It is my great hope every member states including URT should upgrade its national accounting and SSAC capabilities as assistance to establish and maintain state systems for accounting for and control of nuclear material (SSAC) is provided free by IAEA upon request.

Upgrading Detection Capabilities at Borders

It is agreed that terrorists might attempt to steal a nuclear weapon or they could attempt to acquire the nuclear material necessary for constructing a nuclear device or try to acquire radioactive materials with the goal of making a radiological dispersal device. The transportation of these illegal radioactive materials is done through the borders of our nations. These tells us that the entries points of borders of each country should be equipped with radiation detectors to control any illegal movement of radioactive materials through the borders. In United Republic of Tanzania most of its

entry points are not equipped with radiation detectors. Efforts should be made to ensure the most busy entry points are equipped with radiation detectors and that awareness training to border inspectors is provided.

International Cooperation

One of the responsibilities of TAEC is to promote national and international cooperation or collaboration on the applications of atomic energy and nuclear technology already introduced or intended for introduction in the United Republic [1]. Since 1983 the country has participated in a number of IAEA projects such as in the area of waste management and nuclear safety and security. Tanzania has also signed international conventions such as non-proliferation of nuclear weapons treaty (NPT), additional protocols on safeguards and is also participating in IAEA's early notification of radiological accidents scheme and illicit trafficking data base (ITDB). Furthermore, Tanzania has signed agreement with USDOE for upgrading security in facilities that use radioactive material in the country. Also INNSERV mission was invited in the country to assess and advise on the state control and accountability of radioactive materials the cradle to grave concept. It should be noted that through the cooperation with IAEA and USDOE the country has benefited from expert mission advisory from IAEA, postgraduate education in radiation protection, fellowships, scientific visits, upgrading security of facilities which use radioactive material and various equipments for detecting ionizing radiation have been received. It is a task of URT to make sure the cooperation should be maintained and to continue to seek cooperation with more international organization on peaceful application of nuclear energy.

Conclusion

In this paper proposed actions on how to improve safeguards in URT has been presented. Behind this proposed action the country require the support of IAEA and other international organization to support the good political will demonstrated by the government of United Republic of Tanzania.

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ALOS satellite imagery utilization for safeguards

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Abstract. This paper introduces examples of satellite imageries analysis and utilizations for safeguards activities. Recently, many kinds of satellite imageries are available in the world i.e., high-spatial resolution, multi- and hyper-spectral, multi-function, and multi-polarization radars. The problem is that how to obtain the required information from these digital images.

The Japan Aerospace Exploration Agency (JAXA) has successfully launched the Advanced Land Observing Satellite (ALOS, nicknamed “Daichi”) on 24 January, 2006. The mission objectives are cartography, regional observation, disaster monitoring etc. ALOS has three instruments, PRISM, AVNIR-2, and PALSAR, to achieve these objectives. The ALOS research program can be categorized into two main parts of the ALOS mission: calibration and validation (Cal/Val), and application and science. We are presently carrying out the initial calibration and validation works for each instrument. It is very important to improve the absolute accuracy and image quality of the products. We establish and organize the international ALOS Cal/Val and science team (CVST) to do these works effectively. Based on the CVST activities, we are setting test sites and reference facilities in the world, and considering mission operation plans of each instrument to observe the test sites. The application and scientific results will demonstrate the ALOS data utilization capability. We define the digital surface model (DSM), ortho-rectified images for each sensor, and the surface deformation map as high-level and research products. In particular, geographical information such as elevation, topography, land use and land cover maps are necessary as basic information in many fields of practical applications and research areas. This paper describes our research and scientific activities including some preliminary results of calibration and validation, and examples of acquired images and high-level products related to geographical information.

1. Introduction

The Advance Land Observing Satellite (ALOS, nicknamed “Daichi”) was successfully launched on 24 January, 2006 (Japan Standard Time, JST) by an H-IIA rocket from Tanegashima Space Center, Tanegashima Island, Kagoshima Prefecture, where is located in southern part of Japan. The mission objectives of ALOS include cartography, disaster monitoring, etc. In particular, geographical information (e.g. elevation, topography, and land use and land cover maps) is necessary in many fields of practical application and research. To achieve these objectives, ALOS has three mission instruments, two optical sensors called PRISM and AVNIR-2, and a Synthetic Aperture Radar called PALSAR. The characteristics of ALOS, sensors as well as their products, and overviews of the ALOS research and science program were described by [1]. PRISM consists of three radiometers for nadir-, forward-, and backward-looking. It is used for generating a digital surface model (DSM) with high spatial resolution. AVNIR-2 has a function of pointing angle change from +44 to -44 degrees.

This paper introduces our research and scientific activities including some preliminary results of calibration and validation, and examples of acquired images and high-level products that include satellite imageries analysis and utilizations for safeguards activities.

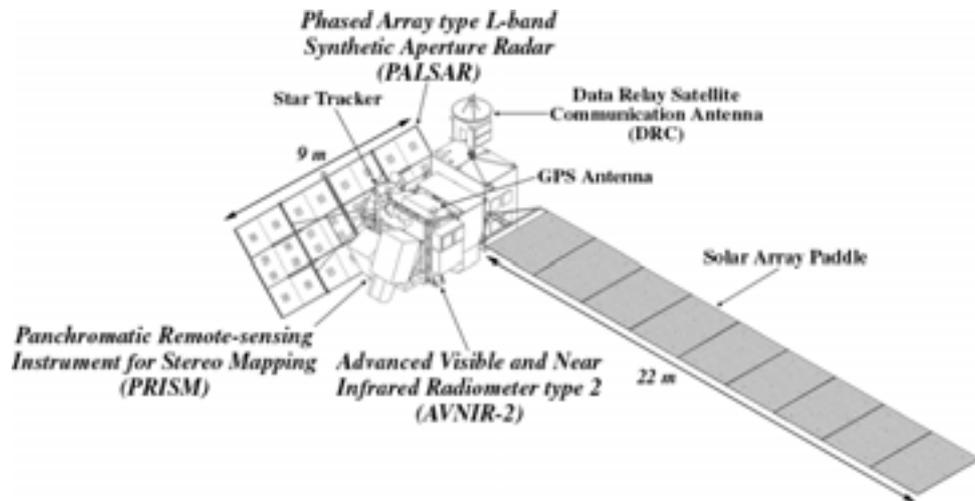


FIG. 1. Overview of ALOS.

2. ALOS Overviews

2.1. Characteristics of ALOS

ALOS follows the Japanese Earth Resources Satellite-1 (JERS-1) and the Advanced Earth Observing Satellite (ADEOS) and utilizes advanced land-observing technology. Figure 1 shows an overview of ALOS. The satellite mass is approximately four tons, and the design life is three to five years. ALOS will fly in a Sun-synchronous orbit with an inclination angle of 98.16 degrees, 691.65 km of altitude at the equator, and a repeat cycle of 46 days. The mission data can downlink at either a data rate of 240 Mbps via the Data Relay Technology Satellite (DRTS) or at 120 Mbps for direct transmission to the ground station. ALOS also has a solid-state data recorder with a capacity of 90 GB. The details of the satellite and its sensors were described by [2] and [3].

2.2. PRISM characteristics

PRISM stands for Panchromatic Remote-sensing Instrument for Stereo Mapping and will be used to generate a digital surface model (DSM) that terrain height information, which is one of the mission objectives of ALOS. PRISM consists of three independent optical systems for forward-, nadir-, and backward-looking to generate an accurate DSM. It acquires the images in the same orbit at almost the same time with a 2.5-meter spatial resolution.

Figure 2 illustrates the PRISM observation geometry for triplet mode and nadir- and backward-looking mode, and Table I summarizes the primary characteristics of the PRISM. Each radiometer consists of three mirrors and six or eight Charge Coupled Device (CCD) detectors for push-broom scanning. The nadir-looking radiometer can provide coverage 70 km wide, and the forward- and backward-looking radiometers each provide coverage 35 km wide. The radiometers are installed on both sides of the optical bench, which has precise temperature control function. Forward and backward radiometers are inclined ± 23.8 degrees from the nadir to realize a base-to-height ratio of one. Each radiometer functions for pointing (within ± 1.5 degrees) and electrical trimming to obtain an image of the same observed area between other radiometers in the cross-track direction. Thus, three fully overlapped stereo images (triplet) 35 km wide are obtained without mechanical scanning or yaw steering of the satellite. Without this wide field of view (FOV), forward-, nadir-, and backward-looking images would not overlap due to the Earth's rotation. Pointing angles of ± 1.2 degrees are reasonable to observe overall coverage in the triplet global observation mode. The acquired data are separated into odd- and even-numbered detectors and compressed by lossy compression with a joint photographic experts group (JPEG) extension, based on the Discrete Cosine Transform (DCT), quantization, and Huffman coding to reduce the downlink data rate from ALOS to the ground stations. The averaged compression data rates can be selected as either 1/4.5 or 1/9.

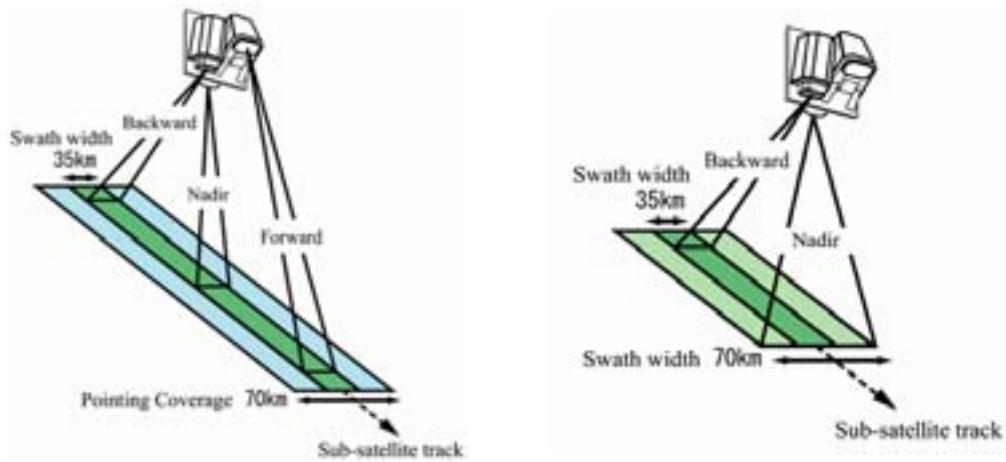


FIG. 2. Observation geometry of PRISM; triplet mode (left) and the nadir and backward mode (right).

Table I. PRISM characteristics.

Number of bands	1 (Panchromatic)
Wavelength	0.52 - 0.77 micrometers
Number of optics	3 (Nadir, Forward, and Backward)
Base to height ratio	1.0 (between Forward and Backward)
Spatial resolution	2.5 m (Nadir)
Swath width	70 km (Nadir or Nadir + Backward) / 35 km (Triplet mode)
Signal to noise ratio	> 70
MTF	> 0.2
Number of detectors	28,000 (70km swath) / 14,000 (35km)
Pointing angle	-1.5 to +1.5 deg. (Triplet mode)
Bit length	8 bits/pixel
Data rate	960 Mbps (Triplet mode)
Data compression	Lossy, JPEG extension (onboard)
Data downlink rate	240 Mbps (1/4.5 compression) / 120 Mbps (1/9 compression)

Table II. AVNIR-2 characteristics.

Number of bands	4
Wavelength (micrometers)	Band 1 : 0.42-0.50 Band 2 : 0.52 - 0.60 Band 3 : 0.61 - 0.69 Band 4 : 0.76 - 0.89
Spatial resolution	10 m (at Nadir)
Swath width	70 km (at Nadir)
Signal to noise ratio	> 200
MTF	Band 1-3 : > 0.25 Band 4 : > 0.2
Number of detectors	7,000 / Band
Pointing angle	-44 to +44 deg.
Bit length	8 bits/pixel
Data downlink rate	120 Mbps

2.3. AVNIR-2 characteristics

AVNIR-2 stands for Advanced Visible and Near Infrared Radiometer type 2. It is a visible and near-infrared radiometer for observing land and coastal zones and provides better spatial land coverage maps and land-use classification maps for monitoring regional environment. AVNIR-2 is a successor

to the previous AVNIR sensor mounted on the first Advanced Earth Observing Satellite (ADEOS), which operated from 1996 to 1997.

Table III. PALSAR characteristics.

Mode	Fine Resolution		ScanSAR	Polarimetry
Center frequency			1270 MHz (L-band)	
Chirp bandwidth	28 MHz	14 MHz	14 or 28 MHz	14 MHz
Polarization	HH or VV	HH+HV or VH+VV	HH or VV	HH+HV+VH+VV
Incidence angle	8-60 deg.	8-60 deg.	18-43 deg.	8-30 deg.
Range resolution	7-44 m	14-88 m	100 m (multi-look)	24-89 m
Swath width	40-70 km	40-70 km	250-350 km	20-65 km
Bit length	5 bits	5 bits	5 bits	3 or 5 bits
Data downlink rate	240 Mbps	240 Mbps	120 or 240 Mbps	240 Mbps
NE sigma zero	< -23 dB (70 km swath)		< -25 dB	< -29 dB
S/A	> 16 dB (70 km swath)		> 21 dB	> 19 dB

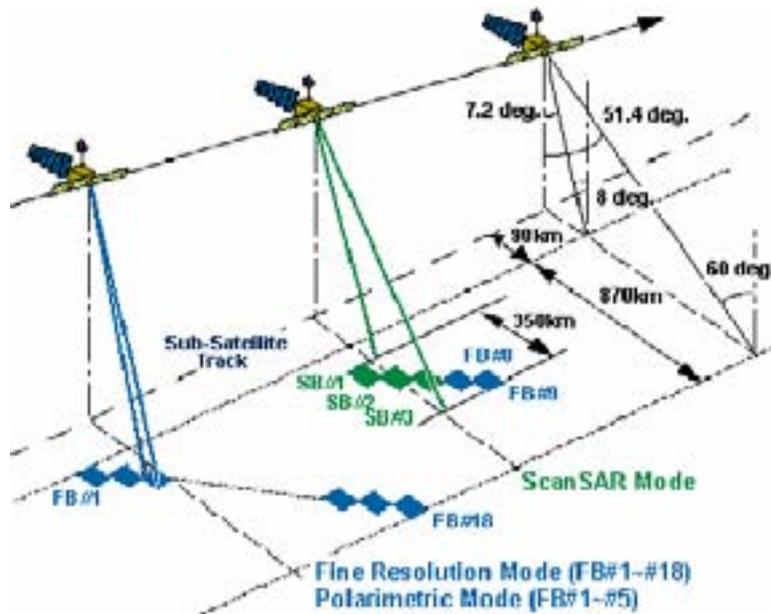


FIG. 3. Observation geometry of PALSAR.

Table II summarizes the characteristics of the AVNIR-2 sensor. Its primary improvement over AVNIR is its instantaneous FOV. AVNIR-2 provides 10-meter spatial resolution images in the multi-spectral region, compared with the 16m resolution of the AVNIR. The increased resolution was realized by improving the CCD detectors (AVNIR 5,000 pixels per CCD, AVNIR-2 7,000 pixels per CCD) and their electronics. Another improvement is a cross-track pointing function for prompt observation of disaster areas. The pointing angle of AVNIR-2 is +/- 44 degrees, corresponding to a +/- 750 km swath width on the ground.

2.4. PALSAR characteristics

PALSAR stands for Phased Array type L-band Synthetic Aperture Radar (PALSAR). It is an active microwave sensor using the L-band frequency for cloud-free and day-and-night land observations, and improved performance over the Japanese Earth Resources Satellite-1 (JERS-1) SAR.

Table III summarizes the characteristics of PALSAR, and Figure 3 illustrates the observation geometry. PALSAR has three observing modes, fine resolution with single or dual polarization, ScanSAR, and full polarimetry. The fine resolution mode is a conventional one; the ScanSAR mode

can observe 250 to 350 km wide (depends on the number of scans) of SAR images at the expense of spatial resolution. This is a swath three to five wider than conventional SAR images.

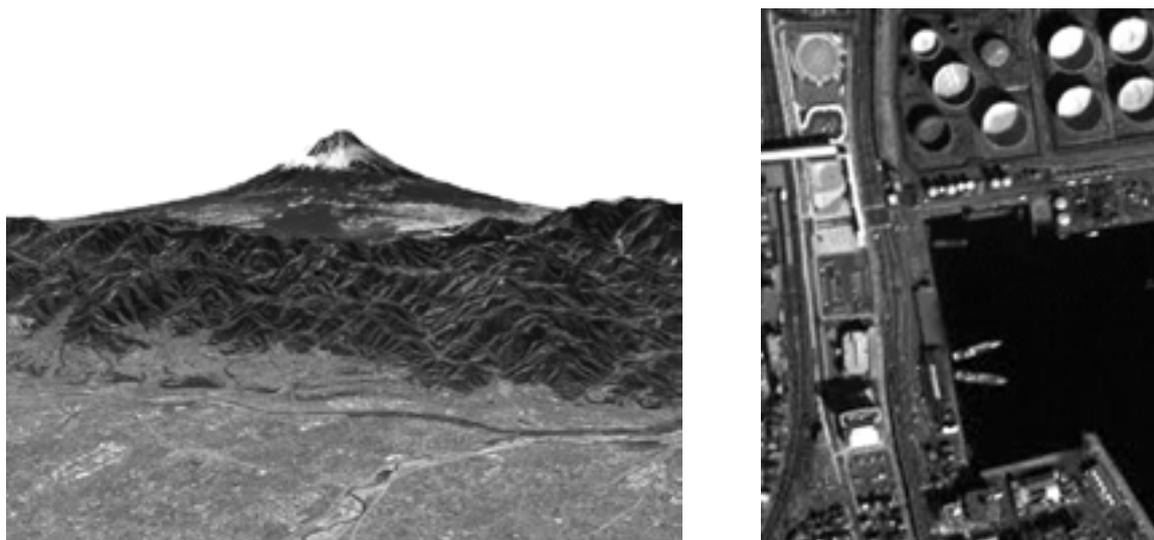


FIG. 4. The first acquired images of PRISM on 14 February 2006; the bird's-eye view of Mt. Fuji, Japan (left), and the enlarged image of Shimizu Port, Shizuoka Prefecture, Japan (right).

2.5. Product summaries

JAXA defines three different product types; the first is the standard product, which consists of Levels 1A, 1B1 and 1B2 for optical sensors and Levels 1.0, 1.1 and 1.5 for PALSAR, processed in the Earth Observation Center (EOC) of JAXA. The format descriptions of the standard products are published via internet [4]. The others are high-level and research products generated in EORC. The high-level products are digital elevation models (DEMs) prepared from PRISM's stereo pair image and the PALSAR interferometry technique, and ortho-rectified images of each sensor. The DEM generated by PRISM is actually not an elevation model but a digital surface model (DSM). We plan to generate the geophysical parameters for the research products, i.e., forest and biomass related parameters, surface deformation detection, sea-ice distribution, soil moisture distribution, and snow related parameters, primary using the PALSAR data. Land-use and land-cover classification, and surface albedo will be estimated by AVNIR-2. These products will demonstrate the ALOS data utilization capability as well as contribute to scientific activity.

3. Examples of ALOS image and analysis

3.1. First acquired images of PRISM

PRISM can be derived terrain information using triplet images. Figure 4 shows the first acquired images of PRISM on 14th February 2006. The left image in Figure 4 presents a bird's-eye view of Mt. Fuji, produced by overlaying the nadir-looking image with generated relative-accuracy DSM, which was generated the image-matching technique to calculate parallax using nadir- and forward-looking images, and nadir- and backward-looking images with tie-points (TPs). The processing software of DSM and ortho-rectified image using PRISM stereo pair images has already developed based on the triplet image matching technique [5], [6]. It basically uses satellite orbit and attitude information. However, the satellite orbit and attitude were not stable nor perfectly controlled at that time. So, we applied relative-accuracy DSM generation method. The right image in Figure 4 shows an enlarged image of PRISM nadir-looking radiometer around Shimizu Port, Shizuoka Prefecture, Japan that acquired same day. We can identify ships, cars, the inside of tanks etc.

3.2. PRISM DSM generation and validation

This subsection describes test results of DSM generation using actual PRISM stereo pair images and the operational software, and their preliminary validations. The DSM generation software has

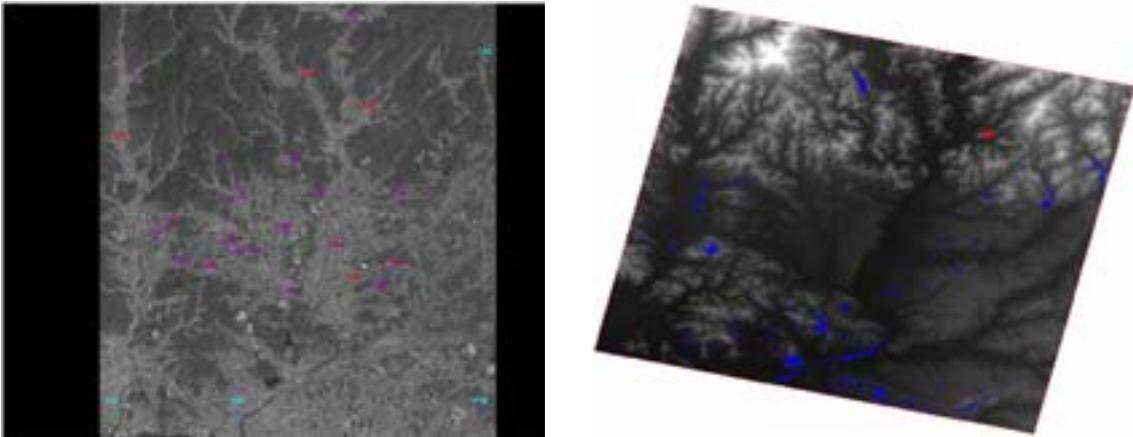


FIG. 5. PRISM nadir-looking image (left) and generated DSM using stereo pair image of left (right) observed at Himeji City, Hyogo Prefecture, Japan, on April 28, 2006.

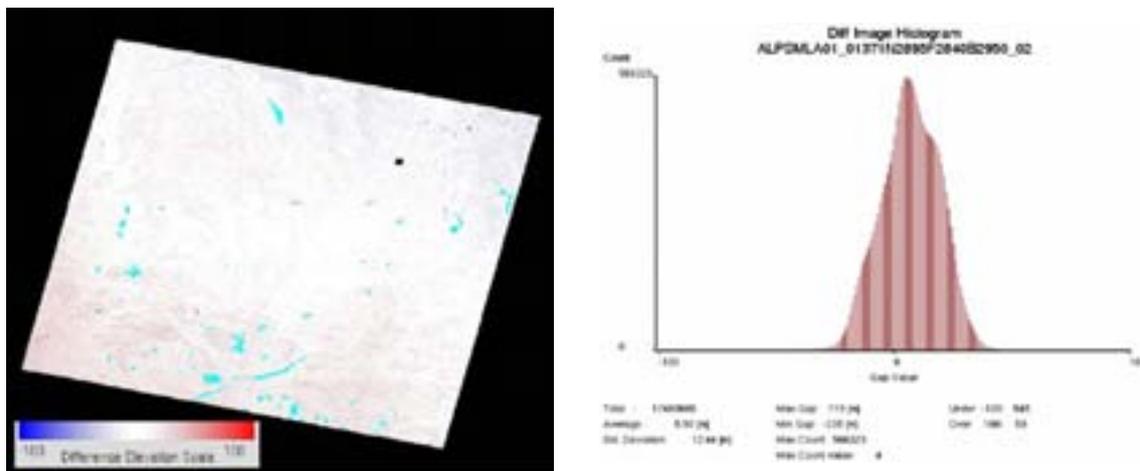


FIG. 6. Elevation difference comparing Fig. 5 with GSI 50m DEM (left, white: 0m elevation difference, gradation from red to blue: +/-100m differences), and its histogram (right).

functions of internal and external orientations using GCP and TP, mask processing, ortho-rectified image creation, stereo image matching on a sub-pixel scale, altitude calculation, and resampling. Furthermore, it has evaluation functions, such as differences between generated DSM and existing DEM/DSM and their static values calculation. We are now trying to generate an absolute accuracy DSM using PRISM stereo pair images worldwide for test generation. Currently, 17 scenes have been processed using external orientation with GCPs and TPs because PRISM products still have large geometric errors, which are over 120 meters in line direction (Y) for nadir- and backward-looking images of PRISM, and over 240 meters in pixel direction (X) for forward- and backward-looking images [7]. The left image in Figure 5 depicts an example of a nadir-looking image of the PRISM level 1B1 product for DSM generation over Himeji City, Hyogo Prefecture, Japan, observed on April 28, 2006. The right image in Figure 5 presents an example of a generated 10m-mesh DSM in a geographical frame using the stereo pair image of left image in Figure 5. The gray scale in right image corresponds to a ground surface elevation of 0 to 1,000 m. Blue denotes mask areas due to water bodies (e.g. rivers and lakes), and red indicates cloud areas. Qualitatively, it looks quite good.

Figure 6 depicts the elevation difference comparing PRISM DSM (left image) with existing 50m-mesh DEM created by the Geographical Survey Institute of Japan (GSI) [8] to evaluate accuracy. The white color of the left image indicates that the elevation difference between PRISM DSM and GSI 50m DEM is almost 0 m; the gradations of red and blue indicate elevation differences from 0 to +/- 100 m. The right image in Figure 6 is a histogram of the elevation difference from the left one. Note the differences of spatial resolution (10m mesh of PRISM DSM and 50m mesh of GSI DEM) and

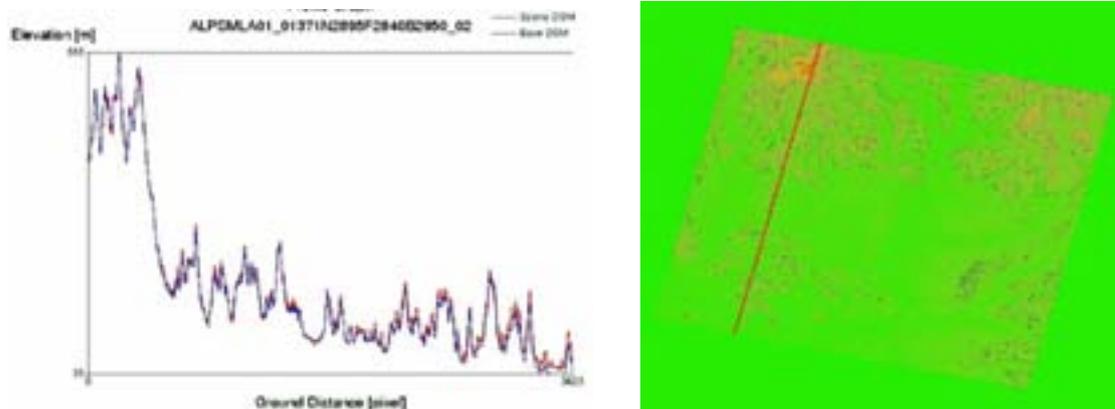


FIG. 7. Elevation profile comparison between generated DSM (red line in left image) and GSI 50m DEM (blue line in left image), and location of the profile selection (red line in right image).

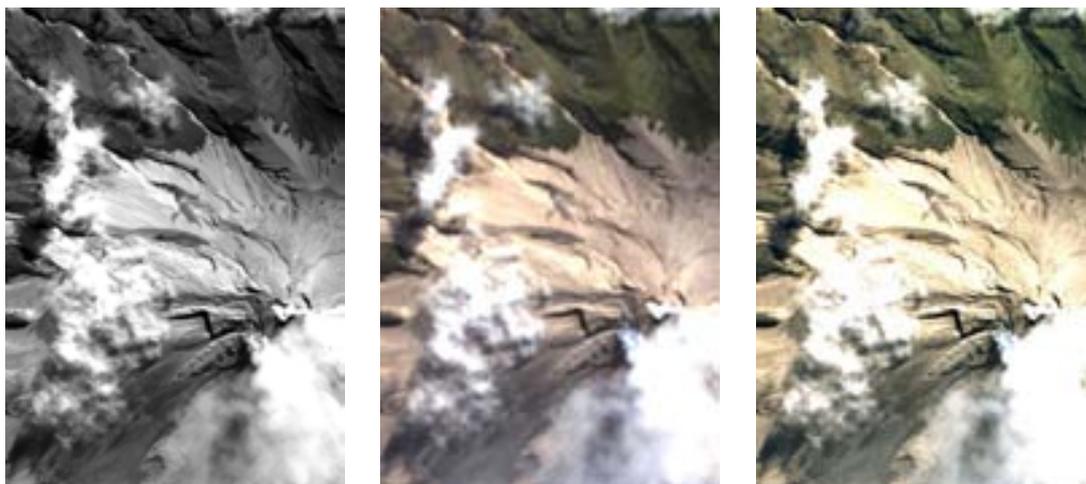


FIG. 8. Monitoring volcanic activities of Mt. Merapi, Indonesia, observed by ALOS on 12 June, 2006. The left image is by PRISM, the center is by AVNIR-2, and the right is pan-sharpened image using PRISM and AVNIR-2.

definition of elevation (DSM and DEM). Despite these differences, good results of elevation estimation of PRISM DSM are evident. Figure 7 compares elevation profiles between PRISM DSM and GSI DEM. The red line indicates PRISM DSM, which is higher than the blue line indicating GSI DEM, because PRISM DSM includes the heights of such objects as buildings and trees.

As the preliminary results of generated DSM validation, the statistical values of Figure 6 are 5.9m average elevation difference, and 12.4m standard deviation comparing PRISM DSM with GSI DEM. We have already processed 17 DSM scenes worldwide. The 17 scenes' average of elevation difference is 13.6 m, and its standard deviation is 30.9 m. We have confirmed that the DSM generation software works well using actual PRISM triplet images. However, DSM generation accuracy is not yet satisfactory. More investigation is necessary to improve the geometric accuracy of standard products, attitude information, and accuracy of DSM generation.

3.3. Pan-sharpened image using PRISM and AVNIR-2

The pan-sharpening technique can be applied to ortho-rectified images of PRISM and AVNIR-2. Figure 8 show images simultaneously observed PRISM and AVNIR-2 over Mt. Merapi volcano, Java Island, Indonesia taken on 12 June, 2006. The left image in Figure 8 is acquired by PRISM with 2.5m spatial resolution; the center one is by AVNIR-2 with 10m resolution. To process pan-sharpened image, the natural color image was firstly generated using AVNIR-2. Second, the image was transformed into Hue, Saturation and Intensity (HSI), and the Intensity was replaced by the PRISM image that was corrected geometrically using GCPs or TPs. The Hue, Saturation and Intensity data were then reversed into a color image. As a result, a virtual 2.5 m ground-resolution color image was obtained as the right image in Figure 8. The damaged area caused by ashes from the volcanic eruption and volcanic smoke can be identified more clearly.

4. Summary

In this study, we introduced the ALOS research and science program, which were including preliminary results of calibrations and validation of generated PRISM DSM, and example of high-level products using actual acquired images. We are now performing intensive calibration and validation work during the initial calibration phase (ICP), which is conducted over a period of 8.5 months after launch. The determination of satellite attitude is not precise at the moment; it is directly affected by geometric accuracy of standard products. Therefore, we must continue to conduct calibration and validation work during ICP. We are also organizing an international Cal/Val and science team (CVST) for ALOS, in an effort to make these activities more effective.

As the preliminary results of calibration and validation, the accuracy of generated DSM by PRISM triplet images is currently 30m standard deviation. This is strongly dependent on the geometric accuracy of standard products that still has large errors as offset components, as well as orbit and attitude determination accuracies. Therefore, analysis of more acquired images during ICP is necessary. The pan-sharpened image looks very well if the images are simultaneously acquired by PRISM and AVNIR-2. We expect it can also be used for safeguard purposes.

ACKNOWLEDGEMENTS

The authors wish to thank our colleagues of CVST, in particular Geographical Survey Institute (GSI) of Japan, for providing reference data and for their many collaboration efforts in activities related to calibration and validation for ALOS. We would like to continue this cooperative work until we have achieved a successful of ALOS mission.

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Potential applications of synthetic aperture radar (SAR) satellite imagery to nuclear safeguards

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Abstract. Particular strengths of synthetic aperture radar (SAR) satellite imagery are highlighted and potential applications of this unique all-weather, day/night information source to safeguards are described and illustrated by examples. The potential applications can be discussed in three categories: *infrastructure analysis*, *surface change detection*, and *subsidence mapping*. Examples are drawn from the literature and from experiments carried out under the Canadian Safeguards Support Program on proxy sites. Many new unclassified high resolution SAR satellites will soon be launched, and their impact on the state-of-the-art in satellite imaging for safeguards is estimated. We conclude that radar imagery can provide day/night all-weather information that is complementary to high resolution optical imagery. We further conclude that radar imagery can provide some unique information that is difficult to obtain in any other way.

1. Introduction

Today, all unclassified SAR satellites in regular operation are C-band or L-Band, using approximately 5.3 gigahertz and 1.27 gigahertz centre frequencies respectively for radar pulse emissions. The best available resolution for unclassified satellite radar sensors at time of writing is offered by Canada's RADARSAT-1, with 8 metre resolution. Within one year, 3 m resolution C-band and 1 m resolution X-band (specifically 9.6 gigahertz) systems will be available.

The resolution of current commercial spaceborne radar sensors such as RADARSAT-1 permits identification of large structures (roads, dams, large ships, airstrips, bridges, large buildings). RADARSAT-2 will have significantly improved resolution (3 m), but identification of small objects remains out of reach for many targets, and the identification possible with high resolution optical systems such as Quickbird or IKONOS cannot currently be matched.

Current C-band or L-band SARs have a particular strength, however, when used in a monitoring mode where multiple observations over time can be combined using sophisticated comparative methods. Combining multitemporal observations using colour composites is a simple but effective method that has been used for many years with all types of imagery. Radar-specific techniques which rely upon the inherent radar phase measurements (so-called *coherent*, or *interferometric* techniques) are much more involved. When conditions are favourable these techniques can render startling results and reveal phenomena such as surface subsidence (on the order of centimetres) or subtle surface disturbance that are invisible to even the highest resolution optical sensors. These coherent surface change detection techniques can be applied to detection and localization of surface material diversion, indirect detection and localization of underground facilities, and observation of possible evidence of

clandestine activities through detection of subtle surface disturbance or subsidence due to tunnelling.

2. Infrastructure Analysis

Infrastructure analysis from satellite imagery is already a valuable tool for nuclear safeguards support, although current practice for unclassified work is almost exclusively based on visual interpretation of high resolution optical sensors such as Quickbird and IKONOS. Resolutions on the order of 1 m or better can be used over relatively small areas so that suspected or known nuclear facilities can be observed and analysed by experts with an understanding of the nuclear fuel cycle.

Radar imagery can be used in a similar way, but offers immunity to clouds and cover of night. The eagerly anticipated high resolution radar sensors are expected to be used for similar applications in infrastructure analysis, but will require specialized training in order to be used with confidence.

Figure 1 shows an overview of a dam site as observed by RADARSAT-1 in its FINE mode, which achieves 8 m resolution. Many significant features (e.g. dam, spillway, reservoir) are visible as labelled in the image. Figure 2 is a simulated TerraSAR-X image of the Rhine near Koblenz, Germany. At a resolution on the order of 1 m, this figure shows remarkable detailed features of various structures and objects: A stadium is apparent at the lower left; and a barge, bridges, etc. can clearly be identified.

Imagery from a radar sensor can be expected to be available nominally five times more often than imagery from a comparable optical sensor due to the ability to penetrate clouds and night. We justify this as a two-fold improvement due to the day/night capability and a further factor of 2.5 based on ability to penetrate clouds. Note that if we assume that a given point is obscured by clouds an average of 60% of the time, that leaves a 40% window for optical imaging. A radar sensor can provide imagery 100% of the time, so $100/40$ yields a ratio of 2.5 more opportunities for radar than for optical due to the cloud cover factor.

The higher frequency of X-band (compared to C- or L-band) makes the radar pulse scatter more strongly from centimeter-scale components of some targets such as the treads on tracked heavy equipment, rivets or bolts on structures, or branches in trees. Although the imagery is more comparable to optical imagery, the physics of microwave scattering is sufficiently counterintuitive that great care is needed to avoid mis-interpretation. We predict that SAR and optical imagery will be used together to provide more information at more closely-spaced time intervals in all types of weather.

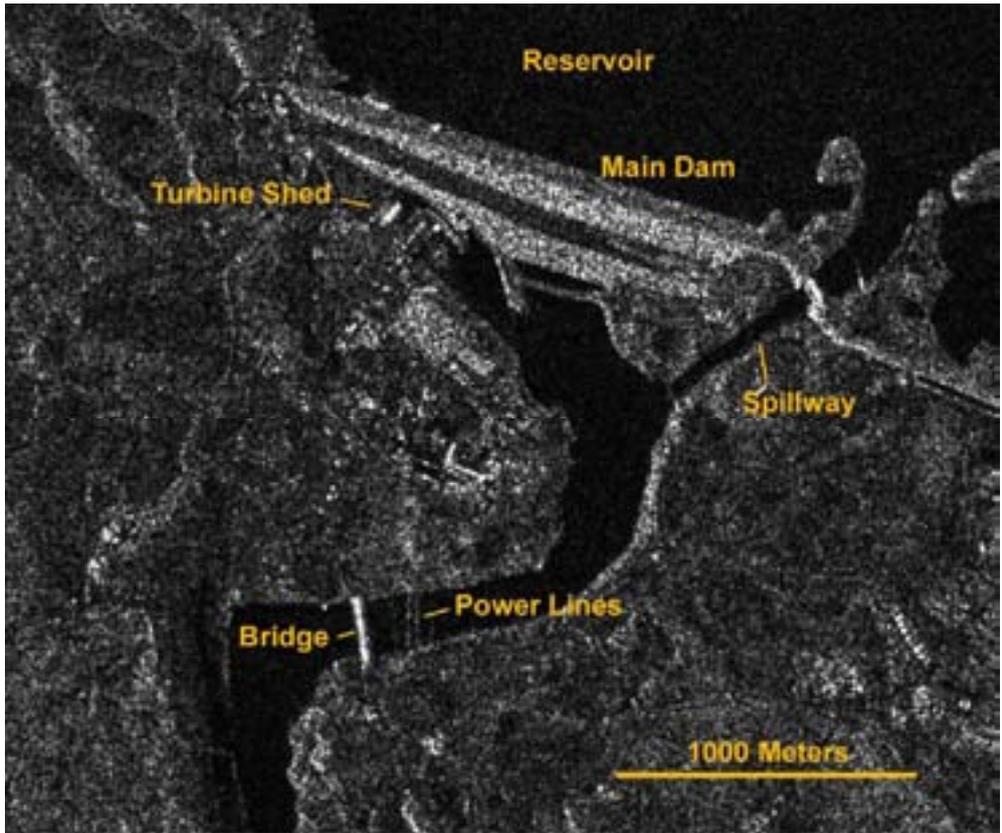


Figure 1. RADARSAT-1 image (8 m resolution) of a dam site in Iraq.



Figure 2. Simulated (1 m) TerraSAR-X image of the Rhine near Koblenz, Germany.

3. Surface Change Detection

Surface change detection is meant to refer to observation of the ground itself over extended areas at multiple times with the explicit aim of comparison over time. This can be useful because it allows observation of such things as tailings dumps, open pit mines, or natural surfaces overlaying underground tunnels or facilities. Changes can be due to direct human change to the surface (e.g. grading) or due to natural processes (e.g. subsidence due to lowering of the water table).

Figure 3 shows a very simple technique for surface change detection whereby multiple radar images acquired over time are carefully aligned and then combined by assigning different colour components to the different images. This is the so-called multitemporal colour composite technique which is widely used for optical images as well.

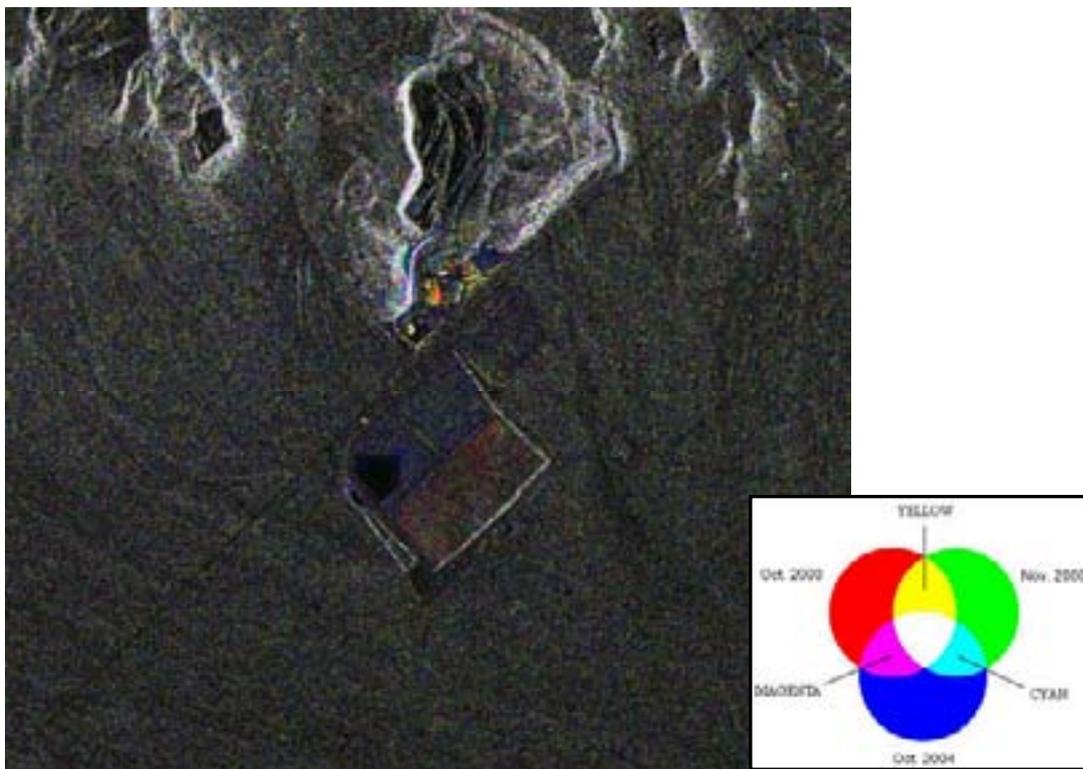


Figure 3. Multi-temporal change detection at the Bullfrog mine, Nevada, using RADARSAT-1.

In Figure 3, the various shades of black-gray-white tones indicate correlation of the radar backscatter, i.e. those features which did not change noticeably between the different dates. The colours indicate a differences in backscatter values on different dates, which can be interpreted and analysed.

Figure 4 uses the same mine site and data, but shows surface change detection results using a radar-specific analysis technique called *coherence mapping*. This technique goes beyond the backscatter strength and looks at the temporal and spatial consistency of a property called *phase*, which depends on the fine structure of the scattering objects. The dark areas on the subsidence map show where working of the mine has disturbed the surface and reduced the phase coherence. Thus the coherence map is an analysis tool for detecting surface disturbance

since it is impossible to disturb the surface and then put it back in precisely the same micro-configuration.

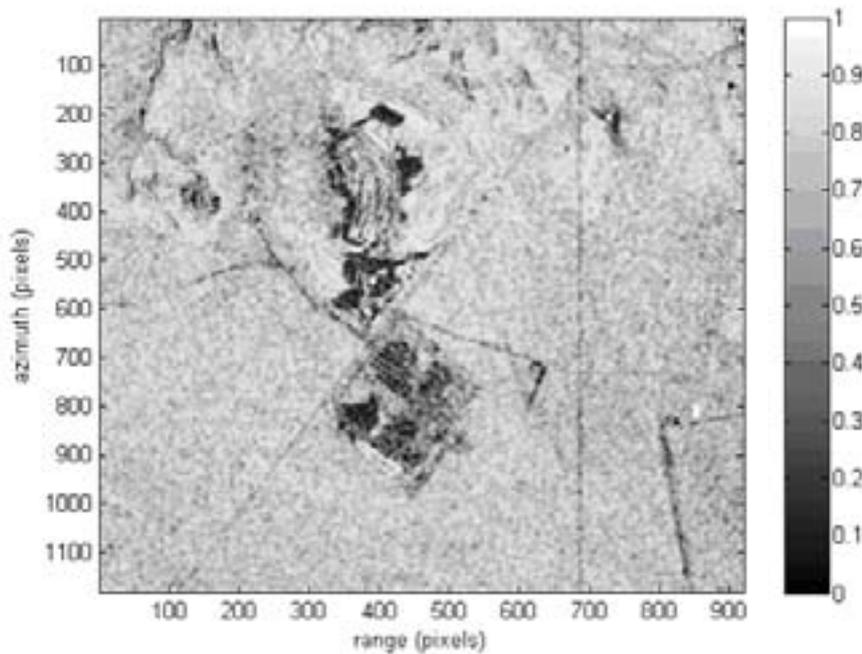


Figure 4. Coherence map of the Bullfrog Mine comparing October 2004 and November 2000.

4. Subsidence Mapping

Figure 5 shows a subsidence map produced using coherent processing techniques for the Nevada Nuclear Test Site (NTS) using RADARSAT-1 images which were taken four years apart. The colour patterns reveal subtle subsidence which is closely correlated to the individual test sites. The colour is combined with intensity from a high resolution IKONOS optical image. An explanation of how subsidence mapping with RADARSAT-1 works is beyond the scope of this paper, but can be found in [1].

The geophysical mechanism for the subsidence at the NTS is explained in [2]. The nuclear tests dating back to the early years of US weapons development have shattered the bedrock, causing a slow draining of the groundwater and lowering of the local water table. This leads to subsidence which could be used in similar scenarios as a means of localizing underground tests. Like the coherence mapping discussed in the previous section, subsidence mapping normally requires that the landscape be largely devoid of vegetation and experience very little rainfall. Work by Vincent [2,3] showed that the ERS-1 radar sensor could be used to measure subsidence at the NTS.

Both vegetation and moisture changes can add phase noise into radar images that prevent effective exploitation of so-called coherent techniques which exploit radar phase.

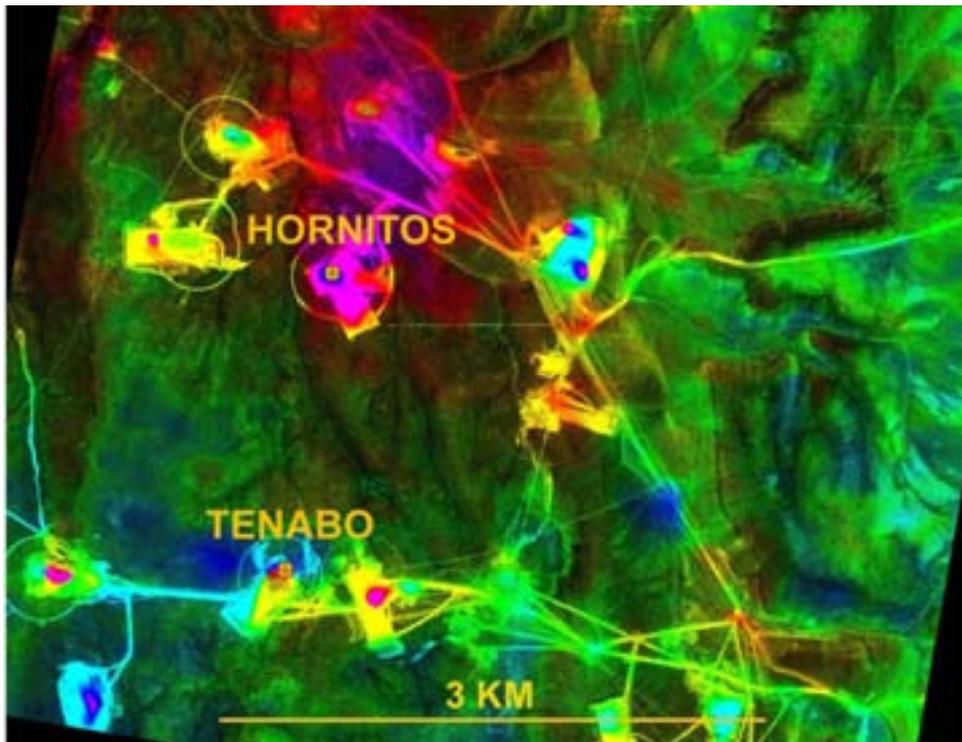


Figure 5. Subsidence signals in the Nevada Test Site for image pair Oct. 2000 – Oct. 2004, pan-sharpened colour subsidence with an optical IKONOS image.

5. Concluding Remarks

In the infrastructure analysis application the radar pictures will show objects that scatter back microwave energy such as metallic objects with appropriate geometry. The higher the resolution of the radar sensor, the more detail in the imagery and also the smaller the infrastructure elements that can be detected and identified. The impact and acceptance of new higher resolution radar imagery will be interesting to follow, since this is the first time 1 m radar imagery has been available in a non-military context. The unfamiliarity of most imagery analysts with visual interpretation of radar imagery will be an initial obstacle. High resolution radar imagery can be used of course for high resolution colour composites for surface change detection as well.

It is already clear and verified that current C- and L-band imaging radar systems can be used for surface change detection and subsidence mapping. This technology is unfamiliar to many working in nuclear safeguards, and our own work aims to develop awareness of and experience in this application. The coherent techniques provide information on surface disturbance and subsidence which is difficult to obtain by any other means.

The X-band systems have a smaller wavelength than the C-band systems (about 3 cm as opposed to about 5 cm), complicating coherent analysis techniques. This is due to greater sensitivity to decorrelating effects such as rain or wind disturbance of vegetation which can

cause the SAR phase to vary unpredictably from image to image. That being said, the usefulness of coherent techniques for C-band SAR were originally underestimated for similar reasons.

In the near future, X-band systems will likely prove useful for coherent surface change detection, and systems and data processing techniques will be refined to allow certain types of applications to work better. An example is the TanDEM-X extension to TerraSAR-X. TanDEM-X will use a pair of satellites in closely-matched orbits to allow nearly simultaneous imaging from two positions to combat possible loss of phase coherence between observations.

As indicated, the techniques discussed above can provide day-night and all-weather information for safeguards applications. For certain situations, radar imagery can provide unique information which is difficult or impossible to obtain by optical means.

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VITA-6.2: Advanced visual tool for information management

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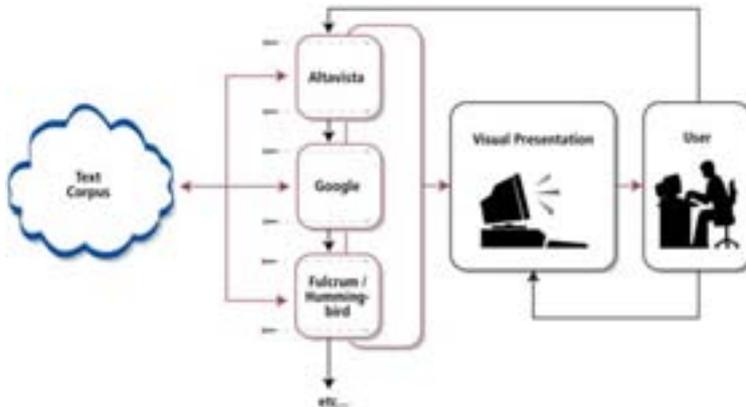
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Abstract. Visual Interface for Text Analysis (VITA), our combined user interface and meta-search engine software application, improves the quality and speed at which intelligence analysts can explore novel massive text corpora via innovations that facilitate user contextual awareness.

1 Introduction

The authors have developed a knowledge management tool to search and organize disparate and large text corpora and display them in a manner for ease of interpretation. One application would be to facilitate the process of uncovering clandestine nuclear supply networks.

1.1 VITA System



VITA sits between the user and an extended set of search engines directed at external [Web] and internal private files (Houston & Jacobson, 2002; Jacobson *et al.*, 2005). The user is easily able to search and then see a conceptual map of documents, arrayed in relationships suggested by the query. The impact for knowledge discovery, summary and presentation can be surprising.

2 Realization

VITA, a "Visual Interface for Text Analysis", is a tool used to direct computer-based document searches. It allows an analyst, via mouse and keyboard action, visually to locate, organize, summarize, and present documents of potential interest. It has control features that allow visual clustering of like documents, thus enabling quick refinement of the search process. The visual features of the VITA-6 series support the observation and investigation of the relationships [often unknown and unexpected] among documents. The software acts to isolate and call attention to otherwise unsuspected documents of importance in ways that other search tools cannot.

Earlier versions were first developed as a research testbed to identify better methods to visualize relevant document clusters and identifying their relationships. VITA has now stabilized into a pre-competitive design, through the interest of users in IAEA and the Government of Canada.

TOP SECRET PERSONAL TOP SECRET

1/10/1998
4/18/1998

From: IS [intelligence] Memo #: 100
To: A. IS [intelligence] Date: 4/18/1998

Subject: Proposal

We have enclosed for you the following proposal from [redacted] scientist [redacted] [redacted] regarding the possibility of helping [redacted] establish a project to enrich Uranium and manufacture a nuclear weapon. The above-mentioned proposal is as follows:

1. He is prepared to give us project designs for a nuclear bomb.
2. Source any requirements or materials from Western European countries via a company he runs in [redacted].
3. Request a preliminary technical meeting to discuss on the documents that he will present to us. However, the current circumstances do not allow for an immediate meeting with the above-mentioned. There is the possibility of a meeting with the intermediary that we have operations and good relations with in [redacted].
4. The notice listed this proposal to gaining profits for him and the intermediary.
5. The project has been given the code name of Tiziana in English [redacted] to use in correspondence and consultation.

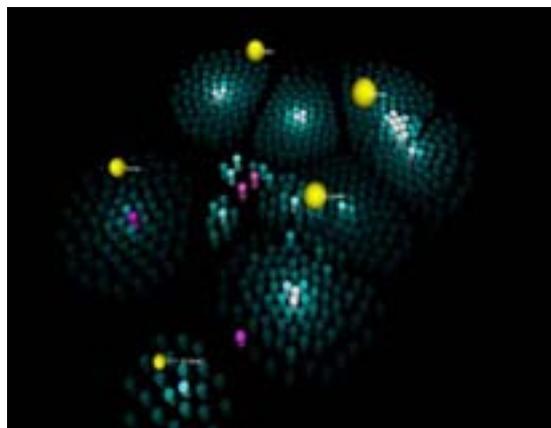


Figure 1 – VITA output

The VITA output in Figure 1 arises from a nuclear proliferation query to an intelligence-related website showed an unexpected outlier [on the left of the cluster, circled.] On investigation, it was a highly relevant item [shown on the right] not expected, that would not have stood out nor been described among the search engine’s returns in the standard lists

2.1 Meta-search.

VITA supports simultaneously querying and aggregation of public web (*Google, Yahoo!, MSN*), public news (*Google News, Yahoo! News*), local desktop (*Google Desktop, MSN Desktop*), and proprietary repositories (*dTSearch, Verity†, Autonomy†*).

VITA-6.6.2 is robust and fast, built on a C# platform that requires normal computing equipment and OS. Eight popular search engines are now supported as listed above. *Verity* will be added shortly, and XML import and export will also be made available, extending the range of uses and applications.

The interface allows a user to blend results from one or all of the supported tools into a common display. Further, it allows for incremental search refinement by integrating additional queries into the display, and pruning extraneous queries [and their associated data clouds. These operations on the display may be done interactively at run time [“on the fly”] or they may appear in new displays, as the user wishes

2.2 Clustering.

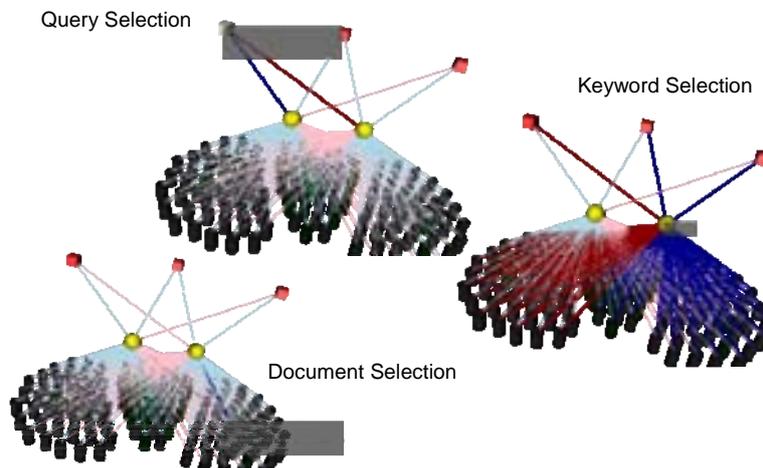
VITA shows its extracted documents clustered according to analyst queries, rather than according to search engine rankings. VITA pushes relevant documents “out of the crowd” and brings them to the analyst’s attention.

On the fly, the analyst can specify any combination of cluster information as relevant, the result of which is reflected by dynamic change in the visualization.

2.3 Visualization.

VITA employs a polished and configurable 3D visual metaphor to concisely communicate relations between multiple queries, results and extracted clusters. The primary elements are queries, terms, results and extracted clusters – each represented by distinct shapes on layered planes. The many forms of inter-relationships are displayed via shape type and size, color similarity, spatial location and proximity and as explicit connecting lines. This display of dozens of queries and 1000s of results, and their inter-relationships creates rich informational landscapes that reflect the interests of the user.

Selection Support



3 Interactive Exploration.

An analyst can make obvious the relationship between queries, their results and extracted topical clusters by incrementally specify multiple queries, with the results of each query being fully incorporated into same information landscape. Queries which clearly have no connection, can be removed or such labeled.

Flexibility to user wishes and needs is a particular feature of the present version. As now built, VITA-x is search-mechanism independent and readily conformable to new user needs and requests. To bring in a new search engine, minimal programming is needed to designate the relevant HTML or XML fields in the code returned from the search engine for input to the interface.



Commercial release is contemplated using VITA 6.2.2 as a baseline.

4 Application

Use in Counter-Proliferation. The visual features of VITA support the observation and investigation of the [often unexpected] relationships among documents. Further, it acts to isolate and call attention to otherwise unsuspected documents of importance, in ways that other search tools cannot.

VITA's features support analysis of proliferation networks: they enable the analyst to locate and identify required relevant documents in a speedy and reliable manner via its visualization and clustering functions. In addition, the aggregation function guarantees to the analyst that all of the sources available were searched. This is particularly important when searching the Internet.

VITA's visual presentation gives the analyst a good idea of the structure of the local document corpus, indicating areas of information needing more research and collection. VITA clustering further enables the analyst to arrange the both web and local documents, in a ranking that depends on the search terms. This pulls more relevant documents into the analyst's eye, even if the search engine used assigns a low ranking to them.

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Towards a strategy and conceptual framework for safeguards exploitation of optical satellite imagery

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Presented by Q.S. (Bob) Truong

Abstract. In this paper we present a framework providing a schematic map linking Safeguards problems or questions to imagery of spectral and spatial resolutions, and is intended to emphasize that it is important to “use the right tool for the job”. There are many types of satellite sensors aloft, designed to address different problems - high resolution black and white imagery is not necessarily the best or most cost efficient data source for all analyses. The number and complexity of satellite sensors are growing quickly. It may be advantageous for the IAEA to consider various strategies to fully exploit the image data available in the future.

1. Introduction

Since the introduction of commercial open-source high-resolution satellite imagery in 1999, it has become an important tool for IAEA verification of declarations, baseline mapping, detection and monitoring of changes and investigation of undeclared activities. Prior to that time, this sort of examination was done by ‘national technical means’, usually in classified military establishments. There is a growing demand for wider satellite reconnaissance and surveillance support from within the IAEA inspectorate, and the Satellite Image Analysis Unit (SIAU) at the IAEA has recently been expanded in response. This paper outlines a conceptual framework and strategy for addressing the challenges to full exploitation of open-source satellite imagery. While there are other important technologies like thermal and radar, we restrict this discussion to optical imagery in the *visible* and *Near InfraRed (VNIR)* and *Shortwave InfraRed (SWIR)*.

2. Answering Safeguards Questions with satellite imagery

The most important IAEA safeguards use of satellite imagery is to answer the question “*What is at a site, and how has it changed?*” It is now and will continue to be based on visual photo-interpretation of high-resolution (~1m) commercial panchromatic imagery to describe *objects* at safeguards sites (Figure 1). Based on their extensive experience with all aspects of the nuclear fuel cycle, SIAU analysts can interpret the imagery to *verify declarations and open source information*, and to *determine operational status of nuclear sites* of interest. It is used for tasks like *structural analysis of buildings and structures and monitoring of construction activities*. There is growing use of slightly lower spatial resolution (2-4m) multispectral imagery from the same satellites to assist in image interpretation, but this comes at added cost.

Limitations of the commercial high-resolution imagery include its rather high cost, the time consuming nature of the manual interpretation, and the fact that these sensors do not image continuously. Rather, they image very small, localized areas on demand (referred to as ‘tasking’ the sensor). They therefore have discontinuous archives, and with interference from cloud, it can take many months to obtain a useful scene of a target of interest. An important use of this imagery is to monitor changes at nuclear sites over time, but this usually requires many expensive task requests. The limited archive currently prevents examination of the past history of sites that have not already been examined, and there is no commercial high-resolution imagery at all prior to the launch of IKONOS in 1999. In some cases, it is possible to find historical coverage of a nuclear site in declassified military CORONA satellite imagery.

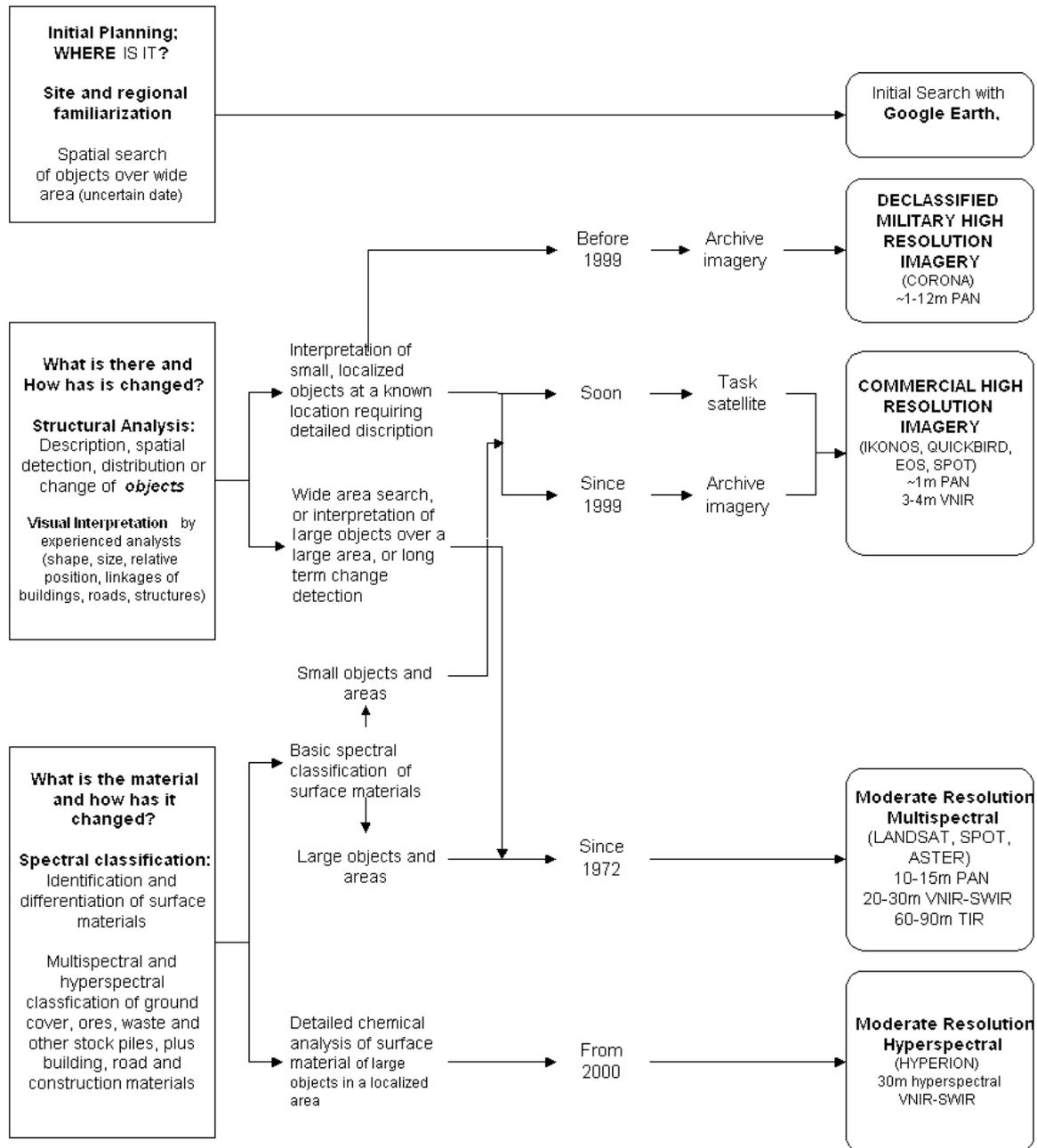


Figure 1. A framework for answering Safeguards related questions with satellite imagery.

Wide area search is not really possible with a high-resolution sensor providing scenes 5 – 10km square. Moderate resolution (10-30 m) multispectral imagery is sometimes used for these searches, where the task is to find a new undeclared site referred to in open sources. Once a candidate site is located in this lower-cost, wide-area, lower-resolution imagery, it is more closely examined using higher resolution imagery described earlier.

Since these moderate resolution sensors have wider swaths and have acquired data continuously since 1972, there are large global archives that also permit studies of *long-term change detection*. Also, since many of this family of sensors are owned and operated by governments, the cost of image data tends to be much lower than from the high-resolution commercial sensors. Some kinds of change detection and monitoring of large areas could be done with these sensors at very low cost.

As in most application areas, the use of the free online Google Earth has spread rapidly throughout the Safeguards community. This program provides a very simple interface to near true colour imagery of the entire globe, at spatial resolutions between 1 and 30 m. Many nuclear sites are included at sufficient resolution to permit visualization of objects the size of a car. It is a very efficient tool for *wide area search*, and *geographic familiarization* of a new site. It can also be used for initial planning of a site visit, but it should be noted that the Google Earth site is not secure, and that the dates of the imagery presented are uncertain, usually at least 3 years old.

In Canada, we have been working to develop safeguards applications of moderate resolution (30 m) hyperspectral satellite imagery. This data holds the promise of *chemical analysis of surface materials*. Our ability to positively identify and differentiate materials such as ores, waste materials, and other stock piles is improving quickly. This allows us to, for example, compare the mineralogy of the active part of a mine, with stock piles and the high grade stock piles near the mill, and interpret the operations of the mine. There is currently only one aging experimental hyperspectral sensor in space, and it can be difficult to obtain imagery of a particular target (This is because onboard power limitations restrict acquisition to one small scene per orbit). Furthermore, the 30m spatial resolution of HYPERION currently restricts its use to relatively large targets, such as stock piles at mines and processing plants.

3. The Future

We are entering a new era in which we will have access to a rapidly growing family of remote sensing satellite sensors of *far* greater capability than currently available. Over the next decade, these new sensors will together measure in almost all regions of the electromagnetic spectrum that are transmitted through the atmosphere. We can expect more than 50 Earth Observation satellites to be launched before the end of the decade, carrying more-capable sensors with higher sensitivity, higher spectral and spatial resolution, and operating in other parts of the electromagnetic spectrum. With less expensive launches and the shift to mini-satellites, space is no longer restricted to the major powers. A large number of small countries and companies are launching their own satellites. With the duplication of sensors, there will be more chances to image any particular target on the ground, thereby greatly decreasing the time between imaging, and increasing the amount of data available. Costs will decrease, and remote sensing data and information products will be much more widely available.

3.1 Training and Capacity building

Wider use of remote sensing technology is growing slowly within the IAEA. Experience shows that potential users do not become interested in any new technology unless they understand the potential benefit, and they can see that it can be integrated with their own knowledge, practice and priorities. Until recently, very few inspectors had any experience with what remote sensing can do. During the past several years, under the sponsorship of the Canadian, Swedish and British Support Programs, the IAEA has held annual satellite awareness training courses for inspectors and other staff. There is real enthusiasm on the part of the course participants, and a growing appetite for satellite imagery support

service to assist them with routine tasks such as inspection preparation. The challenge of building awareness is being resolved by this training. Issues remaining include complexity and cost, and developing new analysis methodologies.

The IAEA imagery analysts also regularly receive training in advanced techniques from a number of member states. Canada has been providing hands-on training in advanced hyperspectral image analysis.

3.2 Standardized methods

As the number and variety of sensors increases, the complexity involved in image analysis is expected to increase – especially for sensors designed for other than visual photo-interpretation. The IAEA does not have the time or capacity to undertake research and development. In Canada, we have been developing radar and hyperspectral applications for Safeguards applications. Member State Support Programs (MSSPs) are developing GIS, change detection, automatic image interpretation and other methodologies. This is a call to the wider community of MSSPs to develop standardized and automatic procedures that can be rapidly applied by IAEA imagery analysts to increase their effectiveness and efficiency.

3.3 Alternative strategies

Even with the recent expansion of the and well-developed standardized methods, it is apparent that the IAEA does not have enough capacity to satisfy the *current* demand for image based information products. In a world with high-resolution imagery of many nuclear facilities openly available on the Internet it would be advantageous to make good use of this source of free information, at least as a first look to narrow down the search for more precise information that will need to be purchased. We suggest that in order to obtain leverage on available funding and expertise, the IAEA may wish to consider a strategy that includes outsourcing some of the non-classified analysis work to MSSPs. In addition, in-house support could be made available to inspectors as “over-the-counter” service of certain simple products which may not required much preparation time from imagery analysts. Other more focused studies using advanced techniques, may well involve use of vetted contractors with security clearance.

4. Concluding Remarks

Our message here is that while the use of high-resolution commercial satellite imagery continues to grow at the IAEA, this is only one small part of a very wide range of remote sensing capabilities that could be more fully exploited. The number and range of remote sensing and geographic technologies continues to increase dramatically. We can expect further “release for public use” of sensors and data types that were classified military technologies in the very recent past. There will be a increasing need to test and demonstrate these new capabilities, and to determine their potential application for safeguards. High resolution imagery will continue to be important for the IAEA, but a rapidly growing number of other sensor types will provide important supplemental information and answers to different questions than can be answered with manual photo-interpretation. There is also a need for development of new analysis methods, training for analysts on the new methodologies and training for inspectors on the most up-to-date sensors and analysis capabilities. The size of the job may require that the IAEA look outside of the SIAU for certain kinds of technical support that can be developed to accommodate IAEA security policy. Now that high-resolution imagery of most nuclear facilities is freely available from open-sources such as Google Earth, it may be advantageous for the IAEA to consider other strategies tailored for various applications to fully exploit the imagery data available currently and in the future.

The IAEA illicit trafficking database programme: Trends and patterns in confirmed incidents involving nuclear material

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Abstract. The IAEA Illicit Trafficking Database (ITDB) programme was established in 1995 and is part of the IAEA's nuclear security programme. The scope of the database covers incidents of illicit trafficking and other unauthorized activities involving nuclear and other radioactive materials. The ITDB Office facilitates the exchange of authoritative information on incidents among the participating States and provides analysis of threats, trends and patterns to support States in combating nuclear trafficking and strengthening nuclear security.

To date, 92 States have joined the ITDB programme. As of December 2005, a total of 827 confirmed incidents involving nuclear and other radioactive materials were recorded in the database. Of these, just under one-third has involved nuclear materials. The frequency of the reported incidents involving nuclear materials and the quantities involved has decreased since 1994. The quantity of high enriched uranium (HEU) and plutonium (Pu) encountered has been small compared with the amounts required for a nuclear explosive. However, trafficking in HEU or plutonium continues to be of concern, and it is possible that small quantities of the seized materials could be samples of larger quantities available for purchase or at risk.

Information reported to the ITDB since 1993 indicates vulnerabilities mainly at the pre-conversion, conversion and fuel fabrication/storage stages of the nuclear fuel cycle. Some cases show that enrichment or reprocessing facilities may have been vulnerable to theft. Information on the organization of trafficking groups and networks, on buyers and end-users and on their intentions and motives is limited. These aspects of trafficking require additional efforts to collect and analyse missing information. The information available on some cases includes evidence that buyers had been identified; one incident involved a member of a terrorist group. The majority of confirmed incidents involving nuclear material have occurred in European States and in the States of the former Soviet Union. In recent years, reported cases have become more pronounced in the States of the former Soviet Union and Southeast Europe. This paper provides a brief overview of the ITDB programme and examines trends in confirmed cases of illicit trafficking in nuclear materials.

1. Introduction

The ITDB is the IAEA's information system on incidents of illicit trafficking and other unauthorized activities involving nuclear and other radioactive materials. It provides essential information support to the IAEA's nuclear security programme. The ITDB's principal objective is to facilitate the exchange of authoritative information on reported incidents among States. In addition, the collected information is analyzed to identify common trends and patterns, assess threats and evaluate weaknesses in material security and detection capabilities and practices. Communication with participating Member States is maintained through the network of national points of contact (POC). Meetings of the POCs are organized to review the database operations.

V. Turkin and W. Hammond

The ITDB covers incidents involving unauthorized acquisition, provision, possession, use, transfer or disposal of nuclear and other radioactive materials, whether intentionally or unintentionally, with or without crossing international borders. It also covers unsuccessful or thwarted acts of the above type, the loss of materials and the discovery of uncontrolled materials.

This paper focuses on nuclear materials and is based solely on the information provided or confirmed by States. No unconfirmed media reporting is included, but it is worth noting that, as of the end of 2005, information had been collected on a further 120 incidents allegedly involving nuclear materials, which had been reported only in open sources. Information, or otherwise, that the incidents were accurately reported in the media is awaited from the States involved. The credibility of many of these reports is questionable, but it is likely that the number of cases reported by States is an understatement of the real situation.

Not all nuclear materials represent a direct security risk. High enriched uranium (HEU)¹ and plutonium (Pu) may be suitable for direct use in nuclear explosive devices with little or no additional processing. Nuclear material in the form of low enriched uranium (LEU), depleted uranium (DU), natural uranium (NU), and thorium (Th) require extensive, technically complex processing to be used in a nuclear explosive device.

Incidents involving HEU and Pu thus deserve maximum attention, but all illicit trafficking in nuclear material is significant. Analysis of such incidents contributes to understanding illicit trafficking patterns and may point to vulnerabilities in the systems of nuclear material protection, accounting and control.

Aside from the nuclear terrorism risks and nuclear security vulnerabilities, nuclear material trafficking should also be viewed as a potential short cut to nuclear proliferation. Combating nuclear trafficking, therefore, should encompass a complex of nuclear security and safeguards related activities, including the collection and analysis of information. The IAEA Office of Nuclear Security (NSNS), which maintains the ITDB, coordinates collection and analysis of nuclear trafficking information with the Division of Safeguards Information Technology (SGIT). Likewise, this report has been prepared together by NSNS and SGIT.

2. Frequency of nuclear material incidents 1993-2005

As of 31 December 2005, the ITDB contained 250 confirmed incidents involving nuclear material. Of these, 12 involved HEU, four involved Pu², 67 involved LEU, and 170 involved source material such as NU, DU and Th. For six cases, the reported information was not sufficient to determine the type of nuclear material involved.³

¹ High enriched uranium (HEU) is uranium enriched to 20% or more in the ²³⁵U isotope. Low enriched uranium (LEU) has a ²³⁵U isotope concentration higher than the 0.71% concentration present in natural uranium (NU) but less than 20%. Depleted uranium (DU) has a lower concentration of the ²³⁵U isotope than NU.

² In 26 additional incidents, Pu was involved in the form of Pu radioactive sources, such as Pu ionization or Pu-Be neutron sources.

³ These figures total more than 250 because some incidents involve more than one category of nuclear material.

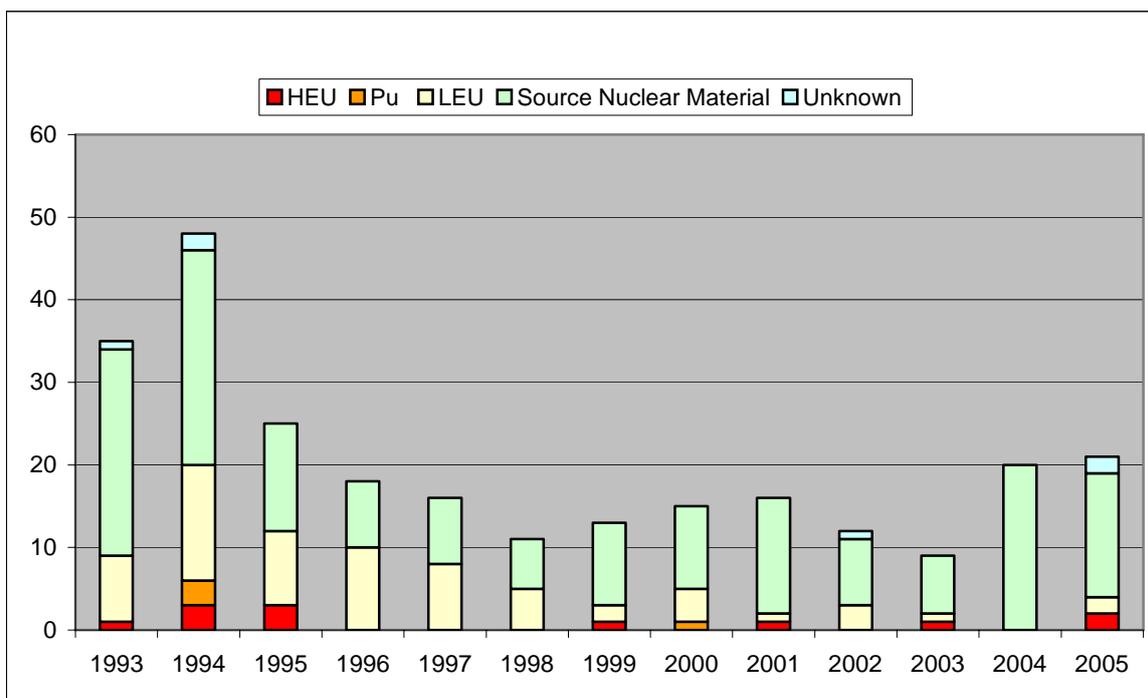


FIG. 1. Confirmed incidents involving nuclear materials, 1993-2005.

3. Assessing quantities involved

In terms of quantities of nuclear material trafficked, two distinct periods can be observed. During 1993-1997, larger quantities were encountered than were later seen in 1998-2005. The number of confirmed incidents with nuclear material dropped by about a factor of two between these two periods, and the quantities involved have dropped by even more.

Assessing trends in quantities of nuclear material in trafficking incidents requires some caution. Various nuclear materials differ in their relative significance for direct or indirect use in nuclear explosives. Certain incidents, however, stand out in terms of the type and quantity of nuclear material seized.

3.1. Pu and HEU

A few reported incidents have involved seizures of kilogram quantities of weapons-usable nuclear material. During the period 1993-95, large seizures were recorded in the Russian Federation and the Czech Republic, including the 1994 seizure of 2.9 kg of 90% HEU in St Petersburg and the 1994 seizure of 2.7 kg of 87.7 % HEU in Prague.

Even though other confirmed incidents with HEU or Pu involved much smaller quantities, it appears likely that some of these materials may have been samples of larger quantities available for illicit purchase. For example, 0.795 grams of 87.5% HEU seized in Landshut, Germany turned out to be a sample of the same material as the 2.7 kg of HEU that was seized in Prague. In addition, gram quantities of material with the enrichment level identical to the Landshut and the Prague seizures were later seized in Ceske Budejovice and Prague in 1995, provoking the question of whether larger quantities of HEU were in illegal circulation.

Since 1995, there have been no reported incidents involving kilogram quantities of HEU or Pu. Recent cases have involved much smaller quantities, such as the seizure of 10 grams of HEU 72.7% U-235 in Rousse at the Bulgarian-Romanian border in May 1999, the seizure of 0.5 grams HEU 72.7% U-235 in Paris in July 2001 and the June 2003 seizure of 170 grams HEU 89% U-235 at the Sadahlo border crossing between Georgia and Armenia. Information available for these cases is still quite limited, but it points to the continuing concern that larger quantities may be available for illegal sale or at risk of

V. Turkin and W. Hammond

theft. It is particularly notable that the HEU seized in France and Bulgaria appears to have come from the same source and might have been samples from the same larger quantity which is still at large.

3.2. LEU and source material

The majority of confirmed incidents with nuclear material recorded in the ITDB since 1993 have involved LEU or source material. During 1993-2005, there were 67 confirmed incidents involving LEU. The quantities seized have been considerable, but the material is of less significance for nuclear proliferation or terrorism.

During 1995-1997, seizures in Kazakhstan recovered hundreds of kilograms of LEU fuel pellets stolen from the Ulba Metallurgical Plant. International efforts, in cooperation with the Government of Kazakhstan, have helped to strengthen physical protection, accountancy, and control of nuclear material at this plant. Other important confirmed cases include the theft, in 1993, of a nuclear reactor fuel assembly from the Ignalina Nuclear Power Plant in Lithuania. By March 2002, most of the stolen assembly had been recovered. In February 1998, Italian police, in an undercover sting operation, seized a Triga reactor fuel element. The fuel element contained 190 grams of LEU 19.9% U-235. The seized item was one of two US-supplied fuel elements that had been stolen many years earlier from the TRICO-II research reactor in Kinshasa, Democratic Republic of Congo. The second fuel element has not been recovered. Also notable are a number of seizures of LEU in Georgia and Turkey during 1996-2005. In some of these cases, buyers have been identified, but the origin of the materials seized is not yet known.

For source material such as DU, NU and Th, 170 confirmed cases have been reported to the ITDB. Many of these cases have involved seizures of DU in the form of shielding containers for radioactive sources or NU in the form of ore concentrate and seizures of Th-containing ores such as thorianite.

The prevalence of low-grade material in reported illicit nuclear trafficking could be the result of less stringent security measures applied to these material types compared to those applied to HEU and Pu, making them an easier target for theft for those who do not know the true value of such material on the illegal market. Since 1998, illicit trafficking in LEU has decreased and stabilized. This may be a consequence of improved physical protection at LEU-related facilities, increased risks of detection, more realistic appreciation of expected rewards, increased sophistication of the trafficking chains or a combination of the above.

4. Vulnerabilities

A combination of the nuclear material type, its physical and chemical form can help identify vulnerabilities at various stages of the nuclear fuel cycle. However, a level of uncertainty is inevitable in this assessment owing to the considerable number of cases for which the physical or chemical forms are unknown. The majority of the reported incidents indicate vulnerabilities at pre-conversion, conversion and fuel fabrication/storage facilities. For example, a large number of cases involved UO₂ fuel pellets with enrichment, mass and geometry similar to those used in the RBMK- or the VVER-type of power reactors. A few incidents involved theft of fresh fuel from power reactors.

A very limited number of cases point to possible vulnerabilities at other stages of the nuclear fuel cycle, such as enrichment and reprocessing. One confirmed incident involved NU hexafluoride (UF₆). Information on two cases involving HEU shows that the materials had been irradiated in a reactor and then reprocessed.

5. Losses and seizures

In about 80% of the incidents involving nuclear materials confirmed to the ITDB since 1993, the nuclear materials were seized or otherwise recovered. The ITDB, however, does not have matching records reporting the previous theft or loss of these materials. The share of such incidents is even higher if incidents involving DU containers are excluded from the statistics.

V. Turkin and W. Hammond

Of particular concern are the incidents where HEU was reported recovered but where information on its initial theft is absent, indicating that the theft was not detected or for some reason was not reported. Since 1993, the ITDB has recorded 10 such incidents. These statistics show that stolen and unaccounted for nuclear materials may remain in illegal circulation. This situation creates considerable uncertainty for law enforcement and border control officials about what to expect in trafficking, and underscores the importance of the second line of defence and intelligence to intercept illegal nuclear trafficking.

6. Nature of incidents

About 65% of all incidents involving nuclear materials can be described as illicit trafficking in the commonly understood sense, including unauthorized possessions, attempts to sell, smuggling, malicious use of nuclear materials, etc. Such activities were particularly high in 1993-1995, averaging about 28 cases per year, and then declined. In 1996-2005, they averaged about 10 cases per annum. A range of factors may have contributed to the decline. The improved protection of nuclear materials in many States has reduced the availability of nuclear materials, increased the risks associated with their illegal acquisition and decreased chances of their theft. In addition, enhanced efforts by police and intelligence to suppress nuclear smuggling, as well as strengthened national laws prosecuting nuclear trafficking have increased the risks associated with attempts to possess, move and trade in nuclear materials illegally.

The heightened risks of illegal acquisition and possession of nuclear materials, in turn, should be viewed against the reduced expectation of reward for trafficking in low-grade nuclear materials, which have constituted the bulk of the reported incidents. As more information has become available on nuclear materials, criminals have become better informed about the true value of such materials to realise that not 'everything nuclear' can be attractive for potential buyers. The same 'educational' effect is apparently true with respect to potential buyers and end-users of the trafficked nuclear material.

7. Organization of trafficking activities

Available information on organizational characteristics of the individuals or groups involved in confirmed illicit trafficking in nuclear materials has been scant across the reported incidents during 1993-2005. The cases where such information is available show that such activities have been predominantly opportunistic, amateurish and supply-driven, with no pre-identified buyer. These types of activities are the easiest to detect because of unprofessional methods often utilized to smuggle and offer the material for sale. The reported information shows that traffickers become particularly vulnerable to interdiction when soliciting buyers. In many cases, law-enforcement and intelligence authorities were able to detect and foil trafficking operations when materials were offered for sale.

The important question is whether professional, organized and/or demand-driven trafficking occurs outside the 'radar screen' of the ITDB. This calibre of illicit trafficking undertaken by organized trafficking or criminal groups could be much more difficult to detect and intercept. Such groups could have the means and resources to acquire nuclear material and deliver it along established criminal networks to an ultimate user with a higher chance of avoiding detection or interdiction.

8. Buyers and end-users

Understanding the specifics of demand for trafficked nuclear materials is key to the assessment of nuclear terrorism risks. Available information on the majority of the reported nuclear trafficking events indicates perceived demand for nuclear materials. This perception could be based on specific information about a buyer; on non-specific information/rumour that someone is seeking to buy nuclear materials; or on the general perception that these materials have high value and can be sold on a 'black market'. For the majority of the confirmed cases, which involved intent to sell nuclear materials, no real buyers have been identified. Such cases concluded in police 'sting' operations when undercover police agents posing as buyers apprehended the sellers red-handed.

V. Turkin and W. Hammond

In one case in 2005, a member of a terrorist group attempted to acquire an unknown quantity of an unidentified nuclear material. This attempt was reportedly unsuccessful. In a number of other cases reported to the ITDB, law-enforcement officials were able to interdict smuggling of nuclear materials on the way to buyers. Information on the identity, intentions and motives of these buyers is not available.

In two cases, nuclear materials were put to malicious use; however, this was done not in the form of a nuclear explosive device but in the form of a crude radiological weapon. In 1993 and in 2005, letters containing an LEU pellet and UF₄ powder, respectively, were detected in two European States. In the latter case, the letters were addressed to a number of governmental and international offices located in the States.

9. Intentions and motives

Available information on intentions and motives behind trafficking events is scarce among incidents reported to the ITDB. The majority of incidents, for which such information is available, show intent to sell nuclear materials to the highest bidder motivated by moneymaking. Information on the buyers' motivations, where such have been identified, is even less known. In one case, for example, the reported information indicated that the buyer intended to acquire nuclear material for re-sale motivated by profit making. However, the end-user of that trafficking case could not be identified. The aforementioned incident, which involved an attempt to acquire nuclear material by a member of a terrorist group, leaves little doubt about the intentions.

10. Regional distribution

The majority of the reported incidents involving nuclear materials have occurred in European States and in States of the former Soviet Union. During 1993-2000, these regions accounted for almost 95% of all reported cases; during the following period of 2001-2005, their share of the total decreased to about 80%.

Within this group of the dominant regions, the earlier reporting was almost evenly spread among the three identified sub-regions, i.e. Western Europe, Eastern/Southeast Europe, and the republics of the former Soviet Union. During 2001-2006, however, the share of Western European States significantly declined from 30% to about 13%, while the share of the States of the former Soviet Union increased from 32% to about 52% of the regional total. In addition, within the sub-region of Eastern/Southeast Europe, the latter period has seen an increase in the share of incidents, which occurred in Southeast Europe.

The shift in the real or perceived demand for nuclear materials may have contributed to the change in the sub-regional patterns in Europe and States of the former Soviet Union. Enhanced efforts of the Western European States to suppress nuclear trafficking are likely to have increased risks facing traffickers and deterrence. Also, improved detection and interdiction capabilities in the States of Eastern/Southeast Europe and the former Soviet Union have also led to the improved interdiction of trafficking cases at their earlier stages.

Statistics showing regional distribution should be taken with care. ITDB membership is not universal, nor is it equally distributed among regions. For example, whereas Europe and the States of the former Soviet Union are almost completely represented in the database, many countries in Africa and Asia do not participate in the ITDB programme. This is likely to effect reporting from these regions, although this effect is not expected to change the formed regional patterns significantly. This conclusion is supported by information from open sources.

11. Conclusion

Since 1993, reported events of illicit trafficking in nuclear materials have significantly decreased. This decrease, however, should be viewed against the backdrop of the continuing occurrence of incidents involving weapons-usable HEU. These cases indicate that larger quantities of the seized samples continue to pose a potentially significant security risk and that some facilities handling HEU may

V. Turkin and W. Hammond

remain vulnerable. Also, an unknown quantity of unaccounted for HEU may be in illegal circulation or vulnerable to theft.

Illegal activities involving nuclear materials recorded by the ITDB indicate the presence of real buyers in some cases, including, in one incident, a member of a terrorist group. However, the ITDB information on the organization of trafficking groups and networks, on buyers and end-users and on their intentions and motives is still quite limited. The ITDB Office, in cooperation with Interpol, has initiated project Geiger, with the objective to collect the missing information on these essential aspects of nuclear trafficking and conclusively assess them.

Integrated information communication technology (ICT) tool for inspectors

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Abstract. With the implementation of the measures pursuant to the additional protocol, inspectors in the field collect various types of data from diverse sources. To perform this task efficiently and effectively, they need software tools associated with a highly automated and integrated information communication technology (ICT) framework. The CIR-Mobile is an initial step in this direction. Through the secure communication of data between the field and IAEA headquarters, it provides users with remote access to inspection reports, inspection plans and seals data. In addition, users can run quality control checks against CIR data both at their workstations and on the server at headquarters. As a result, inspectors' paper work is eliminated or reduced and inspection report completion time is significantly reduced. The next steps will be to integrate under a common umbrella all in-field activities such as inspection preparation and scheduling, containment and surveillance management, environmental sampling, nuclear material accountancy and design information verification and verification reporting, thereby providing a unique, consistent, and user-friendly interface. This paper describes the work of the Division of Safeguards Information Technology (SGIT) to develop a new integrated software tool to support the work of inspectors in the field.

1. Introduction

This paper describes the efforts of the Division of Safeguards Information Technology (SGIT) to develop a new integrated software tool to support the work of inspectors in the field. The ultimate goal of any in-field software tool is to increase the efficiency of inspections by bringing decision capability to the field and/or by providing near-real time access to the data and expertise at IAEA headquarters. This goal can be achieved by implementing an integrated information communication technology (ICT) tool that would eliminate or reduce inspectors' paper work during the inspections, thereby providing more opportunity for inspectors and analysts to analyze and evaluate data for the purpose of drawing safeguards conclusions.

2. Safeguards business domains

Safeguards business processes can be categorized into the following four domains (see Figure 1):

- (a) **Verification domain:** This domain deals with the data collected during in-field verification activities and complementary access and reported by the Divisions of Safeguards Operations (SGOs).
- (b) **State declared data handling domain:** All data provided and officially confirmed by the Member States is processed in this domain. Primary focus is the accounting of nuclear material and the organization of information provided under the additional protocol.
- (c) **Support domain:** Scope of this domain includes supporting processes in the Department of Safeguards, such as planning, procurement, human resources- management and financial management.

- (d) **Analysis domain:** All data, including open source and third party information, needed for analytical and evaluation purposes are processed in this domain. The data are used, inter alia, for the evaluation of States' nuclear programmes and for determination of safeguards effectiveness.

Various software tools are being developed by SGIT to support activities in each of the domains. Although each of these tools merits separate attention, this paper provide insights into the current and future perspectives of the software tools being developed specifically for the verification domain .

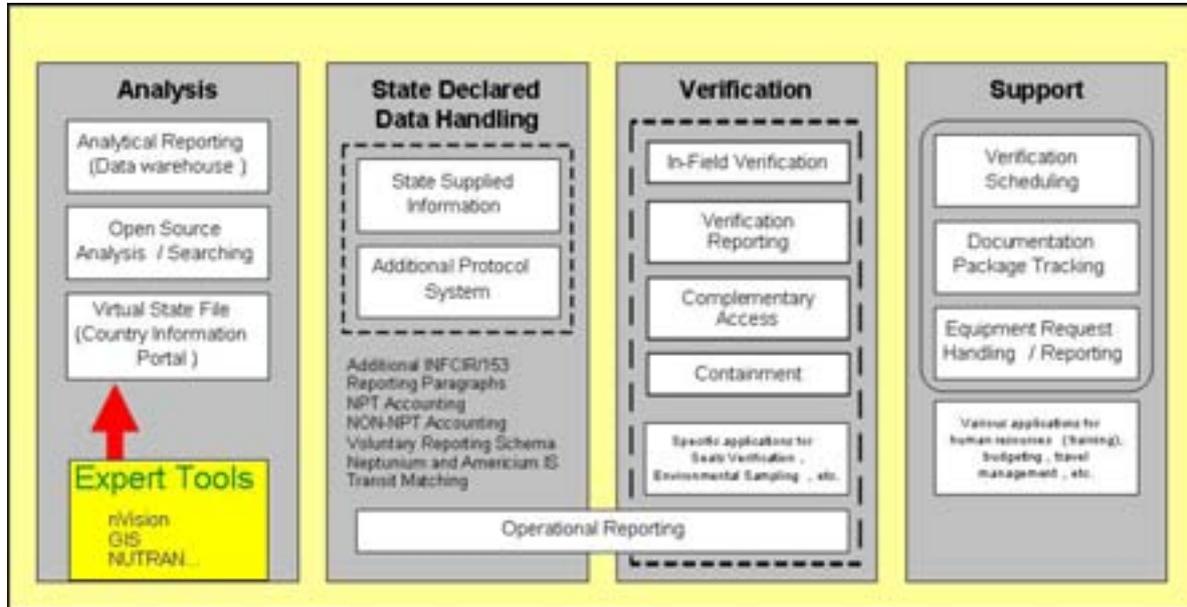


FIG. 1. Safeguards business domains.

3. Software tools currently used in verification domain

The application landscape that exists today in each of the business domains and, in the verification domain in particular, can be characterized as a set of software tools that look and behave differently from each other. Although these tools help the inspector to achieve his goals, they are not well integrated and expose their functionalities through different user interfaces. On one hand, it is not easy to adapt or extend the functionality of the software tools in order to align with the changing business requirements of the Department; on the other hand, the usability of each tool suffers.

The IAEA has developed a number of different software tools to automate tasks falling into one of the following verification activity phases: planning, preparation, performance and reporting. The Common Inspection On-Site Software Package (CIOSP), the Computerized Inspection Report (CIR) Mobile, the Additional Protocol System (APS) and the complementary access system software are examples of the applications that comprise the inspectors' 'toolkit' used for performing and reporting on verification activities in the field. The CIOSP and the CIR Mobile can interact with each other in terms of data exchange, as well as with the APS and the complementary access software. Nevertheless, these applications are separate and only partially satisfy the inspectors' capability requirements for the 'connected mode' and the 'disconnected mode', as explained briefly below.

4. Inspector IN-FIELD capability requirements

The IAEA Workshop on Safeguards Tools of the Future, held from 10-14 October 2005 in Newport, USA, identified in-field capabilities that IAEA inspectors would require in the future. The so-called connected mode capabilities fall into three categories: communications, data processing and security. The workshop highlighted the capabilities that could be satisfied with the existing and/or future technological solutions. Below is the list of major capabilities in each category.

Communications

- Access from the field to information and expertise resident at headquarters or at locations throughout the world.
- Ability to access/confirm declared facility operations using information from headquarters and the field.
- Networking infrastructure: secure, highly available, high-bandwidth communication; access to communications while inside facilities; near real-time identification and access to information and analysts; collaboration tools to facilitate information exchange between inspectors and headquarters personnel.

Data processing

- Central collection point at a facility where data are stored.
- Common look and feel of interfaces to make access to all databases easy and user-friendly.
- Easy access to database information without having to look for it or to know where it is.
- Query expert system and knowledge base.
- Imagery (e.g. pattern recognition, change analysis).
- On-site material analysis tools (destructive analysis (DA), non-destructive assay (NDA)).
- Ability to process data on-site as well as store/forward data to headquarters.
- Data review and interpretation technologies to convert large amounts of data into useful information for real time decision-making.

Security

- De-compartmentalized databases and information resident at headquarters.
- Remote access to headquarters information from the field.
- Access and process information based on access rights (roles).

Currently some of these capabilities have been implemented in the existing hardware infrastructure, such as the virtual private networking. Some software tools, such as the CIR Mobile, are already providing the functionality that is available both in the connected and the disconnected modes. The users of these tools can access their data remotely in a secure way. However, more thinking and effort are needed to implement all of the above-mentioned capabilities into one integrated, customizable and user-friendly tool of the future.

5. Safeguards integrated software tool of the future

5.1. An integrated software tool

Since all applications in the verification domain would be entirely integrated in the future, users in the SGO Divisions would work with one integrated software tool. Less human resources would be required for performing the verification activities, for example because of no double entries, electronic support of paper-based processes, one system, role-based functionality and the ability to integrate

external systems/data. An integrated tool that runs on a standardized/homogenous platform would reduce the number of applications (i.e. processes would require less applications); use one common database; provide calculation of accounting data during the data loading in the nuclear material accounting system; support user-defined reports; and assure unified access to standardized authority/location data.

Guidelines for uniform application design would be followed to develop an integrated software tool that provides an appropriate layered application architecture covering business, data, presentation and process layers. The most commonly used software components in each layer would be designed and implemented for all safeguards applications. A service-oriented architecture (SOA) approach has been chosen to support agile software development.

5.2. Reduction of data replication

The reduction of data replication would reduce the number of interfaces and complexity and improve data integrity and reliability of analysis results. Full integration would be achieved by implementing one operational data store for the applications from different domains. This would significantly improve accessibility to the data from any given application; provide transit matching and exemption analysis across different types of data (e.g. data pursuant to the Treaty on the Non-Proliferation of Nuclear Weapons (NPT), non-NPT data, data reported under the Voluntary Reporting Scheme, and additional protocol data); and make authority/location data act as a central data element that can be used by different applications.

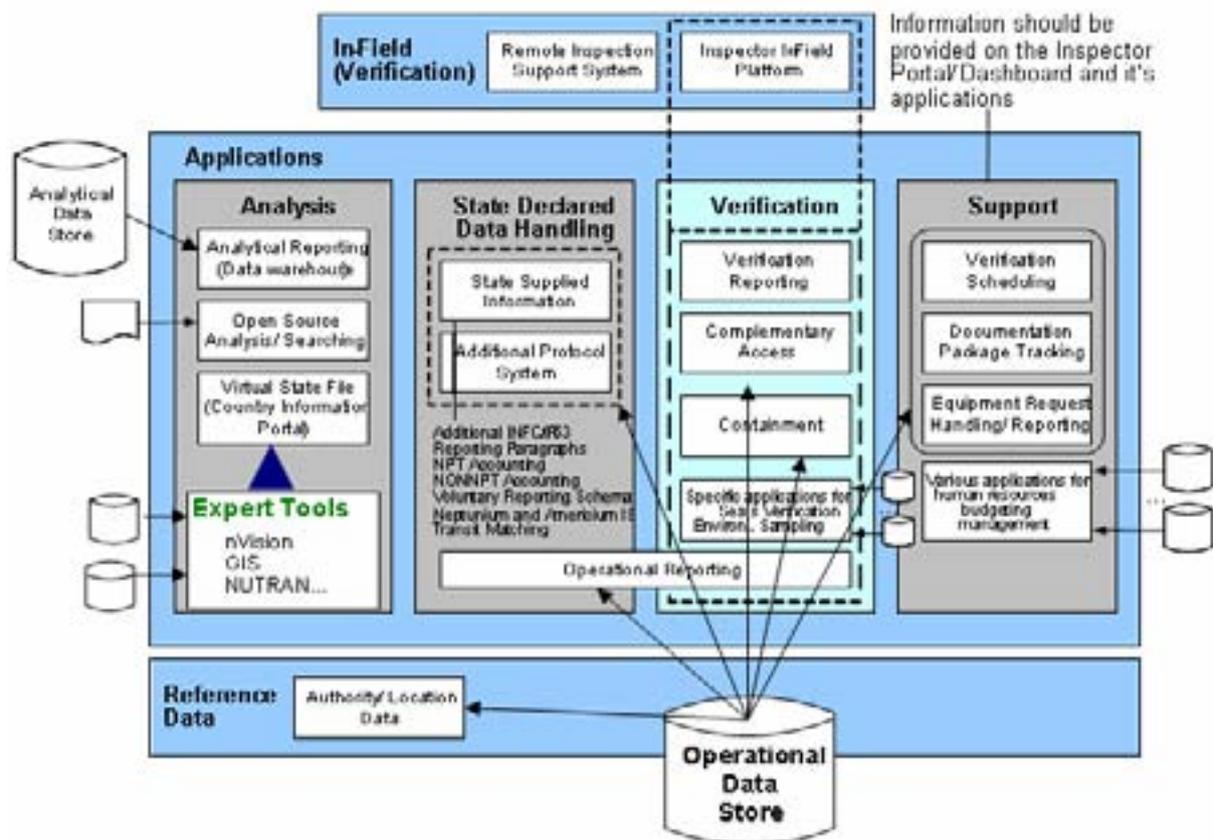


FIG. 2. Reduction of data replication.

5.3. Role-specific user interfaces

Role-specific user interfaces would provide a set of application blocks for all required functionality and information. Inspectors need a 'smart client' application solution (i.e. one that utilizes local computing resources and connects to remote servers through the Internet) due to its off-line

functionality. For connected users, both web and smart client options are suitable solutions. All applications would follow the user interface design guidelines. The consistent user interface would ensure visual and functional consistency within and across all safeguard applications. Consistency would be achieved through an enforced set of common interaction patterns for the most recurring scenarios. (See Figure 3.)

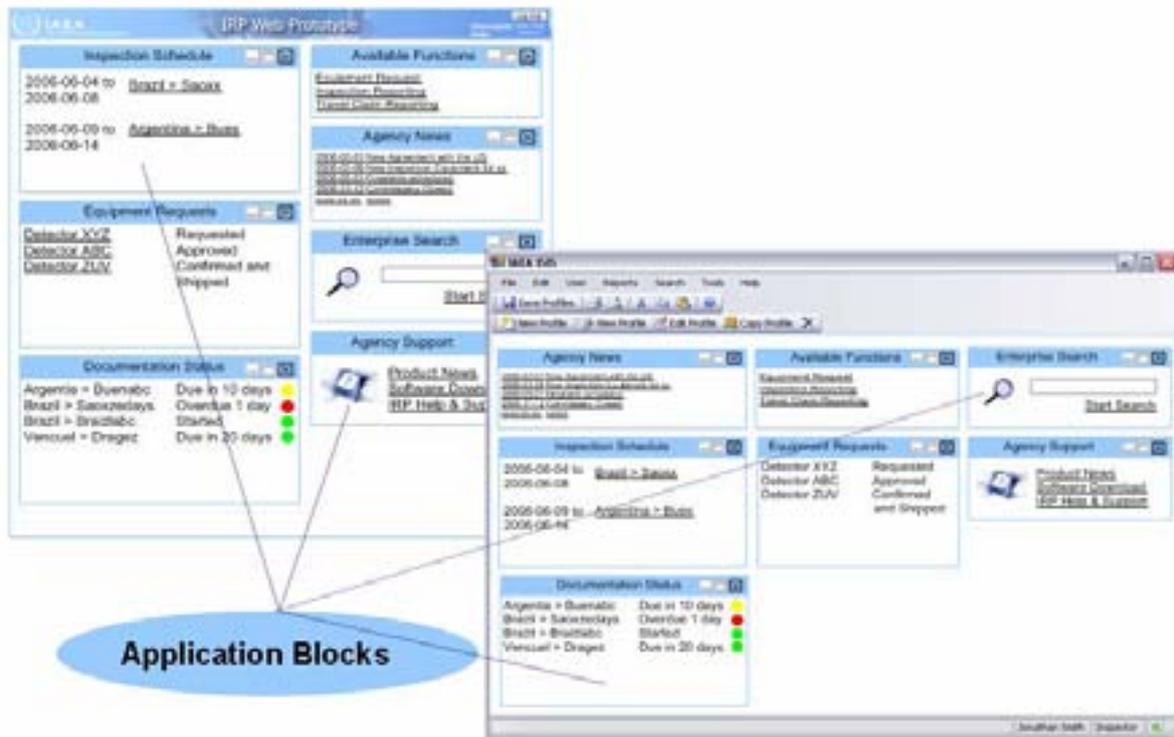


FIG. 3. Role-specific user interface.

5.4. Secure data access

Users would access confidential and highly confidential data through the system that supports their business processes (i.e. there would be no more access from separate locations/LANs). Security measures, such as separated environments, single sign-on, encryption, different views, auditing, would be implemented as described below:

- Testing, development, administration and production environments are separated and invisible to the user.
- Single sign-on: a session/user authentication process that permits a user to enter one name and password in order to access multiple applications; the process authenticates the user for all the applications they have been given rights and eliminates further prompts when they switch applications during a particular session.
- Data encryption: conversion of data into a form that cannot be understood by unauthorized people would be applied to all confidential/sensitive data.
- Different views: data can be viewed based on user profile, domain or application scope etc.
- Auditing security log: security log would record security events, such as valid and invalid log-on attempts, as well as events that are related to resource use, such as creating, opening or deleting data.

Figure 4 illustrates how the secure data access operates.

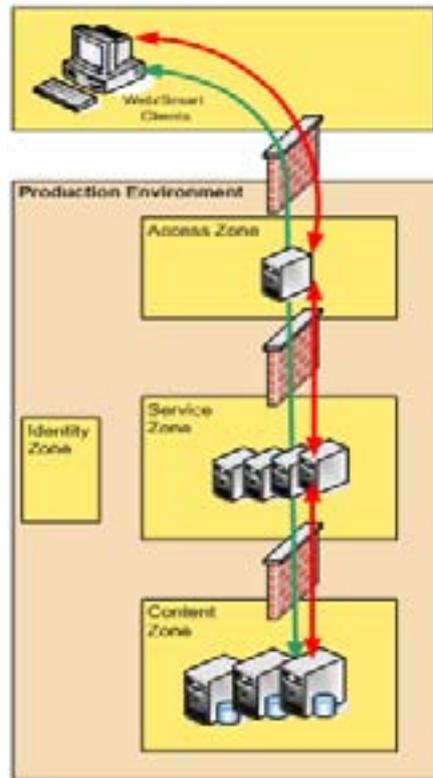


FIG. 4. Secure data access.

6. Conclusions

When implemented, the approach adopted by the ISIS Re-engineering project (IRP) and SGIT to develop future safeguards applications that integrate smoothly and provide user-friendly, consistent interfaces would allow for the following:

- (a) Provide software developers and designers with a solid foundation for agile application development; and
- (b) Provide users of the SGO Divisions with a highly integrated and usable tool that can be customized to suit different user profiles (inspectors, implementation assistants, office clerks, etc.) and different operational modes (connected, occasionally connected, and disconnected).

In sum, the future ICT tool would provide: secure access to the data stored at headquarters servers regardless of where the data are accessed from; integrated application with consistent user interface; and enhanced usability. There would be no requirement for programming skills to run business tasks, less training for users, less paper work (e.g. reports will be provided in electronic form), and shorter time for completion of reporting activities.

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An integrated approach to material balance evaluation

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Abstract. For a number of years the Statistical Analysis Section (PSA) of the Concepts and Planning Division of the IAEA Department of Safeguards has participated in the State evaluation activities preliminary to the implementation of integrated safeguards, as well as in the assessment of verification results in States where integrated safeguards measures have been implemented, through the analysis of State declarations and measurement data collected by inspectors. Two types of analyses are performed: quantitative analyses supporting the verification of the correctness of States' declarations, and qualitative analyses that contribute to conclusions about their completeness. In the traditional context, material balance analyses rest on stratified balances of nuclear material, taking into account the increases and decreases declared for each material type during a material balance period and the uncertainties associated with these inventory changes. By integrating material balance evaluation and environmental sampling results in the context of special investigations, PSA is now performing flow assessments of the chronological balance of nuclear material along the various stages of its chemical and physical transformation within and across material balance areas in a State. The term 'integrating' in this context refers to bringing together information from different sources in order to draw a conclusion about the possibility of nuclear material diversion or proliferation-related activities. The proposed concept does not affect the intrinsic principles of the independent analyses leading to environmental sampling and material balance evaluation results, but rather broadens the scope of traditional material balance studies by bringing together material balance and environmental sampling conclusions at the different stages of an investigation. The concept of integrated material balance evaluation is illustrated in this paper through hypothetical examples.

1. Introduction

Through its contribution to the State Evaluation Report (SER) process, the Statistical Analysis Section (PSA) of the Concepts and Planning Division of the IAEA Department of Safeguards participates in the process of drawing the broader safeguards conclusion required for the implementation of integrated safeguards as well as in the assessment of verification results in States where integrated safeguards measures have been implemented, through the analysis of States' declarations and measurement data collected by inspectors. Two categories of analyses are performed: quantitative analyses supporting verification of the correctness of States' declarations, and qualitative analyses that contribute to conclusions about their completeness.

The Material Balance Evaluation Unit of PSA contributes the SER inputs that pertain to the category of quantitative analyses. Trend analyses of material balance statistics are performed at the State level, to assess the overall consistency of a State's declarations and their agreement with inspector verification results throughout the history of safeguards implementation for the State. Flow analyses focus on declared movements of nuclear material (e.g. receipts and shipments) within and across the State boundaries, to check that these movements are consistent with the State's fuel cycle history. In certain cases, when the necessary information is available, further consistency checks are applied to nuclear production (NP) and nuclear loss (NL) declarations in power and research reactors, based on their power history and declared operating activities.

The environmental sampling results produced by the Inspection Measurement Quality Unit constitute the qualitative component of PSA's contribution to the SER. The principle underlying environmental

sampling rests on the fact that every nuclear process emits small amounts of material that can be found on surfaces and equipment inside facilities or outside in the surrounding area (e.g. soil, vegetation, water). Special techniques have been developed to analyse traces of material collected on swipes so as to obtain information about the process that produced them. In particular, environmental sampling can be used to detect undeclared activities, such as uranium enrichment above declared levels, plutonium and high enriched uranium (HEU) separation, irradiation programmes and clandestine operations in shutdown facilities.

By integrating material balance evaluation and environmental sampling results in the context of special investigations, PSA is now performing flow assessments of the chronological balance of nuclear material along the stages of its chemical and physical transformation within and across material balance areas (MBAs). The term ‘integrating’ in this context refers to bringing together information from different sources in order to draw a conclusion about the possibility of nuclear material diversion. The concept of integrated material balance evaluation is illustrated in this paper through hypothetical examples.

2. Methodology

2.1. *SER trend analyses and flow analyses*

The first objective of material balance evaluation activities is to verify the periodic balance established by facility operators for all MBAs and all nuclear material categories, and to determine, on the basis of a statistical analysis, if the observed shipper-receiver differences (SRD) and/or process imbalances that constitute the material unaccounted for (MUF) are due to legitimate measurement uncertainties or could be the result of loss or diversion. This analysis is currently performed on a material balance period (MBP) basis for bulk-handling facilities only and for nuclear material in quantities greater than one significant quantity (SQ). The diversion scenario covered by this type of analysis is referred to as ‘diversion into SRD or MUF’.

In addition, the balance established by the operator is authenticated on the basis of a statistical analysis of the inspector’s verification results. In other words, the evaluators verify that the balance declared by the operator corresponds with the physical situation observed by the inspectors. This analysis is also performed on a MBP basis for bulk-handling facilities only and for nuclear material in quantities greater than one SQ. The diversion scenario covered by this type of analysis is referred to as ‘diversion into D’, where the D statistic is an extrapolation of the operator-inspector differences obtained on items randomly selected for verification.

For several years, PSA has also been analyzing trends of SRD and MUF over the facility lifetime to detect trend deviations, turning points or long-term biases in order to cover protracted diversion scenarios based on strategies that would only be detectable after a number of successive MBPs. With this end in view, specific MUF components, e.g. transfer to waste (TW) and measured discards (LD), are also monitored to preclude their use as MUF tuners (artificial declarations made for the purpose of adjusting the MUF). Figure 1 shows the advantage of trend analyses for the detection of long-term MUF biases. The situation illustrated in this first hypothetical example typically gives rise to an inquiry involving actions on the part of the IAEA Safeguards Operation Division concerned and possible meetings with the plant operator.

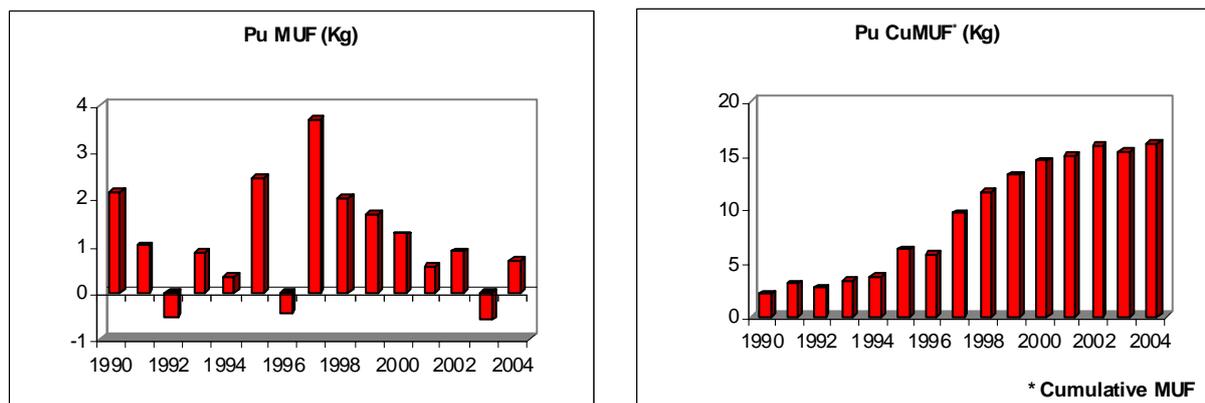


FIG. 1. Plutonium MUF trend showing a positive bias (protracted material loss).

In Figure 2, the MUF trend is compared with the corresponding trend of an inventory change component (TW) that could have been used by the operator to adjust the MUF and conceal the bias observed in the first periods.

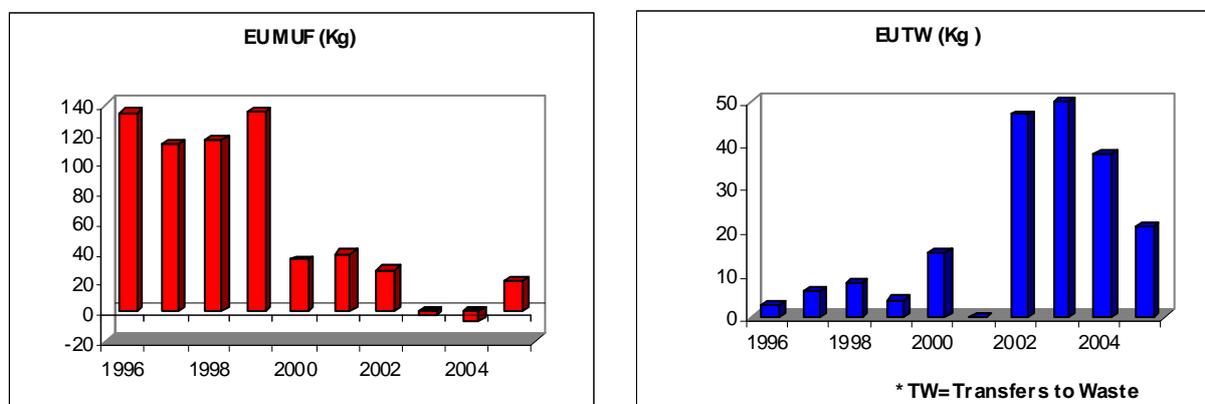


FIG. 2. Example of possible MUF 'tuning' by TW declarations.

In this second hypothetical example, follow-up actions would be initiated by PSA to determine if the increase in TW declarations observed over the last years could be justified on the basis of the plant operation.

In the framework of SER consistency analyses, similar trend analyses of the SRD, MUF and MUF components are now extended to the State level for States possessing bulk-handling facilities. In State-level trend analyses, all relevant material balance statistics are summed over the State's MBAs. The trends are represented in tabular and graphic form, keeping the MBA breakdown information in order to detect possible compensation effects and to assess their meaning on the basis of the material flow between MBAs in the State. Conclusions on the trends are given for each material category (e.g. plutonium, enriched uranium) and outstanding observations are brought forward to the responsible country officer in overall summary conclusions.

More recently, analyses of specific inventory changes declared by item facilities (e.g. reactors) have also been included to verify, for example, the consistency of ratios between U and ^{235}U nuclear losses (NL) and Pu nuclear production (NP), and to assess material flows, for example, Receipts (Receipt Domestic - RD, Receipt Foreign - RF) and Shipments (Shipment Domestic — SD, Shipment Foreign — SF), resulting from the States' nuclear fuel cycle activities. Figure 3 illustrates the principle of flow analyses. Depending on the context of the investigation, flow analyses can be performed for particular MBPs and/or facilities, to clarify specific questions or over the whole plant history to assess the consistency of the declared material flows with the State's nuclear fuel cycle and its evolution.

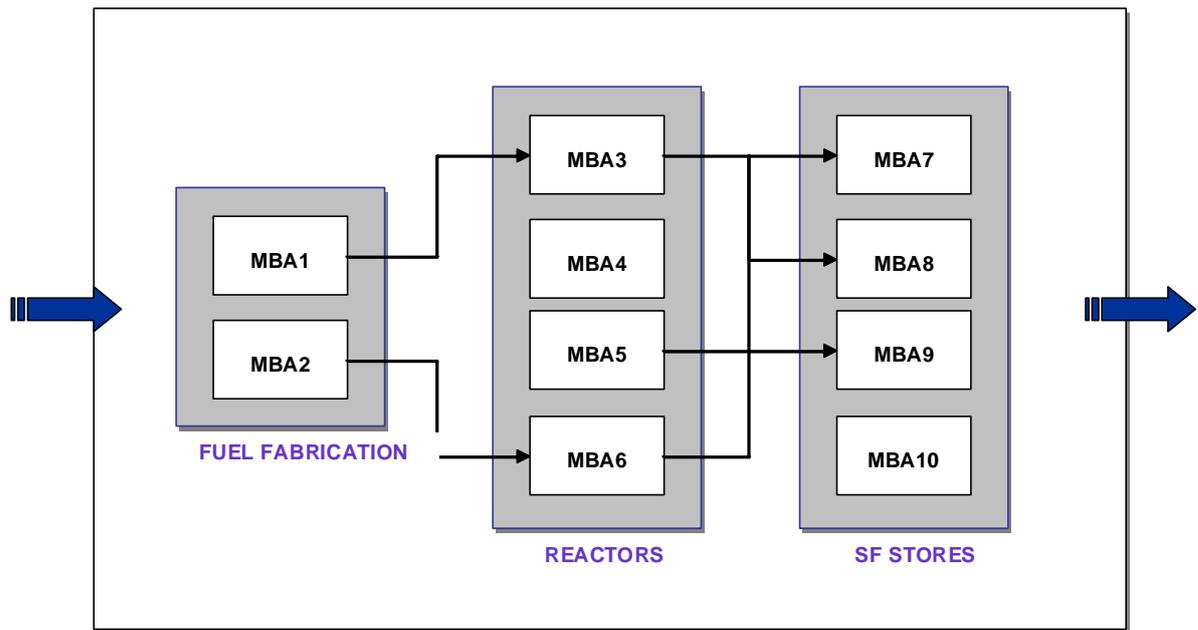


FIG. 3. Illustration of the flow analysis principle.

2.2. Environmental sampling analyses

The results of environmental sampling analysis and destructive analysis (DA) on samples taken for non-quantitative evaluation purposes are evaluated in a qualitative manner, i.e. the results of the analysis provide information about the possible existence of undeclared nuclear material or processes and hence contribute to a statement about the completeness of a State's declarations.



FIG. 4. All nuclear processes emit small amounts of material that can be found on surfaces and equipment.

Special techniques have been developed to analyze traces of material collected on swipes in order to obtain information about the processes that produced them. An example of environmental sampling results is illustrated in Figure 5, which shows the agreement between environmental sampling results from a hot cell and the burn-up correlation of the spent fuel handled.

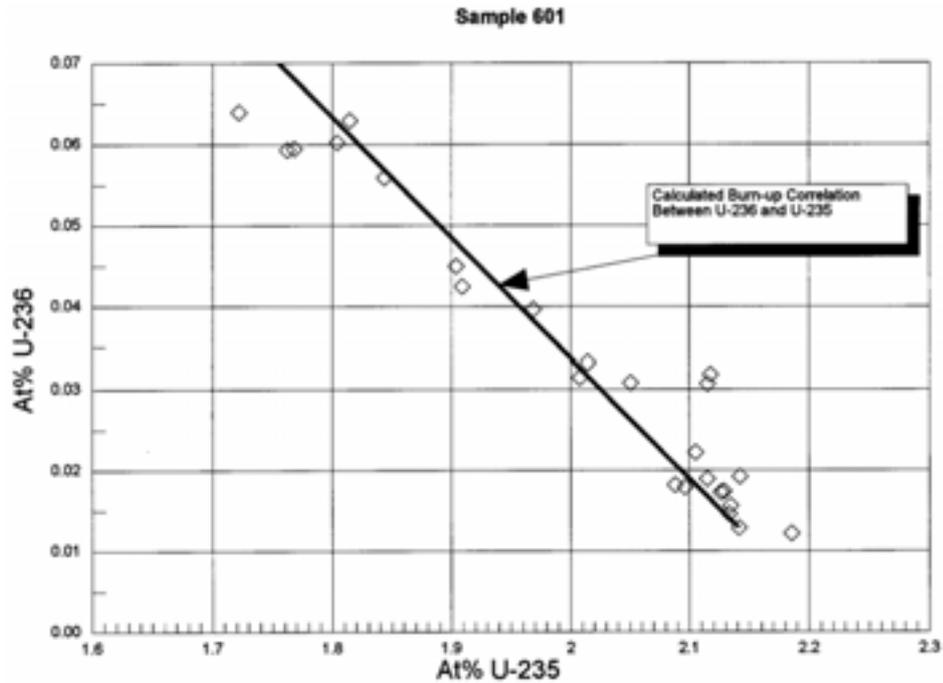


FIG. 5. Isotopic abundances measured in an environmental sample (diamond markers) and burn-up correlation.

The environmental sampling results are presently reported as separate SER inputs and are discussed with the responsible country officers independently from the trend and flow analyses described above. Section 2.3 below outlines a proposal for combining them with an extended type of material balance evaluation in a consolidated reporting and liaison framework.

2.3. *Integrated analyses and flow assessments*

In the traditional context, material balance evaluation activities rest on a stratified balance of the nuclear material present in a facility, taking into account the increases and decreases declared for each stratum between two physical inventories and the uncertainties associated with these inventory changes. The term ‘flow’ in this context refers to the movements of nuclear material across the boundaries of a MBA.

A longitudinal flow analysis of the chronological balance of nuclear material along the stages of its chemical and physical transformations within and/or across MBAs in a State, as schematically illustrated in Figure 6, is currently not performed on a routine basis. However, PSA was recently requested to perform this type of integrated evaluation in the context of special investigations supporting State evaluation conclusions. Integrated material balance studies with this extended definition are not limited to bulk-handling facilities and the lower limit of one SQ does not apply, since covert activities could take place in all relevant locations and could involve undeclared quantities of material under detection thresholds, possibly diverted from the declared flows and/or stocks. Keeping their purpose in mind and given the fact that they are more resource-intensive than classical material balance evaluations, such studies are presently applied as a priority to cases where a suspicion was triggered or confirmed by other sources of information such as unexpected environmental sampling results. This triggering/confirmation role is a first mechanism through which integration of environmental sampling and material balance evaluation activities can be implemented.

The process of performing integrated material balance studies could also be initiated by the Safeguards Operations Divisions as a result of information obtained during inspections, design information verifications (DIV) or complementary access (CA) visits. The same is true for indicators resulting

C. Norman et al.

from the evaluation of open source information, NUTRAN evaluation results¹, information obtained from satellite images and from conventional evaluation results reported by PSA.

Once the need for an integrated material balance evaluation has been established, the proposal is to identify the different nuclear material processes that can take place at a facility or at the State level, on the basis of all available information (e.g. declared activities and materials, design information, industrial knowledge, open source information). A balance of nuclear material in all intermediate forms can then be drawn along a chronological flow and matched with the information available in the State reports such as physical inventory lists (PIL), inventory change reports (ICR) and material balance reports (MBR). It can also be compared for consistency with the results of other verification activities such as DIV, environmental sampling and trend analyses for all relevant parameters at the locations or sites in question, or for the whole state. The discovery of equipment or configurations (as a result of a DIV) and the detection of traces of nuclear material or impurities (as a result of environmental sampling) that are not compatible with the declared processes lead to an iterative revision of potential alternative processes, possibly involving additional environmental sampling campaigns, until the case is resolved and the existence of undeclared activities is confirmed or ruled out.

¹ NUTRAN is the Nuclear Trade Analysis Unit in charge of the verification and analysis of information voluntarily provided by Member States on nuclear supply and procurement.

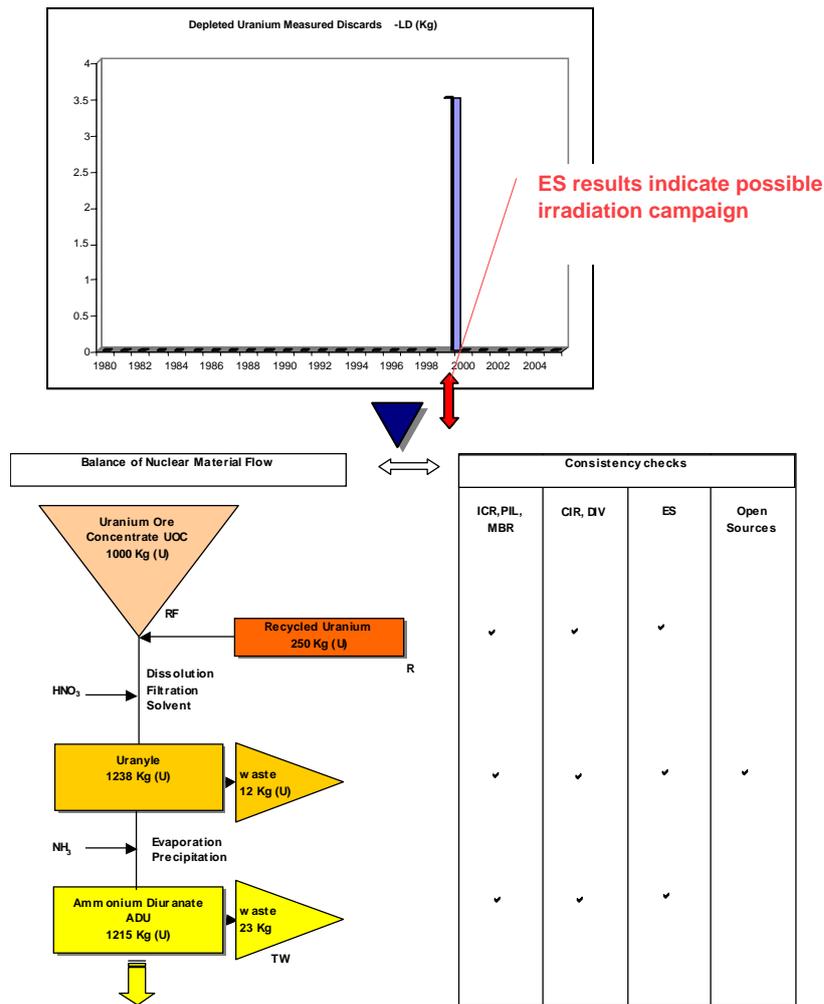


FIG. 6. Unusual trend analysis observations compared with environmental sampling results can trigger a flow assessment in a conversion plant.

3. Conclusions

The proposed concept of integrated material balance evaluation does not affect the intrinsic principles of the independent analyses leading to environmental sampling and material balance evaluation results, but rather broadens the scope of traditional material balance studies by bringing together material balance and environmental sampling conclusions at the different stages of an investigation. This approach also enhances the efficiency of communication with country officers and other sections of the Department of Safeguards, such as the Section for Effectiveness Evaluation, keeping in mind that in the present context classical analysis processes are revised in a common effort to achieve soundly based safeguards conclusions.

Since this broadened analysis principle is potentially intended for all facilities, new resources and methodologies may be needed to implement it on a regular basis, while keeping up with the increased workload of traditional evaluation activities in the strengthened safeguards system framework. In this regard, adequate training and recruitment of specialised staff, such as process engineers, should be considered. It should also be mentioned that in the field of environmental sampling the implementation of new technologies, such as age dating for uranium and impurity analysis, is being investigated.

C. Norman et al.

The first managerial steps for optimising the use of resources in preparation for PSA's potential activity expansion have been undertaken. Existing evaluation tools, including software applications, are currently being streamlined and new procedures have been implemented to improve internal communication and coordination among activity fields.

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Advances in safeguards IT security: Matching protection and actions to risks

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Abstract. Security of information entrusted to the Department of Safeguards and the protection of analyses it performs have been fundamental to the work of the Department since its creation. The technology used to support the availability, integrity and confidentiality of safeguards information has changed over the years as the risks have changed. The growing use of systems with distributed functions and the need for rapid communications and analysis of information from and in the field have emphasized the need for a comprehensive information technology (IT) security framework. The broader range of safeguards-related information being evaluated in the State-level approach to safeguards has increased the amount of classified electronic material that is being used by the Department. Technical infrastructures must therefore support levels of protection appropriate to the risks and threats, while ensuring that staff members have the flexibility and information access they need to do their jobs today and in the future.

Over the past five years, a layered approach to IT security has been put in place, including active identification of attempts to improperly access information resources. Work is underway to ensure that all devices and users are properly and uniquely identified, and that the access to information is based on roles, rather than individuals – since individuals change their roles frequently. Work is underway, in conjunction with the Member States Support Programmes, to make the network architecture more flexible while decreasing the risk of abuse, to create technical procedures and a secure infrastructure for public key cryptography and to ensure the security of mobile communications. The ISIS Re-engineering Project (IRP) is a unique opportunity and a challenge for implementing new tools and new techniques to keep electronic information appropriately secure.

This paper discusses the requirements for a consistent and thorough approach to IT security for safeguards information. It describes the work underway and the options available for future advances in security. It also describes a small number of typical applications that rely on secure information processing, and gives an overview of the approaches being implemented in the new infrastructure resulting from the IRP.

1. Introduction

“If you think technology can solve your security problems, then you don't understand the problems and you don't understand the technology” [1].

The IAEA, particularly the Department of Safeguards, is committed to providing impartial, timely and soundly based safeguards conclusions, thus providing credible assurances to the international community that States are in compliance with their safeguards obligations. It does so in an environment that is dynamic and evolving, both politically and technically. Since information acquisition and its analysis are fundamental elements of effective safeguards, and their security must be assured, the medium-term strategy of the Department includes a requirement to “ensure that an effective Departmental security system, including staff awareness, and physical and electronic measures, is in place” [2].

Recognizing that reliable secure information is a fundamental enabler for safeguards, the IAEA has significantly enhanced its access to all types of safeguards-relevant information. A revised policy on information security was introduced, and the security architecture improved. Secure links and

information channels were established to enable confidential and uninterrupted access to safeguards information for staff members in the field. The first phase of establishing secure electronic document repositories was completed. Information systems were developed and put into operation for processing additional protocol data, such as Member States submissions and declarations and data relating to IAEA analyses and complementary access activities. The IAEA acquired and applied new technologies for analysis of open source information and satellite imagery, and satellite imagery analysis became an integral component of safeguards verification [3].

The focus for safeguards IT security has been on technology solutions to protect information from unauthorized access and to ensure its authenticity. These measures have been very successful, given the hundreds (or thousands) of hacking attempts daily that arrive at the IAEA's electronic doorway. Nevertheless, it is becoming more apparent, as illustrated in the opening quote, that pure technology measures are only the basis of information security protection. Today's safeguards work needs a flexible way to ensure, in a changing environment, that the security processes are appropriate to the risk, practical, affordable and trustworthy, as well as effective and efficient. International standards for IT security management and IT security techniques, in ISO 27001 and ISO 17799, respectively, can provide a way forward, beyond 'locking the electronic doors' towards active, trustworthy and security processes that reflect the needs of the organization.

The integrated security risk management approach emphasizes the fact that IT security is not only about hardware and software. It is also about quality processes that have appropriate human and technical resources, and that are dynamic and appropriately reviewed to ensure their relevance to the changing business processes. For information-driven safeguards, it is essential to have in place not only the technology for ensuring appropriate confidentiality, integrity and availability of information assets but also that:

- (a) Information security management is part of the overall management processes in the organization;
- (b) Information security management includes organizational structure, policies, planning activities, responsibilities, practices, procedures, processes and resources; and
- (c) Information security requirements and expectations are periodically assessed, and the associated risks are assessed, so that appropriate measures can be planned, funded, implemented and reviewed.

2. The risk management approach to IT security

2.1. Theory and standards

Risk management following the ISO 27001 standard combines risk assessment with policy measures and processes to ensure that the risks are reduced. (See Figure 1.) The standard proposes four steps:

- (a) Establishment of the Security Risk Management System (SRMS), including the management framework (policies, objectives, processes and procedures relevant to managing risk, in accordance with the organization's overall policies and objectives);
- (b) Implementation and operation of the required controls and documentation;
- (c) Monitoring and review of the SRMS; and
- (d) Improvement (i.e. taking corrective and preventive actions, based on internal reviews or other information).

The objective of having an active SRMS is to ensure that the security measures that are funded and implemented are appropriate and actually mitigate the risks. It is obvious that a great deal of money can be spent on information security, from establishment of totally isolated environments, through technical measures, document handling procedures and many other prevention and detection activities.

The aim of the SRMS is to be effective and efficient, undertaking only the necessary tasks in a manner appropriate for the organization while avoiding over control and waste of valuable resources. Security should be managed like any other aspect of the organization, in a proactive manner, through awareness, planning, training, action, measurement and reporting. Security management should become an integral part of the whole management of the organization. It is not a one-time exercise, but an ongoing activity in a never-ending cycle. Well-managed security is a business enabler. Well-chosen and properly implemented controls will make a positive contribution to IAEA safeguards' credibility, not just a cost against the bottom line.

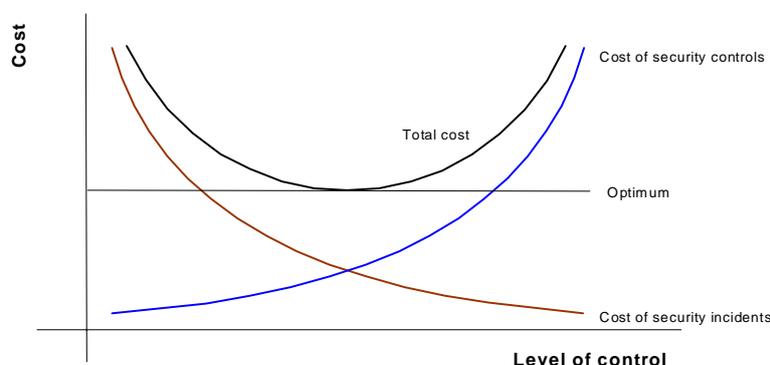


FIG. 1. The risk management curve.

The ISO 27001 standard defines the management requirements for the establishment of an information security risk management system, of which IT security is one part. The code of practice for information and IT security are given in ISO 17799, and the ISO/IEC TR 13335-3 document specifies the guidelines for the management of IT security. Together, these international standards provide a framework that needs to be applied to the IAEA's programmatic requirements and objectives. Figure 2 illustrates the various components of an information risk management programme, some of which are extremely important to the IAEA, others perhaps not so important.

Fundamental to all of these standards is the commitment of management to provide resources, in personnel and funds, that are required to: (a) establish, implement, operate, monitor, review, maintain and improve the SRMS; (b) ensure that procedures support the programmatic requirements; (c) maintain adequate security by correct application of all implemented controls; and (d) carry out reviews when necessary and react appropriately to the results of these reviews. Some of these activities fall within the usual purview of line management, others do not.

IT security cannot be left solely to IT professional staff members, including the senior IT management. It must be embedded in, influenced by and controlled through the business drivers and executive managers.

Finally, there must be an acceptance that there is always a residual risk. Choices are made. Having an SRMS ensures that the choices are not arbitrary and that the associated risks and costs are known in advance.



FIG. 2. Security components.

2.2. Requirements and practice in safeguards

The IAEA's requirements for security of safeguards information, and the resulting IT security measures, are based on policies and agreements with Member States, on programmatic requirements and on threats from the interconnected world in which we work.

2.2.1. Policies and agreements

The IAEA Statute and its Information Security Policy form the basis for all information and technology requirements. The Statute states that “staff members of the IAEA are prohibited from disclosing any industrial secret or other confidential information coming to their knowledge by reason of their official duties for the Agency” [4].

Additional protocols to safeguards agreements have been approved by many Member States, including the requirement for secure handling of information. The Model Additional Protocol statement includes a specific article, requiring the IAEA to “maintain a stringent regime to ensure effective protection against disclosure of commercial, technological and industrial secrets and other confidential information coming to its knowledge, including such information coming to the Agency's knowledge in the implementation of this Protocol” [5]. In addition to these legislative mandates, the Department of Safeguards has specified additional measures to ensure the appropriate confidentiality, integrity and availability of safeguards-relevant information entrusted to the IAEA, or obtained by the IAEA, and the analyses developed from that information.

Parallel to this policy structure, there is a structure of responsibilities, from the IAEA Information Security Officer and Information Classification Officer, through departmental representatives, and in the Department of Safeguards, a departmental Information Security Officer, a departmental IT Security Officer, and divisional information security officers. These duties are performed as one fraction of existing staff member responsibilities. As part of a risk management approach, a review of the staffing requirements needs to be performed on an ongoing basis in order to ensure that sufficient and dedicated resources are available to keep the information and IT security commensurate with an evolving environment.

2.2.2. Programmatic requirements

The State-level approach to safeguards requires a broader range of information to be acquired, analyzed, stored and reported. Significant parts of this information are highly classified and much is now in electronic form. At the same time, this information needs to be integrated into a coherent information analysis picture, in order to give an overall view of that information. Isolated information does not allow for integrated analysis.

At the same time, the speed and location of information analysis are changing. No longer can inspectors rely on total preparation in the office, data collection in the field and analysis of that data in the office. Requirements can change on very short notice, the results of data analysis can create requirements for inspection of other locations during a single trip and the IAEA needs to be prepared for inspection in locations requiring rapid interaction with remote experts. Higher telecommunications requirements are a consequence, not only for data transfer but potentially also for video conferencing and voice communications. All of these “out of the office!” information requirements must be met in a highly secure manner. As one aspect of this approach, work is underway to assess the feasibility of a Secure Global Communications Network [6].

A third programmatic requirement is staff member flexibility. The roles and responsibilities of the IAEA Secretariat staff members change frequently, particularly in the Department of Safeguards. In the past six months, over 300 individual changes in roles and responsibilities have been recorded and acted upon, in addition to the reassignment of information access privileges for most of the staff members of the Department of Safeguards' operations divisions, following reorganizations. Since information access rights are given based on country and role, as well as for support tasks, the directory of safeguards staff electronic attributes now holds approximately 50 000 entries. A role-based system of authentication is required in order to manage this complexity.

2.2.3. External threats

Information held by the Department of Safeguards is increasingly valuable to outsiders, who may try to obtain the information, change the information or prevent others from using the information. These threats could be categorized as:

- (a) Outsider threat: the threat from persons who have physical access to components of the safeguards network and who may or may not be acting under Member State direction (for example, facility operators);
- (b) Insider threat: the threat from IAEA staff members who may or may not be authorized to access the information; and
- (c) General threat: the threat from all other sources.

Processes have been put in place, both human and technological, to minimize these threats. For example, air-gapped isolated networks have served well to eliminate the risks from network-based security attacks. However, as information integration requirements grow, more information from isolated networks needs to be moved from one isolated network to another, creating additional risks that need to be addressed.

The establishment of an active information security risk management system will help to ensure that these measures remain appropriate and effective. The Government of the United States of America has mandated the appropriate application of some 170 controls, either to a low, medium or high level, to protect the information maintained electronically by its agencies [7]. The IAEA has implemented many of the same controls, with policy guidance, but without a specific system of processes supporting their continual review, planning and appropriate enhancement. In times when a simple Google™ search can highlight that the IAEA's website was specifically mentioned at a hacking conference, the IAEA must remain vigilant.

3. Current IT security activities

3.1. Present safeguards baseline IT security

In the first half of 2006, an average of 25 software and hardware security vulnerabilities were reported daily to organizations that help to manage these risks. Since 1995, over 26 000 such vulnerabilities have been discovered. These same organizations report over 4000 intrusion attempts daily as a good working figure, and approximately 90% of all email traffic as being illegitimate attempts to either gain information about individuals or to infest recipients' systems with virus, trojan horse, or other 'malware'. Experts have noted an alarming trend that spam and spyware are now being professionally manufactured by organized criminal groups. These threats cannot be ignored, but represent only the basic threat that all organizations face, regardless of the value of their information. An organization such as the IAEA, whose safeguards 'product' is soundly based safeguards conclusions and resulting credible assurance, relies heavily on the security of its information assets. A technical risk assessment, in 2002, identified fourteen projects, of which two are illustrated below.

3.1.1. Information flow control

One of the basic technical security approaches is to control the flow of data throughout the network. Since its creation, the safeguards network has been protected inside the IAEA network, with no direct access to outside networks. As can be seen below, there are now numerous connections to the outside world, and technology to protect the security of the safeguards network has therefore been increased. Since 2000, the safeguards network has been divided into trusted and distrusted portions and into purpose-specific portions. For example, safeguards servers are in a separate portion of the network, and development environments are separated from production environments. Traffic between network portions flows through firewall appliances that serve as control points, each of which has a specific set of rules to prohibit or allow specific types of traffic. In addition, tools are installed on the network at

strategic locations to detect intrusion attempts. Figure 3 illustrates the current state of the safeguards network architecture.

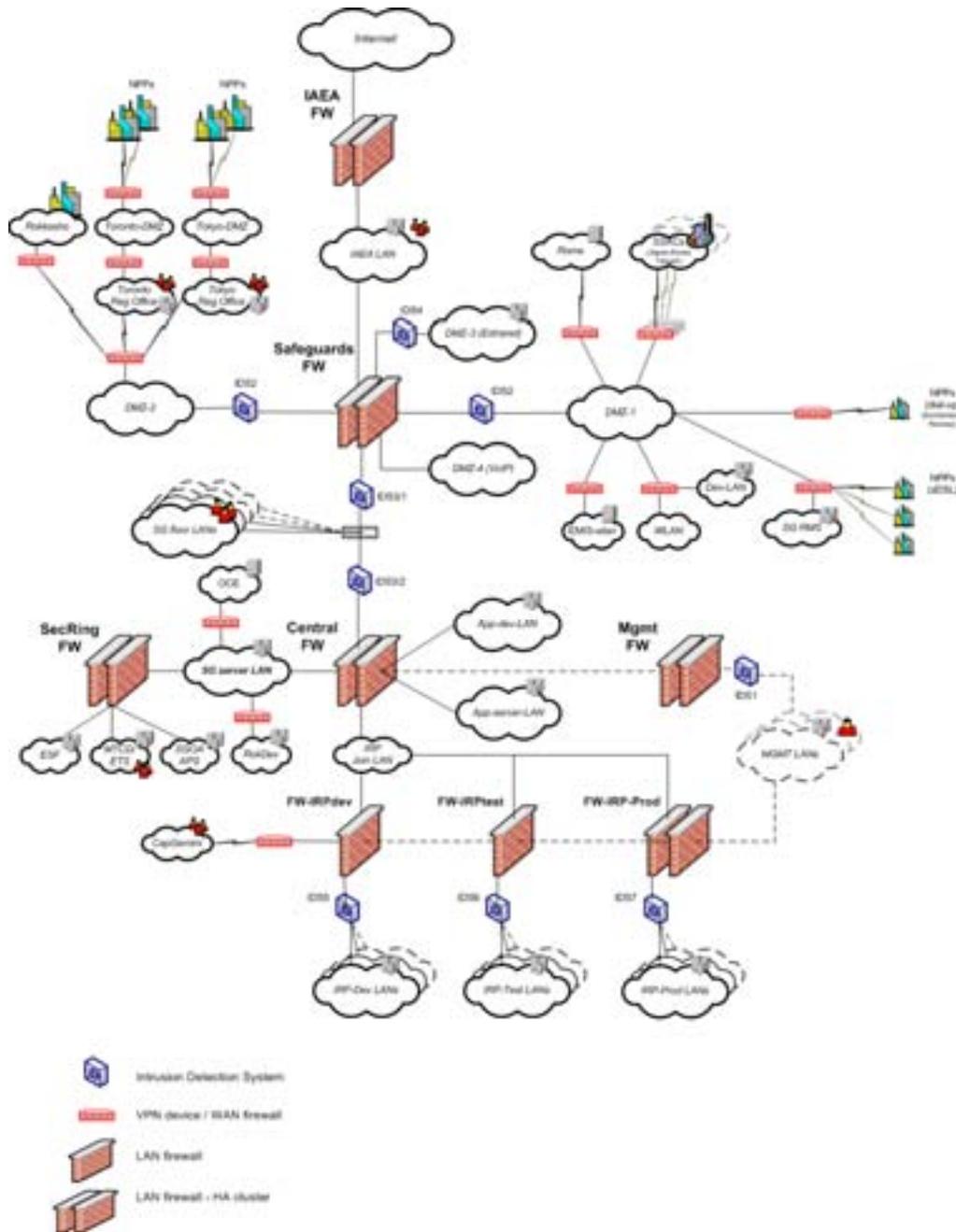


FIG. 3. The safeguards network.

The segmentation of the safeguards network is carried further in the new safeguards IT architecture, being implemented first through the ISIS Re-engineering Project (IRP). Based on the architectural principle that a user should never have direct access to the data, the various activities have been divided into different physical areas, for user access ('access zone'), for applications ('service zone') and for the data ('content zone'). Traffic from one layer to another is controlled through firewall checkpoints. The various zones of the architecture are shown in Figure 4.

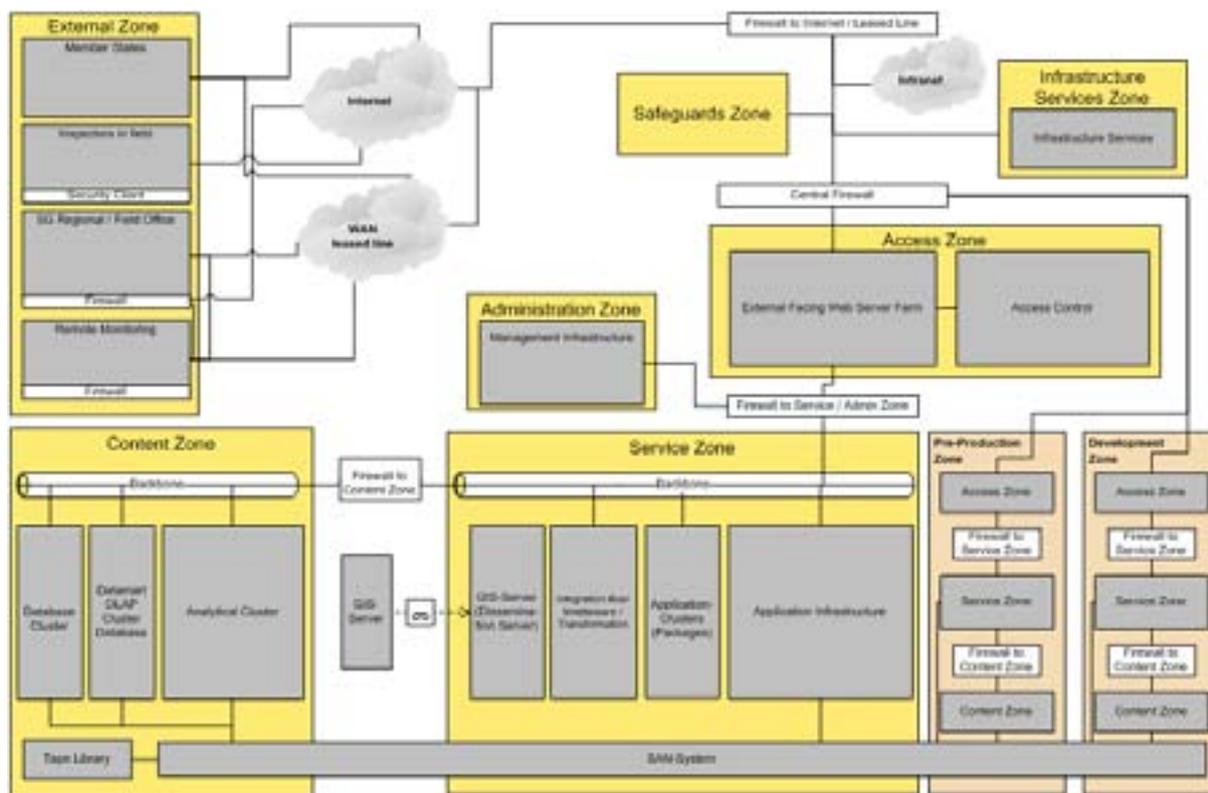


FIG. 4. ISIS technology zones.

3.1.2. Secure communications with Member States

Member States of the IAEA provide the safeguards Secretariat with a wide range of information, some of it highly confidential. Member State supplied information is provided, inter alia, to meet additional protocol requirements, to provide design information and to submit nuclear material accounting reports pursuant to the Treaty on the Non-proliferation of Nuclear Weapons (NPT). The IAEA needs to securely inform Member States about inspections, sometimes at short notice. For these activities, there are two secure technologies: authenticated and receipted electronic mail, or the implementation of shared IT infrastructure.

The process and technology for sending and receiving authenticated and receipted electronic mail are being developed as the 'secure mailbox project' with the support of the Canadian Support Programme [8]. This system will support the electronic receipt of facility operator-supplied information, and will return secure time-stamped replies to the operator and to the corresponding State system of accounting for and control of nuclear material (SSAC). This system is designed to use standard electronic mail features in order to allow the automated transfer of operator information

When there are additional requirements for the transfer of faxes and other scanned material manually, a shared secure infrastructure can be installed. This infrastructure has a cost, not only in terms of procurement but also in terms of continuing support. It does, however, provide more flexibility in transmission. Essentially, the IAEA establishes and manages a secure network segment that begins in a Member State office, typically a SSAC. The architecture is illustrated in Figure 5.

Concept of the Secure Communication

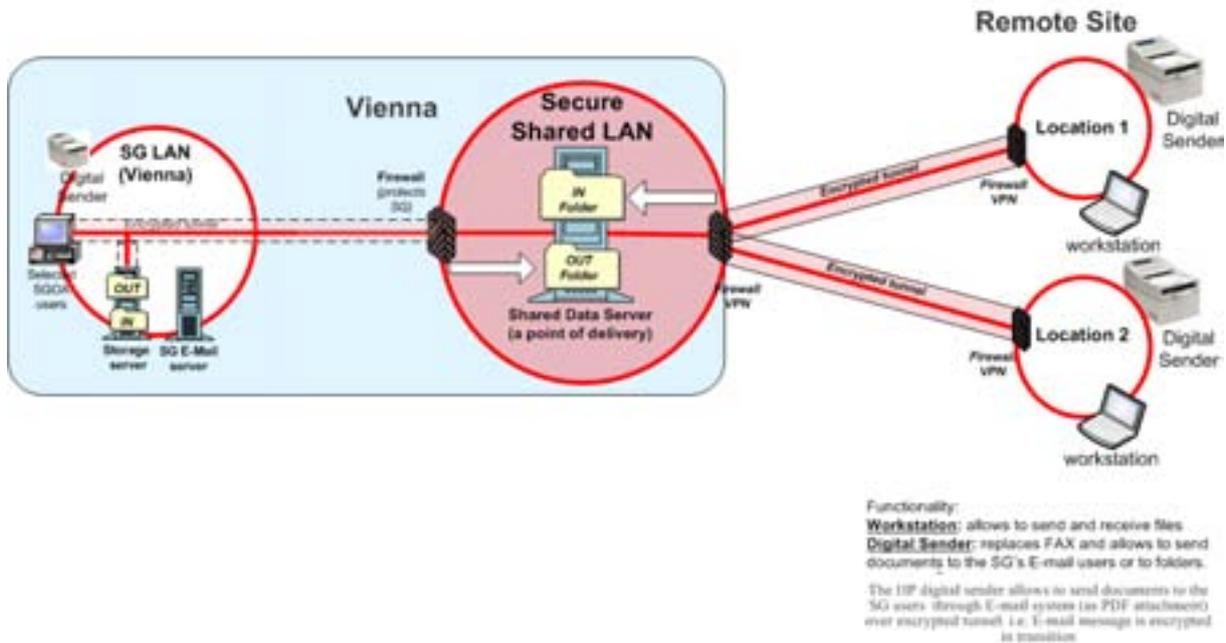


FIG. 5. Shared secure infrastructure.

4. Challenges

4.1. IT security management

IAEA and Department of Safeguards policies give specific responsibilities for IT security management to line managers. In the case of Safeguards, the responsibility as Departmental IT Security Officer is assigned to the Head of the Section for Systems Infrastructure Support. This individual, as Section Head, is responsible for the day-to-day operation of all safeguards IT infrastructure, including security infrastructure. As the Departmental IT Security Officer, this individual has the responsibility of advising on, implementing and monitoring the security of the Department's IT network and of its information held in electronic form.

The need for dedicated experts in IT security risk management and the associated technologies has been identified and work is underway to build an appropriate team that would be separate from the daily IT operations in order to design and implement appropriate controls and technical policies. Such a team will have many of the above responsibilities, including assessing security strategy and policies, performing risk assessments, developing the IT security architecture, monitoring security operations, responding to security incidents and leading IT security projects.

In the interim, the risk studies carried out in 2002 and 2003 need to be re-evaluated, especially to include the new requirements. Following an ISO 27001 approach, Figure 6 illustrates the work to be carried out.

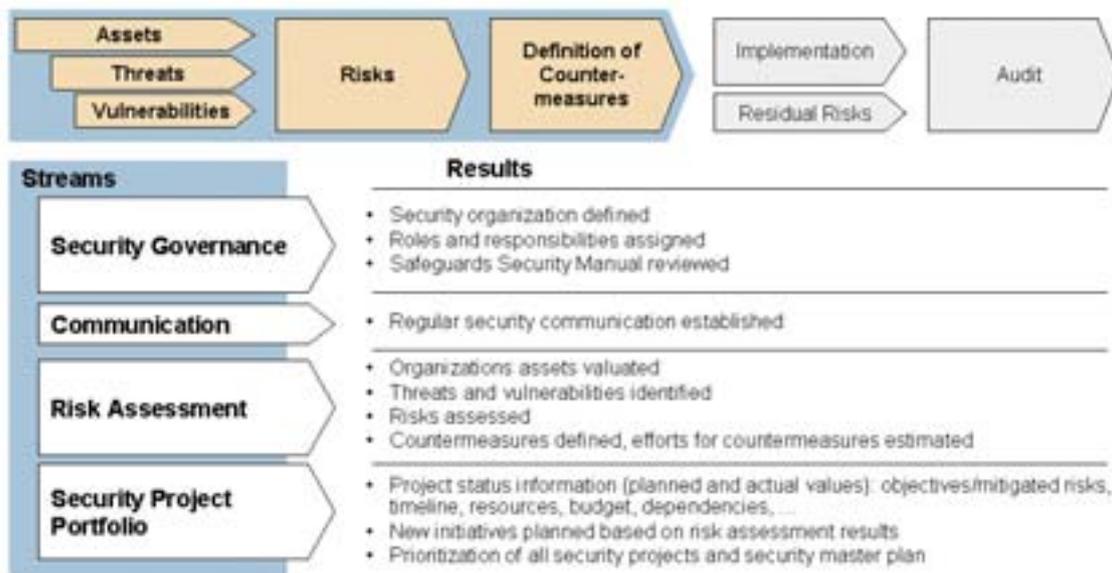


FIG. 6. Development of an information security management system.

4.2. The insider threat

It is beyond the scope of this paper to discuss the threats posed by IAEA staff members who willingly or under duress may breach the security of safeguards information. What can be addressed is the means to make such breaches difficult and to minimize the risk of accidental security breaches.

Every security paper points out that the insider threat is the most likely, and the most damaging. The causes range from the inadvertent injection of malware into the network, through virus-infected documents or browsing suspect web pages, through the browsing of public network drives, using someone else's computer, to not being kept away from sensitive information after an assignment has changed or employment has ended.

The first step in any security programme is awareness. Safeguards staff members have the requirement to take an information security course, including IT security concepts and practices, and an information security briefing is included in the training for inspectors and relevant support staff provided through the Introductory Course on Agency Safeguards (ICAS), again including both information security and IT security requirements and practices.

Another essential step is to identify the various roles, assign access privileges to roles and then only to assign roles to individuals. As noted above, the present practice of assigning specific access attributes to individuals has resulted in a quantity of access privileges that is impractical to control. Despite detailed and in many cases automated procedures, some staff members may keep access privileges for which they are no longer authorized. The IRP is defining the basic set of roles that can be used in a role-based authentication system. Products are being evaluated that (a) will support up to 500 roles, (b) grant a group of administrators the right to create roles while not having any other privileges on the system, (c) provide a role management interface, and (d) create role definitions according to the IT architecture so that they can be used in applications and IT security processes. The safeguards role-based security application also must be compatible with the IAEA-wide standards and solutions, so that Safeguards staff members can continue to work with non-safeguards computer-based information systems.

Presently, a staff member is assigned one network identity – username and password – that is used by all applications on the network. This is considered sufficient for access to the usual office services and unclassified safeguards information. The staff member is given another password, including a security token that changes every minute, for use when identifying himself or herself from the outside the

network (the so-called Virtual Private Network). A third password is assigned for access to confidential and highly confidential information, in order to decrypt it. The fourth identification is physical presence in a limited access room. The staff member's badge must be authorized (in some cases together with another personal identity number) for access through that area's door. These measures, taken together, provide different levels of assurance of the identity of the individual.

In risk mitigation terms, there is still a not negligible risk that a staff member's password may be known by other people, or a badge may be stolen, or that a network administrator may bypass some of the requirements and grant access levels higher than authorized. To reduce the first risk, passwords are changed regularly and are complex.

One of the challenges of the emerging integrated worldwide safeguards network is to ensure proper levels of identity assurance. To read the Safeguards Manual on-line, a network userid is certainly sufficient. How many identity checks should be required when one works away from the office, on highly confidential information? Where and how may such classified information be stored? Relatively small amounts of highly confidential information can be easily isolated. Today the safeguards network has almost ten stand-alone networks, plus several 'edge networks' with only very limited connection. Each stand-alone network requires special handling, not only for the data but also for identity management and authentication of the individuals and devices on that network. Support for these networks adds considerably to the data handling and IT administration tasks, which in turn increases the risk of a security breach.

4.3. Encryption and authentication

The IAEA uses many technologies to protect its data, including a number of cryptographic products. These products are used for protecting transmissions of sensitive information from safeguarded facilities, for protecting sensitive electronic mail and for protecting sensitive information that is stored in the IAEA. Increasingly, cryptographic methods are being used also for authentication, to ensure that the information in the hands of the recipient did indeed come from the purported sender and has not been altered underway.

For several years, safeguards information has been protected through a well-known encryption product that was the first on the market and has proven extremely robust. However, it has several disadvantages, including the processes for the distribution and control of the keys, and especially the fact that its keys do not conform to the X.509 international standard. As a result, while it may continue to support electronic mail encryption, it cannot be used to encrypt web traffic, or to identify devices to one another, or to authenticate access requests from software applications to data stores, or to authenticate virtual private networks. These latter applications are essential aspects of IT security today, and can best be implemented through a standard-compliant Public Key Infrastructure (PKI) and the associated products.

An IAEA-wide project is underway to create a standards-based, trusted and flexible PKI for all of the above uses. The establishment of the key management infrastructure is discussed elsewhere [9], and will not be discussed in detail here. The expected benefits of this common key management approach will include:

- (a) Less duplication and proliferation of security key products and processes;
- (b) The capability to apply security policies on the entire range of present and future objects that need to be protected; and
- (c) Simplified security operations.

The bottom line is that the IAEA is moving from an encryption technology to security architecture. Such a strategic move requires extensive planning and investment to create the security management processes and to identify the required functions and human resources. The difficult part is not the technology (although it must be done well). Rather, the most difficult tasks are drafting the certificate use policies and practices, obtaining the associated approvals and continuing subsequent day-to-day security management. Although requiring technical understanding, the development and operation of

the policy and practice infrastructure is not technical work, but rather elements of an overall information and IT security governance system.

4.4. *Integrated highly confidential network*

As noted in the discussion of the ISIS architecture above, all information required for the ISIS applications (which are essentially the entire spectrum of applications except electronic mail and simple file storage, and IAEA-wide applications) will be stored in a single database cluster, in order to allow integrated access to that information through authenticated and authorized software being run by authenticated and authorized individuals. Such integration will improve security by simplifying IT operations and minimizing external data handling. It brings with it a number of additional challenges, however.

Some of the challenges can be met through traditional IT security measures. For example, the data repositories can be encrypted (using a strongly trusted PKI) and can be securely archived. The data centre can be well protected physically and electronically. Individual segments of the data flow can be strongly controlled to prevent diversion of data through unauthorized channels.

The major challenge is how to configure the IT security to ensure information security. Presently, highly sensitive information is stored only on encrypted stand-alone systems in limited access areas. Now essentially everyone with the need to know on the entire safeguards network will have the potential to be authorized for access to that information. Where do we put the perimeter defenses? At the edge of the ISIS environment? Directly by the data? Should the IAEA have two 'safeguards networks', one for everyday use and another for ISIS use? This would be the traditional 'red net' approach, a totally separate network without client-side input or output devices, and perhaps only available in special purpose rooms. Or should the data be protected by multiple authentications on a single network? Two-phase or even three-phase authentication (e.g. 'something I know, something I have, and something I am') procedures are possible, and can be implemented at each level of data access.

Another question is how to control the use of the data. The Safeguards Division of Information Technology is implementing an architecture to prevent individuals from accessing any data except through authorized software. Security can be enforced at every stage of the software retrieval process. The challenge is then to ensure that proper controls are enforced wherever sensitive data are stored or processed, including implementing proper tracking (the equivalent of 'transmission against signature only') and restricting further dissemination through copying and printing ('no copy allowed').

One solution would be to apply a complete Digital Rights Management (DRM) regime as part of an information security strategy. DRM processes and technologies can be used to identify the usage attributes of all pieces of data, and server, desktop, and laptop software can be implemented to interrogate and respect the DRM attributes. DRM, when properly implemented, can minimize the risk that security restrictions on the use of specific information can be bypassed or reversed. The cost is high – all data must be 'tagged' with additional attributes that are carried with it throughout the information lifecycle.

Digital rights management (DRM) is a set of technologies designed to apply and enforce persistent access restrictions to digital information, as specified by the information provider. Digital rights management can regulate the types of actions that can be done with information (for example, view, print, copy or modify) and the time frame in which that information remains accessible. DRM restrictions may be identity based (e.g. "User A" can view the contents but not modify them) or apply to all users (e.g. the content can be viewed by anyone but only until the end of the month). Other examples include limiting who can view, modify, print or copy the information, when access to the information expires, and what operating platforms the information can be used on. The restrictions are persistent in the sense that they are designed to be inextricably bound to the information. DRM restrictions are unlike file-system based controls in that they are enforced regardless of the storage method or platform that the information is accessed from [10].

These are complex decisions that have significant financial, working method and security impacts. They can only be based on a risk assessment, with all stakeholder parties involved and committed to the decisions.

5. Conclusion

Today's and tomorrow's information-driven safeguards cannot function without trustworthy IT security that allows for the flexible, integrated work processes needed by the IAEA to fulfil its safeguards and verification mandate. The Department of Safeguards has invested heavily in IT security, recognizing that the appropriate handling of information entrusted to it is essential. That investment has been in management commitment, development of policies and especially in implementation of technical security measures. This work has been strongly supported by the Member States, and has been effective.

With changing requirements and threats, an ongoing risk assessment activity is needed. The 'silo security' architecture is no longer appropriate for the programmatic requirements or for addressing the real risks. Traditional access control techniques are proving cumbersome and in some cases relatively easy to bypass. IT security must be put in place to address the working needs and requirements of the Department of Safeguards and the Member States. Based on a business risk assessment, the IT security technical risk analysis can point to appropriate controls.

Implementing an information security management environment requires skilled staff members to develop, review and implement a well-defined set of information security services with performance objectives, a well-defined set of IT security controls and their objectives, an integration of IT security components into the overall IT architecture, security policies and implementation rules.

Ownership and accountability for the risk management functions must be assigned in a way that minimizes conflicts of interest and separation of duties issues. Indeed, greater levels of accountability, transparency and measurability need to be built into security controls.

The IAEA is committed to quality work, and this includes the information and IT security activities. Basing the security risk management activities on established international standards will help to ensure that the processes put in place for the protection of safeguards-relevant information are effective and can be audited and assessed internally and externally.

Concerning the technology approaches to IT security, the IRP has provided an excellent opportunity for re-assessing the requirements and establishing an appropriate architecture. However, security cannot be done solely for the ISIS applications. It needs a Departmental approach for all safeguards information.

The overarching benefit of having an organizational, process-driven standardized approach to information and IT security management is that threats and risks are assessed by the organization, not individuals, and that the choices are agreed upon. Furthermore, such an approach results in a project-driven strategy for implementation rather than fixing each weak link when it breaks.

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Creating an XML schema for enhancing nuclear safeguards information

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Abstract. In this paper we will assess the utility of using eXtensible Markup Language (XML) as a basis for data structuring with the aim of enhancing nuclear safeguards information storage, retrieval, processing and analysis. XML is intended to provide the platform from which a set of reports can be transformed into a variety of useful formats, as well as facilitating efficient structured searching and indexing across multiple documents. It is a textual mark-up language capable of describing any set of data: it is also open source and platform independent and hence a suitable format for the long term preservation of information. The XML format provides the necessary architecture for one to perform a basic level of automated analysis and visualization on the marked-up data.

1. Introduction

Information relating to nuclear safeguards can come from a wide range of sources that tend to be provided in a variety of formats and languages. There is also typically large volumes of data from which it is necessary to identify and extract the most relevant facts. These factors serve to create a substantial data management challenge for the International Atomic Energy Agency, in which the analysis and evaluation of safeguard-related information is a difficult and highly labour intensive task. In order to reduce the information processing workload for agency staff it would be beneficial to adopt a simple, low cost and efficient scheme with which to encode safeguards-related information. This would have the benefits of integrating a wide range of sources into one easy accessible data repository and as well as further freeing staff time by providing a basic level of analysis.

The use of eXtensible Markup Language (XML) as a basis for information structuring is one such way in which the agency could approach its data management challenge. XML is an international W3C standard for the exchange of data and is widely used, especially in the commercial sector [1]. It is open source, platform independent and has support for Unicode which gives it the ability to handle almost any information written in any human language. In addition it and its predecessor SGML has been in use for nearly twenty years, in this time an extensive knowledge base and variety of software packages for creating and manipulating XML have been developed.

In the XML methodology metadata is added to the original document in the form of user defined tags, in a process known as textual mark-up. Tags are distinguishable from regular text in an XML document by the angled brackets (< and >) that surround them. They have a very similar syntax to that of the more well known mark-up language, HTML. The key difference being that while HTML tags tell browsers how to display a particular page, XML tags provide information about the content of the document. For instance the title of this paper

would be marked up as `Creating an XML schema for enhancing nuclear safeguards information` in HTML and `<title>Creating an XML schema for enhancing nuclear safeguards information</title>` in XML. The `` tag tells the browser reading the HTML file to display the text in bold, while the XML `<title>` tag identifies the section of text as a title. With XML one is free to use any tags that are convenient for a given application. In a nuclear safeguards context this would involve the adoption of a scheme that would efficiently classify all elements worth categorizing.

Further examples of XML tags are given in Fig. 1 and the creation of an appropriate set of tags to describe nuclear safeguards-related information will be discussed further in Section 2

In general, modern approaches to textual mark-up have tended to be fully automated. In these schemes algorithms contained within expensive “off the peg” software packages are used to tag and categorize the information contained within a given document. While automatic mark-up allows for the classification of large volumes of data in a very short time, the accuracy of the mark-up and hence the value of the information added tends to be quite poor. Typically only 50 to 80 percent of the elements created will be tagged correctly [2]. An alternative would be to perform the majority of the mark-up manually in an approach based on accuracy rather than speed.

In this paper we propose the construction of a bespoke XML data repository for housing nuclear safeguards information. In our analysis we emphasis the benefits of accuracy rather than speed when performing the textual mark-up. This approach has considerable benefits when it comes to the manipulation of marked-up data and performance of automated analysis.

2. Method

One of the most important and useful attributes of XML is that it is extensible. This means that when applying it to a group of reports the mark-up scheme can be tailored to the project requirements. The particular scheme developed is specified in what is known as a schema. This defines a set of tags through which a document can be marked-up and a set of rules for their implementation (these go beyond XML’s standard syntax rules). To apply the XML methodology to nuclear safeguard-related information a specific schema would first need to be created. The set of tags should be able to encompass everything deemed to be of interest, for example, potential tags to describe nuclear safeguards-related information might include: companies; institutions; people; locations; nuclear infrastructure; and nuclear materials.

The user defined tags as specified in the schema are designed to divide the information marked-up into very broad categories. In order to do anything useful with this metadata it would be necessary to create a number of taxonomies in which the data can be further organized.

Taking the nuclear infrastructure tag as an example it would be useful to have a taxonomy which enabled a distinction to be made between different types of facilities. Once the mark-up is completed this would then allow a user to pull-up a list of, for example, all the light water reactors or all the radiation research centres that exist in a particular country or in all countries. This information would be automatically listed along with a link to its occurrence making it straightforward to jump to the place where it is mentioned in a particular report or set of reports.

Separate taxonomies would need to be developed for each of the user-defined tags and would most likely be hierarchical in nature. Their individual size could potentially range from

hundreds to tens of thousands of terms. At this stage a decision would need to be made as to what level of detail it is desirable to sort the data.

Once the schema and a set of taxonomies have been established the actual textual mark-up can be performed. As was mentioned in the introduction several expensive software tools exist which cater for the automatic tagging of data. These packages offer a way of categorizing an extremely large amount of information in a very short time. The major drawback of this approach, is that this can only be done with a relatively low level of accuracy when tagging. An alternative option is for the majority of mark-up to be done manually by trained personnel with some background in the nuclear safeguards field. This is certainly a labour intensive and a comparatively slow approach but one which offers a potential accuracy of close to 100 percent. The decision as to whether to opt for automatic or manual mark-up will depend on the extent to which accuracy is valued over speed, it may well even be desirable for a hybrid method based on the two approaches to be developed. It should be noted at this juncture that there are several key benefits to using the XML methodology that only become feasible when a very accurate textual mark-up of the original information has been performed. These will be discussed in the Section 4.

In order to classify a particular entity which has been tagged a unique code or ID must be created and assigned. This code should then be associated with a standardised name for the particular entity and then used in all its future occurrences. It is necessary to use a system of coding because frequently situations will occur where the same entity will be described in a number of ways, for example: International Atomic Energy Agency; IAEA; IAEA, UN. By assigning a unique code problems associated with multiple occurrences of the same element being classified as separate entities can be avoided. Recent experiences in the intelligence sector have shown that is not necessarily always a straightforward task. It is well known that one of the key problems that the US intelligence services has in tracking Osama Bin Laden and his supporters is that there are numerous transliterations of their individual names [3]. While it should be substantially easier to assign codes to the majority of entities relating to nuclear safeguards, this example serves to highlight the importance of performing this task well.

Once the information is in an XML format there exist a variety of transformation schemes designed to convert the raw XML file into a number of useful formats. Extensible Stylesheet Language Transformation (XSLT) is the best known of the XML transformation languages. In an XSLT transformation the original report is not altered, but instead a new document is created based on its content. One of the most common XSLT driven transformation is the conversion of the data from XML to HTML.

3. Immediate Benefits

The most immediate benefits of using XML as the basis for information structuring is its efficiency and simple ability to be transformed. Using XSLT or one of the XML transformation languages the marked up documents can be easily transformed into a variety of different formats. As a first stage this would probably involve a conversion of the entire repository into HTML. This could then be uploaded onto a secured server, providing everyone with the necessary level of clearance easy access to the entire repository. It would also be possible to transform the data into a format which could be conveniently downloaded to a hand-held device such as a Personal Digital Assistant (PDA). This would allow IAEA inspectors in the field to have real-time access to safeguards information, as well as giving them the ability to add to or to modify reports. The real-time updating of documents would

need to be linked into a scheme to provide version control, so that older variations of reports would remain accessible.

A probable next stage would involve the extraction of integrated indices and structured searching which would facilitate the tracking of a particular entity across the entire repository. Within the nuclear infrastructure category, for example, the indices would allow for a user to be able to access information on all of a certain type of facility mentioned in the reports that make up the repository, such as information on all heavy water reactors located in Canada. This information could be brought up in a list, with each element associated with a link allowing the user easy access to the place in the document or documents where the mention occurred. This facility is only possible because the data has been encoded using a specific nuclear safeguard schema, together with a set of tailor-made taxonomies. It is convenient to emphasize at this point that with an XML repository the complete text of the original document is preserved. This is important when it comes to analysing the context within which a particular entity is mentioned.

4. Further Applications

A data repository based on the XML methodology to efficiently manage nuclear safeguards-related data has a number of perhaps less obvious potential benefits which become apparent when it comes to the analysis and evaluation of the information.

When the data is in an XML format it becomes possible to implement a number of schemes which can increase the analytical capability of the user. The most ambitious of these involves a form of automated analysis. For this particular application a relational database would need to be constructed from the XML repository using a technique known as data mining. This could be done through the application of an algorithm, which should be able to scan through the various XML files and link associated elements together. At the simplest level this could be accomplished by linking elements together if they occur in close proximity within the original document. These links would then be used to create a network within which strong and weak relationships between the different elements would be displayed. The value of this application will depend strongly on the accuracy of the original mark-up, for instance if this is only 50 percent then every other element that makes up the network will have been incorrectly tagged. The performance of automated analysis in this scheme is only likely to be useful if the accuracy of the original mark-up is close to 100 percent and even then there are likely to be a number of missing links and false positives which will lead to the algorithm needing constant refinement. Nevertheless this potential application offers a simple form of analysis for safeguards information without the need for any further mark-up or human intervention, aside from the development of a sophisticated entity linking algorithm.

Another potential application of the XML methodology is in a geographical visualisation of the data. This is relatively straightforward to achieve and can be done by assigning to the relevant entities within the XML repository a latitude and longitude and then linking this to a mapping software such as Google Map/Earth or actual satellite imagery. This approach would allow an analyst to look at the information from a fresh perspective and give him or her the ability to detect patterns based on geographical clustering.

5. Conclusions

In this paper we have proposed the establishment of a bespoke data repository based on the XML methodology. This is designed to deal with the large volume of safeguards information that the agency is required to process. The creation of this repository would involve the

C. Hobbs and J. Kidd

textual mark-up of a large number of documents and the benefits and shortcomings of approaches based on speed or on accuracy have been discussed. It should be noted that a very accurate mark-up would be required in order to explore some of the exciting potential applications of this scheme as described in Section 4. Key stages in the development of the repository along with some potential applications have been represented pictorially in Fig. 1.

The short term benefits of adopting an approach to data management in which XML is used as the basis for structuring information are numerous. It is a simple and relatively inexpensive technique for encoding data, which is widely used and well supported. A number of schemes exist which enable XML to be transformed into a variety of useful formats. It can be used to tag entities in multiple languages and then facilitate their tracking over a number of documents. It allows for the rapid real-time updating of reports which can be linked into a scheme to provide version control.

Longer term benefits which have been suggested include using the structured format of the data to provide automated analysis and geographical visualisation. Although this is only likely to be feasible if the initial mark-up is performed with an accuracy close to 100 percent.

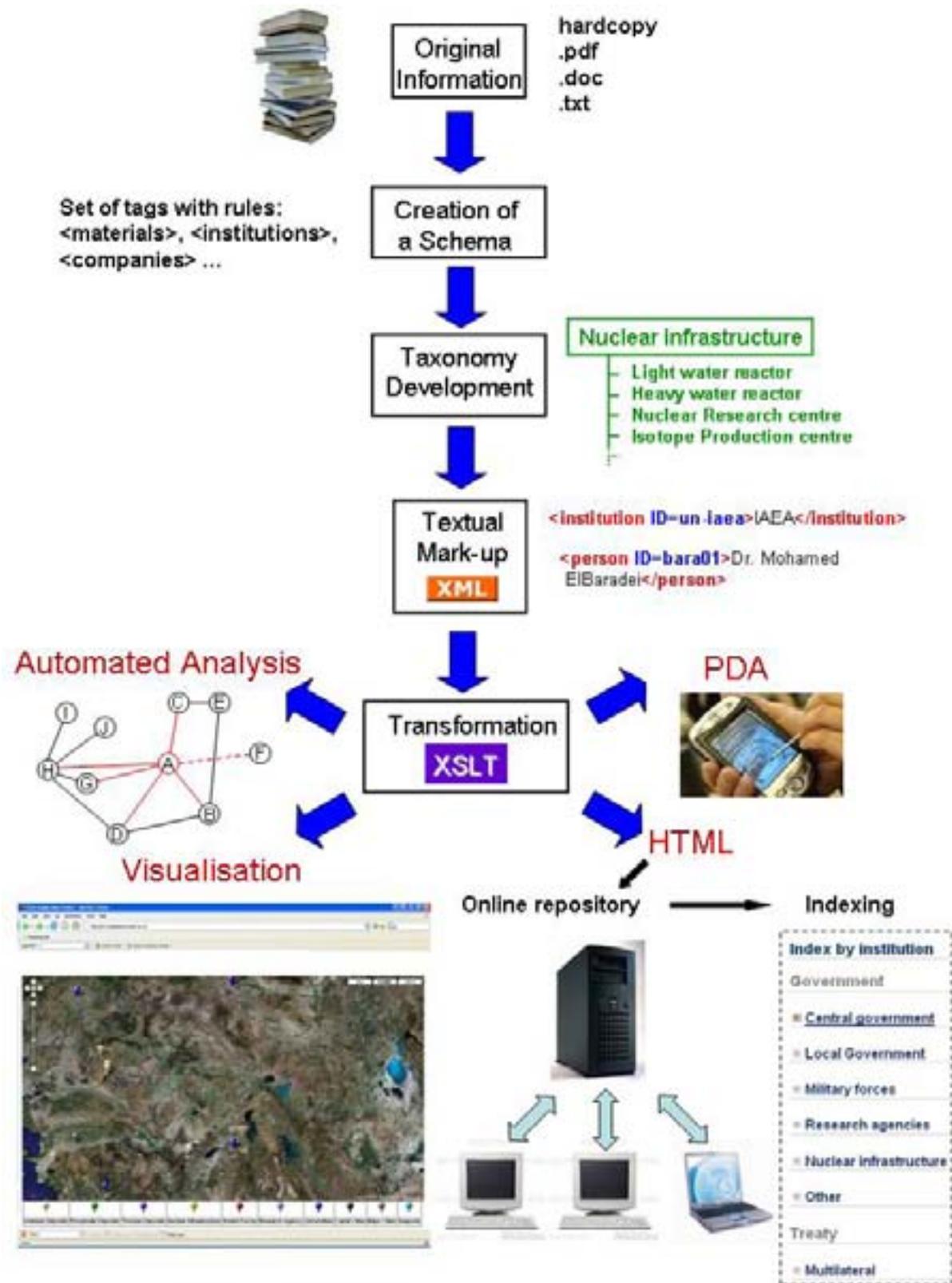


FIG. 1. Key stages in the development of an XML data repository, illustrated with some potential applications.

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A geo-portal for data management and integration in the context of the additional protocol

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Abstract. Since the entry into force of the Additional Protocol, nuclear safeguards is faced with an abundance of information from different sources which require effective tools for the collection, storage, analysis and retrieval of data. Over the last years, the Joint Research Centre (JRC) has developed a number of tools supporting the preparation, management and verification of AP declarations, which were demonstrating the usefulness of Geographic Information Systems (GIS) for this purpose. Following these experiences, DG-TREN has asked the JRC to develop a Geo-Portal integrating the diverse databases at DG-TREN and providing a single point of access to all safeguards-relevant information. Based on the same platform, NUMAS was developed with the objective of integrating information from various sources in view making available information on Nuclear Fuel Cycle. The article describes the objectives and the features of both systems. .

1. Introduction

The goal of integrated safeguards is to improve the effectiveness and efficiency by integrating classical safeguards with the strengthened measures under the Additional Protocol (AP) while maintaining overall cost neutrality. The implementation of strengthened safeguards measures creates a wealth of information including amongst others AP declarations, complementary access reports, inspectors reports and Open Source information [1][2]. In order to achieve maximum efficiency, the responsible organisations require an integrated information system that allows the collection, management and analysis of all relevant data. Such a system must provide central collection and storage of different types of data, support data analysis, the generation of reports and the planning of on-site inspections. Currently many organisations have a series of legacy databases and data sources each with a separate interface that are not cross-linked which each other. A safeguards information system must integrate the existing databases and provide a single point of access to the user.

Most of safeguards relevant information has a geographical component, i.e. it is linked to a geographical feature like a mine, site, facility or single building. Therefore, a Geographical Information System (GIS) is a suitable tool to satisfy the needs of a safeguards information system: a GeoDatabase contains all the geographical features of interest together with the cross-links to the relevant data items in the existing databases, thus integrating the diverse databases [3]. The GeoPortal provides a map-based interface in which the user can browse and query the GeoDatabase and which allows to link from each geographic feature to the relevant information in the existing databases, thus providing a single point of access for the user.

In the past, the Joint Research Centre of the European Commission (JRC) has developed a number of tools supporting the preparation, management and verification of AP declarations [4] showing the

usefulness of GIS tools in the context of the Additional Protocol. Following these experiences, DG-TREN and the JRC are developing SIT-ES, a Geo-Portal which supports DG-TREN in fulfilling their obligations under the Euratom treaty and the Additional Protocol. This article describes the objectives, features and architecture of the Geo-Portal as it is proposed by the JRC.

2. SIT-ES

SIT-ES (Site Investigation Tool for European Safeguards) is the latest evolution of the SIT tools which have been developed to support the management, analysis and verification of AP declarations. It has been designed to the needs of the DG-TREN and is currently under development.

2.1. Objectives

The Directorate General for Energy and Transport of the European Commission (DG-TREN) is responsible for the implementation of the EURATOM Safeguards system and the Additional Protocol within the EU countries. The objective of SIT-ES is to establish a single IT system that supports all parties at DG-TREN in performing the safeguards related tasks with the maximum efficiency and effectiveness. In particular it will support the Additional Protocol team and the Euratom inspectors in their work.

The Additional Protocol team is responsible for managing and verifying the consistency of declarations submitted by nuclear sites operators in the EU countries. After verification, DG-TREN forwards the AP declarations to IAEA and, if necessary, answers any requests for clarification from IAEA. It collects and archives any useful complementary documents that can be used to better describe the sites. SIT-ES allows the AP team to have fast and direct access to their information related to a particular facility or activity declared under the Additional Protocol, including present and historic AP declarations, Complementary Access reports, inspection reports, Open Source documents, etc. The possibility to intuitively browse a map-based interface and quickly find the required information is essential for verifying the declarations efficiently and accurately. The reporting capabilities of SIT-ES help to answer clarification requests from IAEA or other interested parties.

The Euratom inspectors are responsible for material accountancy in EU facilities and are involved in complementary access visits under the Additional Protocol. Fast and easy access to all relevant information is important support for their work. For example, the geographic features of SIT-ES, including the site drawings with the related information of each building can contribute to the planning of site inspections.

2.2. Features

In order to achieve the objectives described above, SIT-ES has been designed as a multi-user GIS system with a central database and a Geo-Portal as interface to the end-user. The SIT-ES has been designed with the following features:

- It is a multi-user information system: a (small) number of expert users are responsible for validating, inserting and maintaining the data. All authorized users then browse the same data set for read-only access. This ensures that all users always access the most up-to-date and accurate information. The system assigns different roles to users according to their function: expert users (i.e. members of the AP team) have full read/write access to the data; standard users have read-only access to all or parts of the data.
- It is a Geographic Information System (GIS): a GeoDatabase contains all geographic features which are of interest for European safeguards, for example mines, research locations and nuclear sites with their buildings that are declared under the Additional Protocol. A map-based interface allows intuitively browsing maps at different scales (from the state level down to the site level), viewing the content of the GeoDatabase in its geographical context and retrieve the associated information.

- It integrates the different legacy databases at DG-TREN (containing for example AP declarations, material accountancy, complementary access reports) and thus acts as a Geo-Portal to the existing information: for each entry, the GeoDatabase contains the cross-links to all related information items in the existing databases. For example, for each nuclear site it contains a link to all AP declarations related to the site, all complementary access reports, all Open Source information, etc. On the SIT-ES map interface, the user can then browse to the site and request any existing information with a simple mouse-click. The request is forwarded to the application that handles the specific database, which then takes care of the retrieval and visualization of the data (see Figure 1).
- The system enforces the security requirements that come with this type of sensitive data. It is installed within the DG-TREN network, where strict security measures ensure the integrity of the data. SIT-ES does not retrieve any of the sensitive business data in the existing databases directly, but only provides an integrated point of access to the user. Retrieval and presentation of the business data and the related security issues are handled by each database application separately. Furthermore, SIT-ES ensures through its role-based access model that each user is provided only with the functionality for which he is authorized.
- It can be expanded to directly support data exchange and dissemination, e.g. using the http protocol within Virtual Private Networks (VPN). Without compromising security constraints, all concerned parties (e.g. inspectors during site visits, country officers or the hierarchy at the EC) could then easily access the information of interest for which they are authorized.

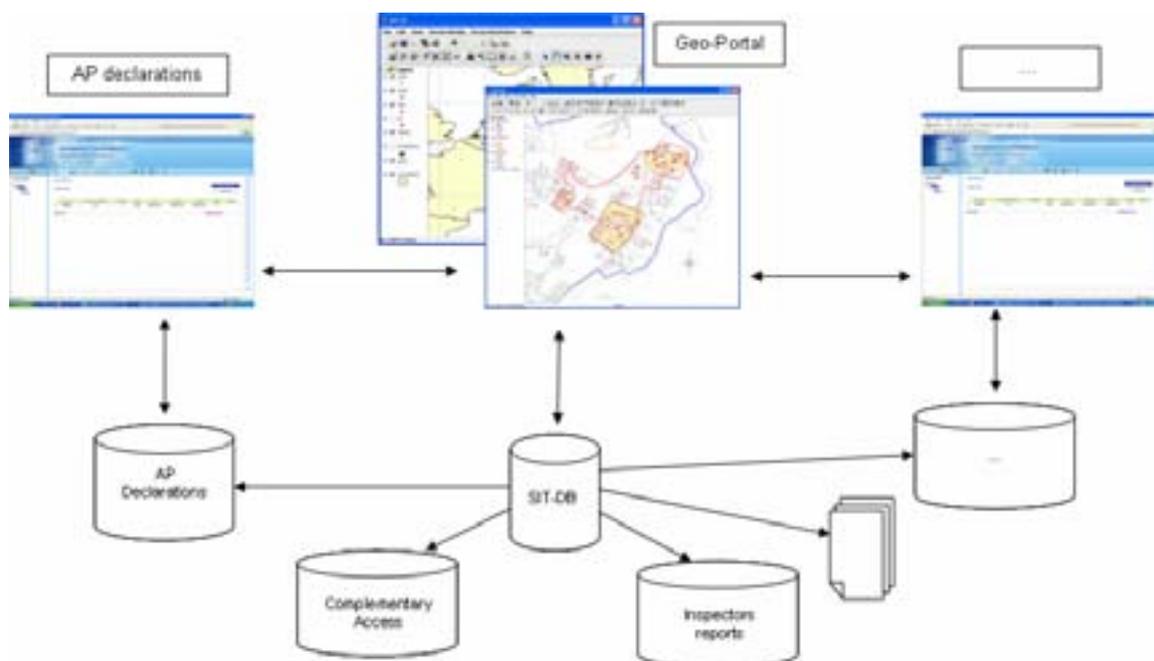


Figure 1. The map-based interface of the Geo-Portal is the entry-point for the user. He can browse and select the feature of interest and query it for further detailed information, which will be provided by the existing database applications.

3. NUMAS

The same technical platform can also function as a basis for other related tools that can be used in the Commission or by other clients. An example is the NUMAS database, which is developed aiming at integrating geographical and textual open source information in the context of non proliferation (basics in nuclear fuel cycle, country profiles and related investigations) [5].

The objective of NUMAS is to integrate information from various sources and to make it available to various levels of users. The growing importance of the non proliferation issues and the growing

E. Wolfart et al.

number of people involved in these activities will make this type of system a necessity. NUMAS is a good demonstrator that could be rapidly arranged until the stage of practical tool.

From one hand it is a structured repository of information relating to various Nuclear Fuel Cycle installations, and therefore to the countries of interest. From another point of view it is an interface towards external data bases. This integration tools corresponds to the growing need to bring together sparse pieces of information for a growing number and variety of users.

The main functions already developed are:

- to create (or to modify) a facility with the relating description table that summarises the characteristics (*attributes / descriptors*) of the facility (type, outputs, owner, geographical coordinates, operative status etc.), and its geographical location
- to create a site where a few facilities are regrouped,
- to find a facility / location by its name or by locating it on the map,
- to attach to any facility, or to a country relating documents such as reference or synthesis documents, satellite images, maps, reports, etc. and to access them.

The developed demonstrator allows therefore collecting and storing information about nuclear facilities and displaying it in graphical form. Descriptors are recorded after validation. Selected documents such as satellite images, reference or external documents etc. can be attached at facility or country levels. Therefore the factual information is structured in the following way: (country) -> (facilities) -> (descriptors; documents; images). As part of the experts tasks, synthesis documents can also be drafted and attached at the appropriate level (facility, country), allowing the presentation of the results of investigations on a given country.

Hundreds of facilities are already recorded in NUMAS. Documents of interest are regularly added (satellite images, basic technical documents, etc.)

Spatial queries are easier to use than manually typed queries; they can be made from the information available in the database. A simple logical operator enables the expert asking questions such as: searching for all the facilities of a given type (i.e. given part of the nuclear fuel cycle), with output comprised between two values, with a given status etc. A complex operator, based on NFC structure allows making searches and display results for facilities of a given type in a selected country.

An interface was developed allowing making simple or advanced queries in an external document repository managed by Verity, a commercial document management system. An interface for queries done with the help of topic trees was also developed, therefore demonstrating the feasibility of the interface.

Further developments being considered or under implementation:

- Access privileges: NUMAS is conceived as multi access tool. Experts can access it with the purpose of getting factual information about a given facility (location, owner, status...). Experts can also feed the system with the results of their investigations (maps, external or reference documents, satellite images...). Synthesis reports drafted by experts at the level of facilities or countries can be sensitive and therefore deserve to be protected. This will be the role of the display level. Remote access may also be necessary, mainly because protected information should be shared between EC services that are located at different places.
- One of the foreseen developments is to use NUMAS for training people involved in non proliferation activities. The growing importance of the subject and therefore the growing number of people having to handle this type of information requires that training tools are available.
- Organisation and traceability of comments at the report elaboration stage. Synthesis documents require reviews and validation before their release to a larger audience: technical review for

evaluating the data and its overall coherence (ex. Plutonium production vs. reactor fuel mass vs. mines production); geopolitical one (ex: energetic situation of the country) or more political. The comments made under these reviews should be taken into consideration and answered. A development enabling the management of comments to the synthesis documents (facilities, countries) is under consideration.

- Development of new interfaces and visualisation is also under consideration. They can be geographical (representation of quantities per country, of links / flows between countries or facilities, comprehensiveness of industrial process, etc.) or functional (level of compliance with a standard target such as completeness of an activity etc.)
- Extension and adaptation of the existing connection to external data base / repository of documents for use in real environment and feedback experience. IAC (Information and Analysis Centre) data organisation consists of document indexing, their categorisation (manual or automatic), and of features for advanced searches in the specific field of application (topic trees, proximity operators, thesaurus). Additional relevant functionalities concern multilanguage, duplicate detection and management of security and accesses.

Topic trees are complex queries with a tree-like structure. Topic trees are based on physical model of the Nuclear Fuel Cycle of the IAEA; they are a “translation” of the physical model into linguistic modelisation using words and logical rules in particular proximity rules. They allow performing context-related searches and categorisation of documents. The Enrichment topic tree has been developed; it reflects the Enrichment phase of the Nuclear Fuel Cycle. It should be completed by those relating to other phases of the NFC.

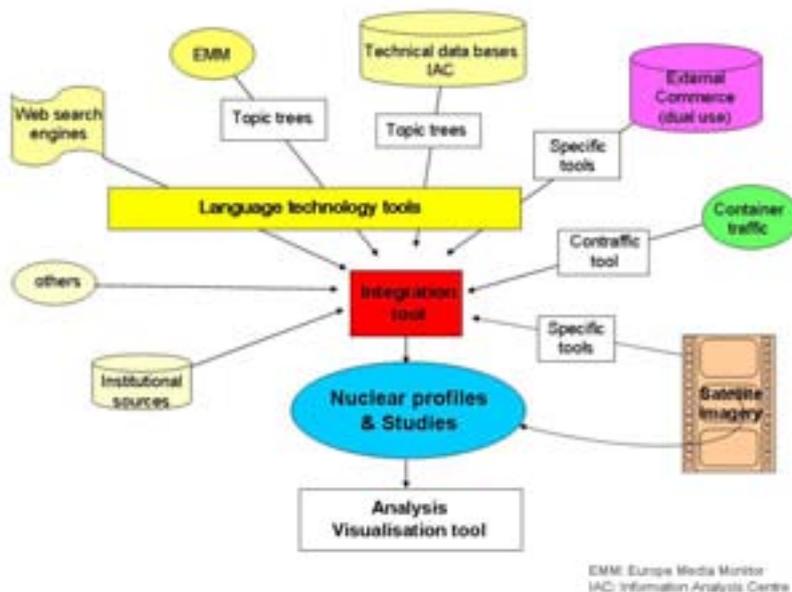


Figure 2. Functional scheme of the integrating tool NUMAS.

4. Conclusion

The development of the Site Investigation Tools (SIT) has shown that Geographic Information Systems (GIS) are suitable tools for the management and analysis of data related to the Additional Protocol. They put the safeguards related information in their geographical context and the GIS interface allows the user to quickly find and retrieve information by querying the associated object (e.g. a site or a building) on a map.

The JRC and DG-TREN are currently developing SIT-ES, a multi-user GIS system that integrates existing safeguards-related databases at DG-TREN and acts as a Geo-Portal to all safeguards-related data. The system will support the DG-TREN in fulfilling their obligations under the Euratom treaty and the Additional Protocol with maximum efficiency and accuracy.

E. Wolfart et al.

From its side, NUMAS intends to demonstrate the usefulness and rationality of a tool incorporating a geographical interface for integrating information coming from various sources.

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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.

G. Whitaker et al.

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 loss of key staff and critical resources could result in delays in the work and pessimism among stakeholders.

Accounting and verification of data from the operators

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Abstract. Ionizing radiation sources (IRS) data base and operation processes with them are based on regulations, which define necessity for IRS data base and requirements for its maintenances. These provisions are included as one of the tasks for Regulatory Authority assigned by the Saeima (the Parliament) and further elaborated into government regulations.

An additional aim for the data base of IRS is to provide data for emergency preparedness purposes – for the assessment of potential harmful effect due to accident with IRS and response to such accidents. There are also similar requirements for the control of process for importing of IRS into European Community.

Ionizing radiation sources and operation process data base actualization (data input, correction and canceling) occurred in compliance with determined algorithms, instruction and guidelines. The operator's accounting reports give additional on-line information, RAIS and SSAC data base requires to provide the Radiation Safety Center (RDC) with information, which shall be comprehensive, accurate and in conformity with legal requirements.

Given key studies illustrates the possibility to prevent incomplete information distribution and assist mathematical calculus to verify actual nuclear material quantity by operators related to safeguards issues.

General

Ionizing radiation sources (IRS) data base and operation processes with them are based on regulations, which define necessity for IRS data base and requirements for its maintenances. These provisions are included as one of the tasks for Regulatory Authority assigned by the Saeima (the Parliament) and further elaborated into government regulations.

An additional aim for the data base of IRS is to provide data for emergency preparedness purposes – for the assessment of potential harmful effect due to accident with IRS and response to such accidents. We have to know always about any high radioactivity source, its location; the application of it – we have to be capable at any time trace back the IRS from its production to the last known location.

There are also similar requirements for the control of process for importing of IRS into European Community. After the approval of import/export of sources from/to the European Community any further changes of location of ionizing radiation source must be reported, e.g. changes of both its location and operation.

Ionizing radiation sources and operation process data base actualization (data input, correction and canceling) occurred in compliance with determined algorithms, instruction and guidelines, which were summarized in IAEA printed handbook: „Working Material Regulatory Authority Information System RAIS 3.0 A management tool for the regulatory activities” and Radiation Safety Center Quality System's Guidelines.

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In addition the operator's accounting reports give the on-line information, RAIS and State System for Accounting and Control data base requires to provide the Radiation Safety Center (RDC) with information, which shall be comprehensive, accurate and in conformity with legal requirements.

Registration of data is based also on Council Regulation (EURATOM) No1493/93 on shipment of radioactive substances between Member States; in conformity with this Radiation Safety Center receives the information from appropriate operators regarding delivered IRS.

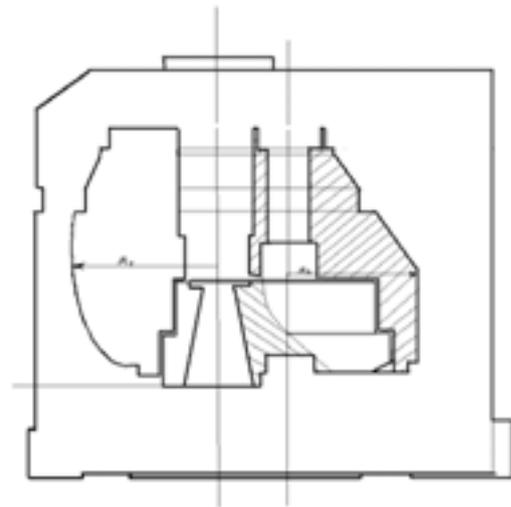
Annually the stored data is updated according to the law „On Radiation Safety and Nuclear Safety” and the recommendation for accounting of IRS is based on the “Code of Conduct on the Safety and Security of Radioactive Sources”, published in January 2004. The implementation of the Council Directive 2003/122/EURATOM of 22 December 2003 (Para's 3rd and 4th) on the control of high-activity sealed radioactive sources and orphan sources will be improved in nearest future.

Key Study

For the shielding of AGAT-1 depleted uranium holding is used (see drawing No.1). As ionizing source usually will be loaded Co⁶⁰ or Ir¹⁹², and respectively whole documentation and information is related to the gamma-source, which is loaded into equipment, but next to nothing - information regarding the depleted uranium.

Consequently a specific problem arose, flowed from the basic technical characteristic because there is incomplete information in source pass in connection with the shielding material.

Such gamma-equipment, which currently is in use in Latvia at two oncology hospitals were designed before 1990s and at that time no certificated shielding material's weights were declared. Initially, in 1993-1995, for the initial inventory and ionizing sources accounting data's summarization, shielding material weight was accepted and later declared as 500 kg. After that (in 2005) based on calculations, performed by the operators, apropos of de-exempted material (under complements Articles No 36(b) of INFCIRC/153¹) verification, *de facto* quantity was calculated less as previous declared, and depleted uranium components weight gain approx. 487 kg).



Drawing No.1

Gamma-sources beam's circuit card [from the AGAT-1 equipments technical characteristics eH1.197.000.TO]

Data for the calculations:

- Sphere range $R_1=175$ mm;
- Cylinder range $R_2=154$ mm;
- Depleted uranium density= 18.7 g/cm³;
- Collimators outlets volume= 950 cm³.

¹ Information required:

- quantities, uses and locations of “quantity exempted” materials
- estimated quantities, uses and locations of “use exempted” materials not yet in non-nuclear end-use form.

I. Lovjagina

Given example illustrates the possibility to prevent incomplete information distribution and assist mathematical calculus to verify actual nuclear material quantity by operators related to safeguards issues.

Safeguards matter for some transport containers

Report reviews reflect situation regarding nuclear materials' accounting its specified calculation and reporting on them to the IAEA.

Several years experience of accounting and control of nuclear materials in Latvia demonstrate some challenge to perform depleted Uranium's net weight estimation procedure in containers, designated for radiographic sources. The number of containers under consideration was 5 with the total initially declared weight – 104 kg. After rigorous study of containers, which were sent for storage, it was found, that shielding material used in containers is depleted uranium, covered with stainless steel. These containers for SSAC were previously defined only by weight (gross), it was mentioned only in notes, that depleted Uranium is covered with stainless steel.

The producer of containers for gamma-radiography at the time of their producing in passport data declared container's material as "heavy metal container". In IAEA accounting data these 104 kg, which actually means Gross weight = Depleted U weight + shielding stainless steel, cause impossibility to determine precisely the single constituent parts of containers – identification of weight for material under safeguards (depleted uranium). In printed sources search algorithm was unusable, because identification numbers on containers' surface were close to be completely erased. Only for first 3 containers numbers were partially extant: [Gammarid (prod.1977) weight 12 kg, Gammarid (no data) 6 kg and Stapelj (Serial No.1923 5m, prod. 1982)]. Remaining 2 containers were exclusively produced (non-standard type) and have no marking at all (only to be recognized by size and weight).

There was an attempt to use International Catalogue of Sealed Radioactive Sources and Devices as possible resource for supplementary information, by search parameter "*Depleted Uranium*" 497 devices were founded. Possibility to filter by device type – "*transport containers*" lends our chance to minimize potentially searched containers types to 121. If to filter by "*storage containers*", one can find only 19 types. 5 different types of Gammarid containers there were detected, wherefrom one type is in conformity with our partially assessed data. Part of holder information contains only general information (e.g. container's description by outside dimension (diameter, OD), internal dimension (diameter, ID), and containers estimated Gross weight value. Unfortunately, such information is not sufficient for directly determination of real weight for depleted uranium in these containers.

During re-calculations, performed by operator, simple mathematical functions were used, and public known relevance between elements' density and containers' weight, in view to covered material – stainless steel presence and containers inside volume, outside dimension, calculation are possible, but it is time-consuming (or we have to use more sophisticated calculation tools from industry).

Proposals

IAEA with co-sponsoring organization should expand International Catalogue of Sealed Radioactive Sources and Devices in view to change of present uphold approach, where containers exist only in tieback with radioactive sources. It seems preferable to introduce separate accounting also for transport and storage containers, not as relevant equipment or device for radioactive sources. In some cases chance to find searched containers' type depends on search parameter "containers type". Therefore, it is recommendable to expand "storage containers" search chapter.

Measurements and verification challenges of drums containing HEU/LEU nuclear waste under safeguards using IQ3 in South Africa

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Abstract. The Republic of South Africa (RSA) acceded to the NPT on 10 July 1991 and all drums containing nuclear waste of the past nuclear programme which are subjected to Safeguards could not be quantified as there were only estimate values. The initial declaration on these drums was based on historical values, process parameters, and estimated values which the IAEA could not verify since there was no suitable or available means of quantifying the nuclear material inside the waste drums.

The South African Operator of the drum storage facility could not establish an accurate inventory in this material balance area since 1991, as such the IAEA could not perform any physical inventory taking and physical inventory verification since 1993.

South Africa together with the IAEA approached the USA (DoE) and an IQ3 drum scanner was loaned on an agreement to South Africa in 2001 for the measurement of all drums under Safeguards both HEU/LEU nuclear wastes.

Drums, with different material content i.e. amount of material inside the drums, waste matrices, densities, uranium concentrations and enrichments were identified as the main challenge for measurements and preparations of calibration standards

All drums containing HEU nuclear waste has since been identified measured with IQ3 and declared and sealed by the IAEA. These drums together with their source documents were declared to the IAEA in 2005.

Currently our main challenge is that drums containing LEU nuclear waste are being scanned with IQ3 and declared to the IAEA on a monthly basis. The denial of verification of these drums by IAEA is due to the unavailability of supporting documents.

Keywords: IQ3, verification, non-proliferation, safeguards, physical inventory verification

1 INTRODUCTION

The South Africa government acceded to the Non-Proliferation Treaty of nuclear Weapons (NPT) on 10 July 1991 and this was followed by the signing of the comprehensive safeguards agreement with the IAEA on 16 September 1991 (INFIRC/394).

Following the signing of the agreement a team of Safeguards inspectors were assigned to verify the correctness of South Africa's declared inventory of nuclear materials. These included the examination of contemporary operating and accounting records and analysis of the nature and quantity of nuclear material.

During the initial declaration nuclear materials on these drums was based on historical values, process parameters, and estimated values which the IAEA could not verify since there was no suitable or available means of quantifying the nuclear material inside the nuclear waste drums.

Due to the lack of equipment and unavailability of new methods to measure drums with nuclear waste the USA (DoE) department was approached by IAEA to assist South Africa (Necsa) in measuring nuclear waste with different matrices which has been outstanding for the past 15 years. The USA (DoE) then loaned IQ-3 equipment via the IAEA for safeguards measurements.

2 METHOD

2.1 Identification of Drums

Each drum was opened in a specially designed controlled area in order to determine the content, fill height as well as taking digital photographs of each drum. The drums data are entered into an electronic database, issued with a bar-coded number and then registered on a Waste Tracking System (WTS). This allowed the facility operator to accurately keep track of 50 000 drums of which approximately half are under Safeguards.

2.2 Drum Sorting

All drums were assayed, using a HM-5 hand monitor in order to determine radiation levels and also weighed after identifying the matrix material. Drums with radiation levels greater than twice background are identified and earmarked for IQ-3 measurement (Safeguarded drums). Drums with radiation levels less than twice background are identified as non-Safeguarded and not IQ-3 measured. These non-Safeguarded drums were sampled and measured on a random basis using the IQ-3 to validate the screening criteria based on the HM-5. The results proved conclusively that these screened drums contained negligible amounts of nuclear material

Safeguarded drums are then classified as either compressible or non-compressible i.e. low density of high density matrices. Bottles filled with crushed filter candel were measured individually in the IQ-3 using a custom made assay jig.

2.3 Calibration Method

Efficiency calibration was done for each drum type and this was achieved through the use of calibration drums with densities from 0.02 to 1.3g/cc. Americium lithium source i.e. multi energy line was used. Energy calibration was used by using a line source

Calibration was done for less than 0.7g/cc and relative low concentrations. The drums were subdivided into two main groups. For each group type a calibration drum and line standards were set. These are being verified by using uranium standards in calibration drums.

2.3.1 Measurements

The drum measurement (scanning) was fully automated. The measurements were done according to the drum size and the fill height of the drum. Each drum measurement takes approximately 20 minutes.

Energy calibration measurements are done on daily basis to verify the efficiency of IQ-3 and the spectrum obtained is compared with the known calibration curve. After each measurement, the results are interpreted and corrected where possible.

3 MEASUREMENT AND VERIFICATION CHALLENGES

The establishment of calibration curves of various nuclear materials with different matrix structures had been established for the IQ-3. The main challenge is obtaining a homogenous representative sample for measurement from the nuclear waste material with different matrix structure. At times the matrix structure of the nuclear material is either solid or liquid or both

The other main challenge is the establishment of high density and low density calibration curves for Algel containing nuclear materials with different concentrations of uranium.

4 CONCLUSION

All known HEU nuclear waste in South Africa has been measured and odd ones are being identified in the process of identifying LEU nuclear waste. This nuclear waste has been placed under IAEA seals and thus significant progress has been made by South Africa in addressing anomalies

South Africa has established not only new capabilities established NDA measurements but had trained and established more NDA specialist. All known historic HEU drums had been declared to the IAEA and awaiting their inspection as well as their verification.

Advances in NDA of Pu in Pu-Be neutron sources

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Abstract. Non-destructive methods for assaying sealed Pu-Be sources have been developed, by using neutron and gamma-spectrometric measurements. In the first method, a combination of gamma-spectrometric determination of Pu isotopic abundances and neutron counting were used. A passive neutron coincidence collar was designed with 14 ^3He counters embedded in a polyethylene moderator surrounding the sources to be measured. The electronics consist of independent chains of pulse amplifiers and discriminators as well as a shift register. The neutron yield of the sources was determined by gross neutron counting, and the Pu content was assessed by adopting specific (α,n) reaction yields of individual Pu (and Am) isotopes, normalized to a calorimetric calibration. A pure gamma-spectrometric method was also developed, by determining photopeak areas of ^{239}Pu lines, and applying absorption correction. In the third, pure neutron method, real coincidence count rates (R) were also recorded in addition to total (T) count rates, and an empirical correlation between the ratio R/T and Pu content was used. The results of individual methods agree well. In a series of measurements, determination of the Pu content of 75 Pu-Be sources resulted in a total Pu amount of 563 g in contrast with a declared nominal value 2050 g.

1. Introduction

As containing nuclear material, Pu-Be neutron sources represent an issue both for safeguards and illicit trafficking. About 200 such sealed sources are in Hungary (and a lot of them in several neighbouring countries). Currently these sources are mostly out of use, and a number of them are temporarily stored at our institute. Their Pu content is regularly reported to and inspected by IAEA. However, this quantity is not declared; neutron output and source activity have only been given by the supplier. From these quantities nominal Pu contents were calculated, dividing activity by the specific activity value 2.3 GBq/g (0.0617 Ci/g) of ^{239}Pu [1], implying at the same time that pure ^{239}Pu content was assumed. These nominal Pu contents are still being declared and accounted for.

For most of the sources, the conversion factor between neutron output and activity is 2.70×10^{-5} neutron/s-Bq (1×10^6 n/s-Ci). This corresponds to the lower limit of the (α,n) specific yield given as $(1-2) \times 10^6$ n/s-Ci for ^{239}Pu -Be sources in the literature [2]. Since the specific activities of the various Pu isotopes differ substantially, the neutron yield depends very heavily on the actual isotopic composition. Our aim was therefore to determine it, and then to infer a more realistic Pu content of Pu-Be sources. The isotopic composition (isotopics) was determined by high resolution gamma-spectrometry (HRGS) which, on the other hand, is necessary anyway for identification of smuggled and seized, accidentally found, or not documented sources.

For determining neutron output (strength), a neutron coincidence collar was built, measuring total (gross) and coincidence count rates. The first model was operating with 9 ^3He tubes, 9 preamplifier-amplifier-discriminator chains, a commercial JSR-11 shift register (loan from IAEA), and a notebook computer using standard Agency software (INCC code). Detection efficiencies of the two moderator configurations were 2.8 and 8.8 % for Pu-Be sources [3, 4].

A new, upgraded detector system was developed in the frame of the Hungarian support programme to IAEA safeguards (task HUN A 1503 "Verification of Pu in PuBe neutron sources by neutron assay"), with 14 ^3He tubes, new, faster electronic units, and a JSR-14 shift register (loan from IAEA). The independent channels have the advantage of diminishing dead time of signal processing. Different

detection efficiencies (5, 9.6, and 11.3 %) can be selected using three moderator configurations, depending on the neutron output of the source to be assayed. These configurations were optimized with the aid of Monte Carlo simulation.

Since sources of certified Pu content were not available for us, calibration of our methods was carried out by using the results of measurements of a series of sources by a calorimeter provided by the Institute for the Protection and Security of the Citizen (EC JRC IPSC, Ispra, Italy) in the frame of a collaboration. An additional, independent NDA method was also developed using pure gamma spectrometry, without neutron measurement [5, 6]. Furthermore, a method of using pure neutron measurements only was developed as well, without γ -spectrometry at all. Based on total (T) and coincidence (R) count rate measurements, this so-called R/T method relies on a correlation established between the ratio R/T and the Pu content [7].

In this report we summarize the results of our development work. Source strengths range from 10^4 to 10^7 neutron/s, while the nominal Pu contents range from 0.1 to 178 g. The Pu–Be neutron sources are encapsulated in steel cylinders of 3–5.5 mm wall thickness. Their outer diameter and height vary from 10 to 35 and 19 to 45 mm, respectively. An additional encapsulation was necessary due to the expired guarantee time of the original encapsulation as well as to the rusty surface of some of the capsules. In the course of this re-encapsulation, the used Pu–Be sources in their present form – i.e. in their present capsules - were sealed into stainless steel (KO-36 type) capsules with a wall thickness of 1.5–2.0 mm. The capsules were sealed by using Argon gas-protected welding. The new capsules (diam. 22.2–42, length 50–147 mm) contain 1–4 old Pu–Be sources in order to decrease the required storage room for the sources in the final disposal.

2. Determination of the isotopic composition

It is important to know the isotopic composition of PuBe sources, because neutron output and Pu mass depend sensitively on it. Pu isotope mass fractions and the ratio of Am to total Pu mass were evaluated by HRGS using the advanced commercial Multi-Group Analysis computer code MGA++ [8, 9], analyzing gamma- and X-rays below 300 keV. The technique is based on the peak-ratio technique.

Large area planar Ge detector (diameter 50 by 20 mm thick, resolution 688 eV FWHM at 122 keV) was used. Spectra were taken for 10–50 min counting time at 50–60 cm source-to-detector distance, while the sources were taken out of their containers. The results show that isotopic composition varies in broad ranges. The ^{239}Pu abundance amounts to from 75 up to 96%.

3. Determination of the Pu content

3.1. Combined neutron-gamma method

Gross neutron output of the source, taken out of its container and inserted in the cavity of the collar, is measured for 2000–3000 s.. The Pu content is calculated from neutron output and isotopic composition, relying on specific (alpha,n) yields adopted for individual Pu isotopes and Am, as:

$$M_{Pu} = \frac{N}{M \sum_i f_i g_i} \quad (1)$$

where

N is the neutron output,

M is the multiplication in the source due to secondary (neutron-induced) reactions (see below),

f_i is the abundance of the i -th (Pu and Am) isotope,

g_i is the specific (alpha,n) yield of the i -th isotope.

Summation goes over all the isotopes, including that other than Pu as well. Since f_i values are expressed in terms of percentage of the total Pu content, $\sum f_i$ exceeds 100% by the abundance of ^{241}Am , and the formula gives the Pu content only (considered to be a “true” value in contrast with the nominal one).

Specific (α,n) reaction yield (g_i) values for the Pu (and Am) isotopes were determined by starting with the specific alpha activities from the literature, multiplied by n/alpha ratios, which convert activity to neutron output. These ratios are sensitive to alpha energies, and thus are different for individual isotopes. The products obtained in this way are maximum attainable values. They can be much less, if the Pu and Be constituents are incompletely dispersed and mixed in the source material upon production of the sources. Thus, the products of the two factors are to be normalized. This was carried out by using the results of calorimetric measurements. The heat output was measured for 19 PuBe sources by the ANTECH Small sample calorimeter model 601 provided by JRC IPSC, Ispra [10]. The instrument was previously calibrated in the PERLA laboratory of the IPSC using certified reference materials. By combining heat results with isotopics determined by gamma spectrometry, Pu masses were determined, relying on specific heat values from the literature. Using such a calibration, normalized specific (α,n) yields (g_i) were obtained, fitted to calorimetry, as follows (yields of $^{241,242}\text{Pu}$ are neglected):

$$^{238}\text{Pu}: 2.89 \times 10^7 \text{ n/g}\cdot\text{s}$$

$$^{239}\text{Pu}: 8.86 \times 10^4 \text{ n/g}\cdot\text{s}$$

$$^{240}\text{Pu}: 3.25 \times 10^5 \text{ n/g}\cdot\text{s}$$

$$^{241}\text{Am}: 5.76 \times 10^6 \text{ n/g}\cdot\text{s}.$$

These figures are valid for sources produced in the late Soviet Union till September 1978. According to an informal notification from the manufacturer, sources produced later are of specific neutron yield twice as high. We had no opportunity to check this statement as yet.

The ^{241}Pu and ^{242}Pu specific yields, being of the order of 10^3 , negligible at the usual isotopic ratios, were not considered. Similarly, the neutron yield from spontaneous fission of isotopes of even mass number was neglected as well.

1. In addition to gross neutron (totals, singles) count rates, coincidence (doubles) count rates were also recorded. Real coincidences (reals, after subtracting random coincidences) are due to secondary reactions, i.e. neutron-induced fission of the Pu isotopes (self-multiplication) and the $^9\text{Be}(n,2n)^8\text{Be}$ reaction. Contribution of spontaneous fission neutrons, as treated in Refs. [3, 4], can be estimated to be negligible.

The multiplication in the source itself was taken into account by the correction factor M . This correction may amount to 25–30 % for the strongest sources, and can be determined by using coincidence measurements or Monte Carlo calculations. The results were practically the same by the two methods. An equivalent way is the use of the analytical formula

$$M = 1 + 0.034 \log M_{Pu} + 0.0153 \log^2 M_{Pu} + 0.00283 \log^3 M_{Pu} \quad (2)$$

and, if necessary, of iteration (usually 2 steps are sufficient).

Uncertainty of the Pu mass determined by calorimetry was mainly due to the error in determining the isotopics, which was in general taken to be 3 - 4 %, while the systematic error of neutron output was given originally as 10 %. Nevertheless, the standard deviation among the measured sources of the same nominal (declared) neutron output was 2-3 % only, therefore the precision of the combined method may well approach 5–6 %, even though the absolute value may differ from the real Pu mass by 15–20 %.

2.2. Pure gamma spectrometry

Pu masses were also determined by pure HRGS, without neutron measurements. The method relies on absolute intensity measurements of ^{239}Pu photopeaks, applying attenuation correction [5, 6], and taking into account the ^{239}Pu abundance determined by γ -spectrometry.

The source is taken out of its container and the 375 and 413 keV photopeak areas of ^{239}Pu are measured by a large planar Ge detector for 10 – 20 min counting time in a far-field geometry (at 50 – 200 cm distance from the source, depending on its size and strength). Attenuation correction is applied, assuming a parallel beam falling on the detector surface and that the Pu-Be source has a cylindrical shape. The method is described in the companion paper [6] in detail, hence it is not repeated here.

The abundance of ^{239}Pu f_{239} is determined from the same gamma spectrum. The precision of the method is about 6%. An accuracy of 6-15 % can be attained, depending on source strength, measurement time, knowledge of the source diameter-to-length ratio, and wall thickness of the source.

2.3. Pure neutron counting (“R/T method”)

In addition to total neutron counting, coincidence counting can be exploited as well. Pure neutron measurements are carried out in this way, without γ -spectrometry at all. Based on total (T) and coincidence (R) count rates, this so-called R/T method [7] relies on a correlation established between the ratio R/T and the Pu content determined by calorimetry, using the graphs or the corresponding empirical formulae below (Fig. 1).

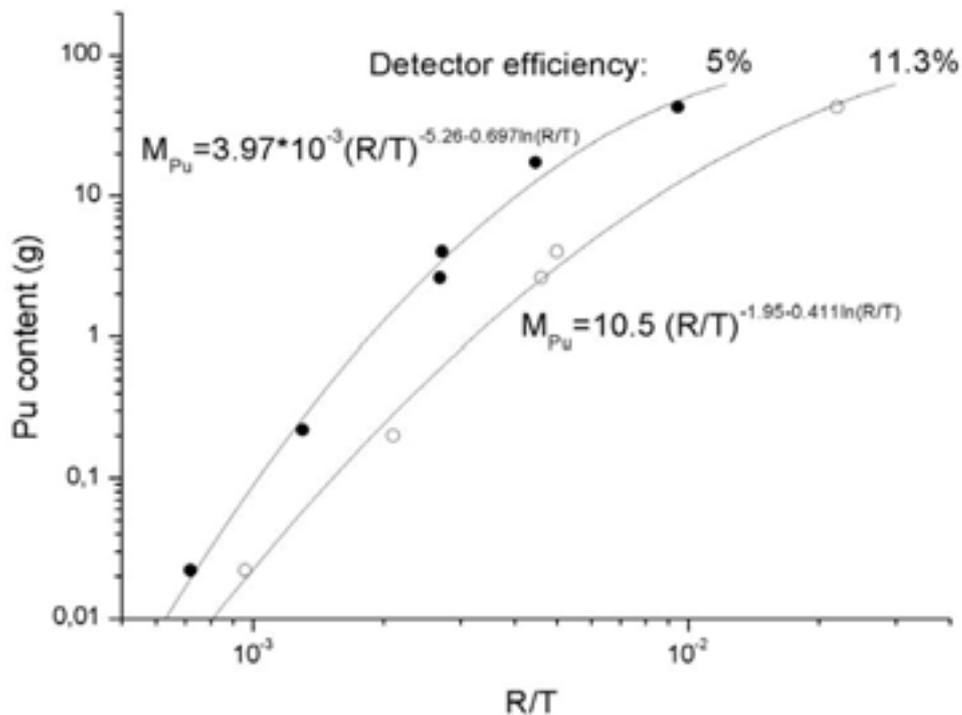


FIG. 1. Calibration of the R/T method against calorimetry.

The couple of parameters $[R_{norm}, T_{norm}=N]$ normalized to 100 % efficiency do not depend on the parameters of the neutron coincidence collar. An empirical formula, fitted to calorimetric results

$$M_{Pu} = 0.225(R/T)_{norm}^{2.65-0.23\ln(R/T)_{norm}} \quad (3)$$

gives Pu content by an accuracy of within 23 % on average. This method may need longer counting time than the two previous ones.

Comparison of results obtained by various methods is seen for 7 representative sources in Table 1.

Particularly, the couple $[R_{norm}, T_{norm}=N]$ can be used for estimating both the isotopic composition and the total Pu content (“R/T-T method”):

$$(R/T)_{norm} = 1.787 \left(\frac{N/M}{a + b * f_{239}} \right)^{0.44+0.024\ln\left(\frac{N/M}{a+b*f_{239}}\right)} \quad (4)$$

$$M_{Pu} = \frac{N/M}{a + b * f_{239}}, \quad (5)$$

where the values a and b are constant and found to be $(21.4 \pm 2.1) \times 10^5$ and $-(0.214 \pm 0.01) \times 10^5$ n/s.g, respectively.

Errors of the “R/T-T method” are about 2-3% for M_{239} and 15-20% for the Pu content..

Results obtained by using Eq. (4) are plotted in Fig.2 for various f_{239} -s, which were estimated from the function $[(R/T)_{norm}, T_{norm}=N]$ by graphical method.

Table 1. Main characteristics of 7 representative Pu-Be sources. Pu contents are determined by various methods (errors are in brackets).

Declared neutron output (n/s)	²³⁹ Pu fraction (%)	Pu content (g)				
		Nominal	Combined n-gamma	Pure gamma spectrometry	Calorimetry	Pure neutron counting
1.1×10 ⁴	83.15(0.83)	0.18	0.022(0.003)	0.021(0.003)	0.0223(0.0041)	0.023(0.003)
1.1×10 ⁵	79.76(0.80)	1.8	0.22(0.01)	0.225(0.004)	0.224(0.007)	0.23(0.03)
2.68×10 ⁵	95.21(0.19)	4.0	2.48(0.13)	2.55(0.10)	2.62(0.08)	2.6(0.3)
2.26×10 ⁶	76.92(0.82)	37	3.93(0.23)	4.0(0.3)	4.0(0.12)	4.4(0.7)
5.58×10 ⁶	76.17(1.0)	45	9.62(0.63)	9.5(0.5)	10.4(0.31)	7.33(2.5)
1.1×10 ⁷	75.7(0.91)	178	17.05(1.15)	17.8(1.6)	17.3(0.52)	15.3(2.9)
5.27×10 ⁶	94.91(0.19)	85	42.1(3.0)	44.0(1.5)	44.3(1.3)	46.9(4.7)

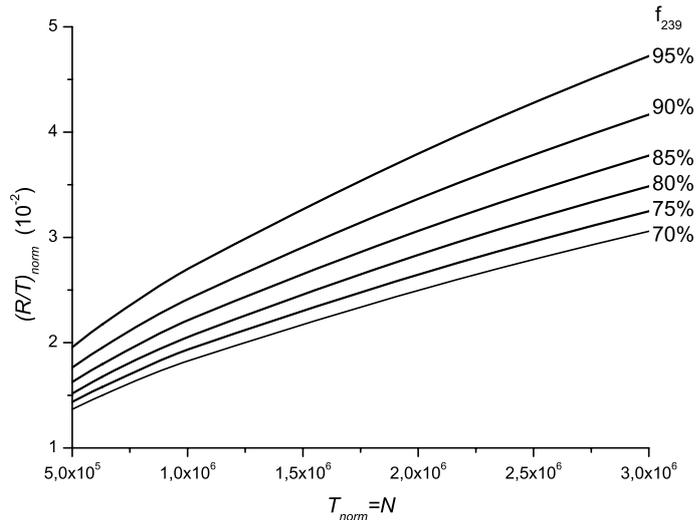


FIG. 2. $(R/T)_{norm}$ for various ^{239}Pu abundances as a function of $T_{norm} = N$.

4. Field test

As a particular effort for revealing illegally transported or vagabonding neutron sources in the frame of combating illicit trafficking of nuclear materials, the method and the equipment (together with the gamma-spectrometer) were employed in an in-field demonstration exercise, where the seizure of an illicit transport of a Pu–Be source was simulated and the Pu-content of the source was determined. The whole equipment was powered by a car battery, in this way forming a mobile measurement station. The scenario followed the provisions of the government decree, complemented with elements of the recommendations of the IAEA [11] and of the model action plan for seized nuclear material recommended by the international technical working group on nuclear smuggling [12]. Participants in the exercise were national public authorities (police, customs, atomic energy authority, etc.) and foreign observers from the IAEA and the Institute for Transuranium Elements (EC JRC ITU, Karlsruhe).

5. Summary

In a comprehensive measurement programme for 75 Pu–Be sources stored in our institute, neutron output, isotopics, and Pu content of the sources were determined. As a result, it has been turned out that facility and State inventories are based on incorrect, highly overestimated values. Nominal Pu masses have been overestimated even by an order of magnitude in some cases (where the ^{239}Pu abundance is about 75 % of the total Pu content). The measurements resulted in a total amount 563 ± 15 g of Pu, in contrast with 2050 g according to the sum of declared nominal values in the files. A new inventory of the Pu content of all the Pu–Be sources in the country is planned to be taken as well. The method is offered for routine IAEA use and also to other countries facing similar problems.

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Determination of the age of research-reactor fuel rods

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Abstract. It is demonstrated that a previously developed method for non-destructive, gamma-spectrometric age determination of uranium can be successfully used to determine the age of research-reactor fuel rods. It relies on measuring the daughter/parent activity ratio $^{214}\text{Bi}/^{234}\text{U}$ by low-background, high-resolution gamma-spectrometry using intrinsic efficiency calibration. The method does not require the use of any reference materials nor the use of an efficiency-calibrated geometry, and in many cases it could be applied by safeguards inspectors even on the site of the inspection.

1. Introduction

The age of uranium-bearing items is an important piece of information relevant both in nuclear safeguards and in combating illicit trafficking of nuclear material. It can help in determining the origin of seized or found nuclear materials, and, on the other hand, knowing the date of production of the material can help the safeguards expert to verify the correctness of the declaration of safeguarded material.

It has already been demonstrated that both mass spectroscopy [1], [2] and gamma spectroscopy [3], [4], [5] can be reliably used to determine the age of uranium, provided that a suitable sample is available. Items from which it is not possible to take out a sample, however, may frequently occur in safeguards verification or in combating illicit trafficking of nuclear materials. The purpose of the present research is to demonstrate that gamma spectroscopy can also be used to determine the age of such items, i.e., items that can only be investigated non-destructively, as a whole. In particular, in this study the age of VVR-SM and EK-10 research-reactor fuel rods has been determined.

The initial methodology for gamma-spectroscopic uranium age dating was derived during a "Round Robin" exercise, in which the properties of a highly-enriched uranium (HEU) powder were assessed by several laboratories, from the point of view of nuclear forensic science [6]. The uranium-age obtained by this gamma-spectrometric method was in agreement with the results reported by other participating laboratories, which used mass-spectrometry [5], [6].

The non-destructive method for age dating presented here relies on measuring the daughter/parent activity ratio $^{214}\text{Bi}/^{234}\text{U}$ by low-background, high-resolution gamma-spectrometry, assuming that the daughter nuclides have been completely removed during last separation or purification of the material.

^{214}Bi is a daughter of ^{234}U , which decays through ^{230}Th to ^{226}Ra , which in turn decays to ^{214}Bi through three short-lived nuclides. The time needed for secular equilibrium between ^{226}Ra and ^{214}Bi to be achieved is about 2 weeks, so it can be assumed that the activities of ^{226}Ra and ^{214}Bi are equal at the time of the measurement. Therefore, using the law of radioactive decay the activity ratio $^{214}\text{Bi}/^{234}\text{U}$ at time T after purification of the material may be calculated as

$$\frac{A_{Bi214}(T)}{A_{U234}(T)} = \frac{A_{Bi214}(T)}{A_{U234}(0)} = \frac{A_{Ra226}(T)}{A_{U234}(0)} =$$

$$= \lambda_2 \lambda_3 \left[\frac{\exp(-\lambda_1 T)}{(\lambda_2 - \lambda_1)(\lambda_3 - \lambda_1)} + \frac{\exp(-\lambda_2 T)}{(\lambda_1 - \lambda_2)(\lambda_3 - \lambda_2)} + \frac{\exp(-\lambda_3 T)}{(\lambda_1 - \lambda_3)(\lambda_2 - \lambda_3)} \right], \quad (1)$$

where $A_{U234}(0)$ denotes the activity of ^{234}U at time $T=0$ while λ_1 , λ_2 and λ_3 are the decay constants of ^{234}U , ^{230}Th and ^{226}Ra , respectively. Because $\lambda_1, \lambda_2, \lambda_3 \ll 1$, formula (1) can be developed into a Taylor series around $T=0$, so it can be approximated as

$$\frac{A_{Bi214}(T)}{A_{U234}(T)} = \frac{1}{2} \lambda_2 \lambda_3 T^2. \quad (2)$$

This equation can be used for calculating the age, T , of uranium samples after the activity ratio $^{214}\text{Bi}/^{234}\text{U}$ has been determined by gamma spectroscopy.

In order to eliminate the geometrical effects relative efficiency calibration is used to determine the ratio $^{214}\text{Bi}/^{234}\text{U}$. In this way the age of uranium samples of arbitrary physical form and shape can be determined [3]. In this work we use the peaks of ^{234}Pa and $^{234\text{m}}\text{Pa}$, which are a short-lived daughters of ^{238}U , to construct a relative efficiency curve, and to determine the activity ratio $^{214}\text{Bi}/^{238}\text{U}$. Furthermore, the activity of ^{234}U is determined relative to ^{235}U . If the activity ratio $^{235}\text{U}/^{238}\text{U}$ is also known one can calculate the activity ratio $^{214}\text{Bi}/^{234}\text{U}$ and thus the age of the sample can be obtained.

2. The fuel rods

In this study age dating of VVR-SM, VVR-SM/3 research-reactor fuel assemblies and broken pieces of EK-10 research-reactor fuel rods has been investigated. The fuel rods were available at the research reactor of the Atomic Energy Research Institute of the Hungarian Academy of Sciences in Budapest.

The nominal ^{235}U enrichment of VVR-SM and VVR-SM/3 rods is 36%. The total uranium mass in VVR-SM fuel rods is 107 g and the ^{235}U mass is 38.5 g. In a VVR-SM/3 fuel assembly three VVR-SM assemblies are tied together, therefore the total uranium and ^{235}U masses are three times bigger than for VVR-SM. The total length of the VVR-SM assemblies is ~85 cm, the active length is 60 cm.

The total uranium mass in an intact EK-10 fuel rod is 80 g and it contains 8 g of ^{235}U in the form of UO_2 dispersed in magnesium or magnesium oxide with 1 mm thick aluminium cladding; active length is 500 mm.

For the measurements fuel rods with different dates of production have been selected. In this way the age of the measured VVR-SM and VVR-SM/3 fuel rods varied in the range between 6 to 39 years, as it could be determined from the date of production kept in the records, which was also engraved on the cladding of the assemblies. The age of the EK-10 fuel was not exactly known, but from personal communication with the people at the research reactor, it seemed very likely that the fuel had been transported to the reactor site either before 1959 or 1967.

3. Determining the activity ratio $^{234}\text{U}/^{238}\text{U}$

The activity ratio $^{234}\text{U}/^{238}\text{U}$ was measured by a medium-area semi-planar high-purity germanium detector, ORTEC SGD-GEM 3615. The detector crystal diameter is 35.5 mm, the crystal length is 15.9 mm, while the distance from the crystal to the detector cap is 35 mm. The spectra of the fuel rods were taken in the energy range 0-300 keV, in an open room. The distance between the fuel rod and the detector cap was in each case adjusted in such a way that the dead time of the detector was never larger than 4-5%. Each spectrum acquisition lasted at least 1 hour and the isotopic composition of uranium was evaluated using the MGAU code [7], version 4. From the results provided by MGAU the $^{234}\text{U}/^{238}\text{U}$ abundance ratio was calculated. The $^{234}\text{U}/^{238}\text{U}$ activity ratio was obtained by using the values $T_{1/2}(^{234}\text{U})=2.455\times 10^5$ and $T_{1/2}(^{238}\text{U})=4.468\times 10^9$ for the half lives of ^{234}U and ^{238}U , respectively, [8], and it is given in Table 1.

Table 1. The measured ^{235}U enrichment and activity ratio $^{234}\text{U}/^{238}\text{U}$ for the selected fuel rods.

Type	Identification number.	Measured ^{235}U enrichment (%)	$^{234}\text{U}/^{238}\text{U}$ (Bq/Bq)
VVR-SM	28	36.28 ± 0.42	80.9 ± 5.2
VVR-SM	527	37.17 ± 0.33	89.1 ± 3.7
VVR-SM	211	37.22 ± 0.41	82.7 ± 4.6
VVR-SM/3	51	36.68 ± 0.55	84.6 ± 6.6
EK-10	-	10.07 ± 0.08	17.8 ± 1.3

4. Determining the activity ratio $^{214}\text{Bi}/^{238}\text{U}$

The activity ratio $^{214}\text{Bi}/^{238}\text{U}$ was determined by low-background gamma spectroscopy. A low-background iron chamber lined with copper, available at the site of the research reactor, was used. The wall thickness of the iron chamber is 10 cm. On the bottom of the chamber there is hole through which a coaxial HPGe detector is inserted to the low background area, while the detector dewar remains outside, i.e., below the chamber. The detector used was a CANBERRA GC3518-7500SL with active volume $\sim 140\text{ cm}^3$, relative efficiency 35% (in the standard definition, with respect to a 3x3 inch NaI detector, measured at 20 cm from the detector) and peak-to-Compton ratio 61:1.

Table 2. Energies and emission probabilities of the relevant gamma-peaks in the low-background spectra.

Energy (keV)	Emission probability (%)	Emitter
569.30	0.0203	^{234}Pa
609.31	46.1	^{214}Bi
766.37	0.3220	$^{234\text{m}}\text{Pa}$
1000.99	0.8390	$^{234\text{m}}\text{Pa}$
1193.69	0.0135	$^{234\text{m}}\text{Pa}$
1510.20	0.0129	$^{234\text{m}}\text{Pa}$
1737.73	0.0212	$^{234\text{m}}\text{Pa}$
1831.36	0.0172	$^{234\text{m}}\text{Pa}$

Spectrum acquisition lasted for about 6 hours for each VVR-SM rod, while the broken pieces of EK-10 were counted for one day. Several background spectra were also taken overnight and also over the

weekend. The background spectra were free from the relevant uranium-related peaks, while the background count rate of the 609 keV line of ^{214}Bi was found to be 0.0078 ± 0.0028 cps.

For each fuel rod an intrinsic efficiency calibration curve was constructed, using the peaks of ^{234}Pa and $^{234\text{m}}\text{Pa}$ which are short lived daughters of ^{238}U . Using this intrinsic efficiency curve and the count rate of the 609 keV line of ^{214}Bi , the activity ratio $^{214}\text{Bi}/^{238}\text{U}$ was calculated for each assayed fuel rod. The energies and emission probabilities of the relevant gamma peaks were taken from reference [9] for ^{234}Pa and $^{234\text{m}}\text{Pa}$, while for ^{214}Bi they were taken from reference [8] and are shown in Table 2. A large, 3 mm thick lead absorber plate, shielding the detector from the entire fuel rod, was placed next to the detector cap in order to reduce the dead time of the detector caused by the low-energy peaks of uranium. In this way the dead time was about 5 %. The reduction of the count rates of the relevant gamma energies due to the presence of the absorber was small. The results of the low background measurements are given in Table 3.

Note that the 609 keV peak of ^{214}Bi could not be evaluated for the triple assembly, VVR-SM/3, because within the available measurement time, no 609 keV peak could be observed above the very intensive Compton continuum of the high-energy peaks of uranium daughters.

Table 3. The background-corrected count rate of ^{214}Bi and the activity ratio $^{214}\text{Bi}/^{238}\text{U}$ for the selected fuel rods.

Type	Identification number	^{214}Bi count rate (cps)	$^{214}\text{Bi}/^{238}\text{U}$ (Bq/Bq)
VVR-SM	28	0.1396 ± 0.0103	$(3.26 \pm 0.26) \times 10^{-4}$
VVR-SM	527	0.1277 ± 0.0097	$(2.91 \pm 0.24) \times 10^{-4}$
VVR-SM	211	0.0519 ± 0.0094	$(1.40 \pm 0.26) \times 10^{-4}$
VVR-SM/3	51	< 0.0335	$< 2.71 \times 10^{-5}$
EK-10	-	0.1133 ± 0.0078	$(7.97 \pm 0.56) \times 10^{-5}$

5. Results

From the activity ratios $^{234}\text{U}/^{238}\text{U}$ and $^{214}\text{Bi}/^{238}\text{U}$ the activity ratio $^{214}\text{Bi}/^{234}\text{U}$, and thus the age of uranium can be obtained using equation (2). The measured ages are consistent with the ages which can be calculated based on the date of production of the fuel rods (Table 4). In fact, the uranium age is in each case larger than the assumed age of the rods. This is, at least partially, due to the fact that the uranium material from which the rods have been produced had been chemically separated and enriched before the fuel production took place.

A limitation of this method is that for very young items (with ages of few years) the measurement time should be very long in order to obtain meaningful results. Nevertheless, in this work it has been demonstrated that in most cases ~6 hours of spectrum acquisition is sufficient to determine the age of a fuel rod available at a research reactor.

Table 4. The age the fuel rods and the age of uranium in the rods.

Type	Identification number	Enrichment (%)	Year of production of the fuel rod	Age of fuel rod (years)	Measured uranium age (years)
VVR-SM	28	36	1967	39	45 ± 4
VVR-SM	527	36	1968	38	40 ± 3
VVR-SM	211	36	1985	21	29 ± 4
VVR-SM/3	51	36	2000	6	< 13
EK-10	-	10	before 1967 or before 1959	39 or 47	47 ± 4

It can be concluded that gamma spectroscopy can be used for age dating of realistic uranium samples of arbitrary shape which cannot be dismantled, such as fuel rods. Gamma-spectroscopy is a widely available, simple and cost efficient tool and some kind of low-background facility can be found in many laboratories world-wide. Therefore the described gamma-spectrometric method for age-dating of research-reactor fuel rods can be easily utilized (e.g., by IAEA inspectors on the move) almost anywhere where the need for such an investigation arises.

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Statistics of the neutrons and gamma photons emitted from a fissile sample with absorption

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Abstract. The calculation of multiplicities and number distributions for neutrons and gamma photons has recently been investigated. The calculations were based on master equations and probability generating functions. In this work we extend the models to include absorption of neutrons and gamma photons within the sample. By introducing modified factorial moments, the resulting expressions for the number distribution can be brought to a form equivalent to that of factorial moments. The results of the calculations show that absorption in a fissile sample will not affect significantly the statistics of the neutrons leaving the sample, compared to the non-absorbing case. But for photons the effect will be twofold, since the multiplication relies on the fission events induced by neutrons, and then the generated gammas themselves can also be absorbed. For fissile samples the heavy isotopes will have a large absorption of photons that increase with sample mass. This self-shielding effect will lower the number of emitted photons from the sample so drastically that for more massive samples it becomes favourable to examine samples using neutrons instead of photons in coincidence and multiplicity measurements.

1. Introduction

In some recent publications we have reported the calculation of multiplicities and number distributions of neutrons and gamma photons in a multiplying sample [1]. By acquiring knowledge about the number distributions one can get a good insight into the behaviour and the characteristics of a certain sample. In experiments the statistics are used to deduce the sample mass and isotopic composition by using multiplicity and coincidence measurements.

The inclusion of absorption in the model will affect the statistics of a multiplying sample in different ways depending on what type of emissions we are interested in. For neutrons the process of absorption will eliminate neutrons from the fission chain, and thus lower the probabilities of larger neutron bursts. For gamma photons on the other hand, the generation is not dependent on the photons, but only on the neutrons, therefore the effect for gammas will be dependent on that of neutrons as well as the fact that gammas will be absorbed. The total effect is that absorption will significantly lower the number of gamma photons that escape the target and becomes available for detection.

The sample mass will be the main parameter that affects the number distribution. For non-absorbing samples an increase in sample mass will only lead to an increase of generated neutrons and gamma photons. When including absorption, an increase in mass leads to an increased probability to induce fission. Contrary for the photons an increase in mass only means that a photon has to pass through

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more material before escaping the sample and becoming available for detection. This effect of self-shielding will naturally be coupled to the increase in multiplication of neutrons for increasing sample mass, and quantifying these two effects has been a main objective for this work. An important motivation for finding the relationship is the fact that knowing this effect allows us to make better predictions on what type of emission is most favourable to be observing for different sample masses and compositions. If the self-shielding effect is pronounced, observing neutrons will be more beneficial, while if the gamma photons have a low probability of absorption then detecting them is normally favourable due to their higher multiplication compared to that of neutrons. Furthermore for samples of intermediate mass the best strategy might be using for example organic scintillating detectors that are capable of detecting both neutrons and photons.

As in earlier studies the calculations are based on master equations of the probability distribution. These were treated and solved by the introduction of probability generating functions (PGFs). Since the higher order terms become increasingly complex we have used the symbolic computation language Mathematica [3], to find the probability distribution terms in a recursive manner. Earlier works have been performed to quantify the relationship between mass and multiplicities [1]. Absorption and the inclusion of the process of detection has also been investigated to some extent [5]. We believe that we have found a slightly different way of including absorption that is more coherent with the physical processes involved for the neutrons and gamma photons.

2. Theoretical treatment

Master equations for the generating functions of the number of neutrons and gamma photons have been derived earlier [1]. It was assumed that the probability of inducing fission was p and this property was known and depending on sample mass. In the present work we also include absorption. The way this was done is different depending on if we consider neutrons or gamma photons respectively. The goal of our calculations will be to find the number of neutrons (or photons) that escape the target, meaning they were not absorbed or consumed in a fission process, this is the number distribution that is relevant when later considering the process of detection.

2.1. Neutrons

The coupled backward-type master equations for neutrons reads as follows [4]:

$$h(z) = (1 - p)z + p q_f [h(z)] \quad (1)$$

and

$$H(z) = q_s [h(z)]. \quad (2)$$

These two equations refer to the probability generating functions of the number distributions $p_1(n)$ and $P(n)$ respectively:

$$h(z) = \sum_n p_1(n) z^n \quad , \quad H(z) = \sum_n P(n) z^n. \quad (3)$$

Here, $p_1(n)$ describes the probability to have n neutrons generated from a single neutron, and $P(n)$ is the same for starting with one initial event which in our case will be a spontaneous fission. The generating functions for the number of neutrons created in a spontaneous or induced fission event are:

$$q_s(z) = \sum_n p_s(n) z^n \quad , \quad q_f(z) = \sum_n p_f(n) z^n. \quad (4)$$

This model was earlier used to find the number distribution for samples where absorption was not considered. Since including absorption is the goal of these calculations we need a way of expressing

the event of absorption for a neutron. Earlier the only event was inducing fission with a probability p or failing to do so with probability $(1-p)$. For neutrons one can include absorption into this process by noting that, what regards the neutron balance, absorption appears as an induced fission where no neutrons were emitted. So by increasing the probability $p_f(0)$, absorption will effectively be included in the model. An important consideration that needs to be done is the fact that we need to increase the value of p since that will not only be the probability for a neutron to induce fission but also the mutually exclusive event of being absorbed.

The meaning of the probability $(1-p)$ then changes from having been the probability to not induce fission earlier, to now being the probability of leaking out from the sample. Since we will be making comparisons with Monte Carlo simulations done with the MCNP-PoliMi code [7], we will take the probabilities for inducing fission and absorption from that code which contains a number of nuclear data tables. Finding the number distribution will proceed as usual, by using the fact that it is the Taylor expansion coefficients of $h(z)$ and $H(z)$ respectively:

$$p_1(n) = \frac{1}{n!} \left. \frac{d^n h(z)}{dz^n} \right|_{z=0} \quad \text{and} \quad P(n) = \frac{1}{n!} \left. \frac{d^n H(z)}{dz^n} \right|_{z=0}. \quad (5)$$

The terms are once again dependent on finding the first term, i.e. the probability of having zero neutrons escaping the sample, which can be found from an N th-order polynomial:

$$p_1(0) = (1-p)z + pq_f [h(z)] \Big|_{z=0} = pq_f [p_1(0)] = p \sum_{n=0}^N p_f(n) [p(0)]^n. \quad (6)$$

Similarly to our previous work, plutonium samples will be considered, which fact effectively limits N to 8. As compared to the case of no absorption, we see that the change will be introduced in the $p_f(0)$ factors that have now changed values according to the method described above, as well as the increase of the first collision probability p .

Modified factorial moments will be occurring when calculating the number distribution. These factorial moments depend on nuclear data, but will change in the case of including absorption since we collected the absorption into the induced fission events.

$$\left. \frac{d^n q_f(h)}{dh^n} \right|_{z=0} = q_f^{(n)}(h) \Big|_{z=0} = \overline{v_f(v_f - 1) \dots (v_f - n + 1)}_a. \quad (7)$$

Here the subscript a indicates that we have included absorption into the data. The modified moments depend on $p_1(0)$ as a weighting factor, and since this quantity will have a slightly different value when including absorption, this effect will also change the numerical value of the modified moments for both induced and spontaneous fission. When it comes to the algebraic expressions of the probability terms, they will be formally the same as in the case of no absorption. The first term e.g. looks as follows:

$$P(1) = \frac{dq_s(h)}{dh} \frac{dh(z)}{dz} \Big|_{z=0} = \overline{v_s} \frac{1-p}{1-pv_f}. \quad (8)$$

Likewise, there will be a continued formal equivalence between the number distribution terms and the factorial moments expressed by the relation:

$$P(n) \Big|_{\overline{v_f}, \overline{v_s}} = \frac{1}{n!} \left\langle \tilde{v}(\tilde{v} - 1) \dots (\tilde{v} - n + 1) \right\rangle_{\langle v_f \rangle, \langle v_s \rangle} \quad n = 1, 2, \dots \quad (9)$$

This way of introducing absorption is computationally favourable since the master equations does not change and will therefore not require extra computing time. Furthermore with the factorial moments and probability terms looking as in the case with no absorbtion, one can directly use those expressions and just insert the slightly changed values of the modified moments to get the statistics and multiplicities with absorption included in the model, and thus finding the proper probabilities of leakage neutrons. The leakage neutrons are then the ones that can be detected by external detectors, which could be incorporated by introducing a detector efficiency in the factorial moments.

2.2. Gamma photons

The physics behind the generation of gamma photons is more complex compared to that of neutrons. The reason behind this is that photons do not have any self-multiplication, instead they rely on neutrons for multiplication (induced fission events). This makes it impossible to include absorption for gammas in the same way as we did for neutrons since they do not induce fission. In other words, it does not make sense to just increase the probability of having zero gammas produced in a fission event, because it would not give correct results. The physical process of absorption for gamma photons comes after they have been created from spontaneous or induced fission.

A photon will after creation undergo one of two possible events: either leaking out from the sample with a leakage probability p_L ; or being absorbed with probability $(1 - p_L)$. This process can be described by a master equation giving the probability of having a certain number of photons leaking out if one starts with one initial photon:

$$l_1(n) = (1 - p_L)\delta_{n,1} + p_L\delta_{n,0}. \quad (10)$$

Introduce now the probability generating function

$$l(z) = p_L z + (1 - p_L). \quad (11)$$

The master equations we use will now describe the number of leaked or escaped photons if the function $l(z)$ is incorporated correctly:

$$g(z) = (1 - p) + p r_f [l(z)] q_f [g(z)] \quad (12)$$

and

$$G(z) = r_s [l(z)] q_s [g(z)]. \quad (13)$$

The functions $g(z)$ and $G(z)$ now correctly describe the number of escaped gamma photons from one initial neutron and one initial neutron event respectively. Getting the number distribution is achieved in an analogue manner compared to that of neutrons. One observes that the number distributions are Taylor expansion coefficients of $g(z)$ and $G(z)$ respectively:

$$f_1(n) = \frac{1}{n!} \left. \frac{d^n g(z)}{dz^n} \right|_{z=0} \quad \text{and} \quad F(n) = \frac{1}{n!} \left. \frac{d^n G(z)}{dz^n} \right|_{z=0} \quad (14)$$

The master equations also contain two PGFs for the number of photons produced in a spontaneous and an induced fission event, respectively:

$$r_s(z) = \sum_n f_s(n) z^n, \quad r_f(z) = \sum_n f_f(n) z^n. \quad (15)$$

To start the recursive formula, we search for the quantity $f_1(0)$ from eq. (12). The resulting polynomial reads as follows:

$$f_1(0) = (1 + p) + p \sum_{n=0}^{24} f_f(n) [1 - p_L]^n \cdot \sum_{n=0}^8 p_f(n) [f_1(0)]^n. \quad (16)$$

Calculating higher order terms will require derivatives of Eqs. (12) and (13). Just like in the case of no absorption, one encounters modified moments when the derivatives of $q_\alpha(z)$ and $r_\alpha(z)$, $\alpha = \{s, f\}$ are evaluated at $z=0$:

$$\left. \frac{d^n r_\alpha(z)}{dz^n} \right|_{z=0} = \sum_m m(m-1)\dots(m-n+1) [1 - p_L]^{m-n} \equiv \bar{\mu}_\alpha(n); \quad \alpha = s, f; \quad (17)$$

and

$$\begin{aligned} \left. \frac{d^n q_\alpha(z)}{dz^n} \right|_{z=0} &= \sum_m m(m-1)\dots(m-n+1) \cdot p_\alpha(n) [f_1(0)]^{m-n} \equiv \\ &\equiv \bar{\nu}_\alpha(\nu_\alpha - 1)\dots(\nu_\alpha - n + 1) \equiv \bar{\nu}_\alpha(n); \quad \alpha = s, f; \end{aligned} \quad (18)$$

For the first source induced probability we arrive at

$$F(0) = r_s[l(z)] q_s[g(z)] \Big|_{z=0} = r_s[l(0)] \cdot q_s[f_1(0)] = \bar{\mu}_s(0) \cdot \bar{\nu}_s(0). \quad (19)$$

Calculating the probability for having one photon escaping when starting with one initial neutron or one initial source neutron event will lead to the appearance of the leakage probability p_L :

$$f_1(1) = \frac{p r_f[l(z)] l'(z) q_f[g(z)]}{1 - p r_f(z) q_f[g(z)]} \Big|_{z=0} = \frac{p \bar{\mu}_f(1) p_L \bar{\nu}_f(0)}{1 - p \bar{\mu}_f(0) \bar{\nu}_f(1)} = p_L \cdot \underbrace{f_1(1)}_{\text{no absorption}}. \quad (20)$$

Further,

$$\begin{aligned} F(1) &= r_s[l(z)] l'(z) q_s[g(z)] + r_s[l(z)] q_s[g(z)] \Big|_{z=0} = \\ &= \left(\bar{\mu}_s(1) \bar{\nu}_s(0) + \bar{\mu}_s(0) \bar{\nu}_s(1) \frac{p \bar{\mu}_f(1) \bar{\nu}_f(0)}{1 - p \bar{\mu}_f(0) \bar{\nu}_f(1)} \right). \end{aligned} \quad (21)$$

For factorial moments the leakage probability p_L will also be visible. Due to the formal equivalence between the number distribution terms and the factorial moments which we also had in the case of neutrons, it is easy to derive higher order terms of the multiplicities. The first two terms are:

$$\langle \bar{\mu} \rangle_a = \bar{\mu}_s \cdot p_L + \frac{p \bar{\nu}_s p_L \bar{\mu}_f}{1 - p \bar{\nu}_f} = p_L \underbrace{\left(\bar{\mu}_s + \frac{p \bar{\nu}_s \bar{\mu}_f}{1 - p \bar{\nu}_f} \right)}_{\langle \bar{\mu} \rangle}, \quad (22)$$

$$\begin{aligned}
 \langle \mu(\mu-1) \rangle_a &= \overline{\mu_s(\mu_s-1)} \cdot p_L^2 + 2 \cdot \overline{\mu_s} \cdot p_L \cdot \overline{v_s} \cdot \frac{pp_L \overline{\mu_f}}{1-pv_f} + \overline{v_s(v_s-1)} \cdot \left(\frac{pp_L \overline{\mu_f}}{1-pv_f} \right)^2 \\
 &+ \overline{v_s} \cdot \frac{p}{1-pv_f} \left[\overline{\mu_f(\mu_f-1)} \cdot p_L^2 + 2 \cdot \overline{\mu_f} \cdot p_L \cdot \overline{v_f} \cdot \frac{pp_L \overline{\mu_f}}{1-pv_f} + \overline{v_f(v_f-1)} \cdot \left(\frac{pp_L \overline{\mu_f}}{1-pv_f} \right)^2 \right] = \quad (23) \\
 &= p_L^2 \cdot \langle \mu(\mu-1) \rangle.
 \end{aligned}$$

Getting higher order terms will take more and more effort and it is necessary to resort to Mathematica to handle the derivations in a recursive manner for computational efficiency.

3. Numerical treatment

The analytical expressions derived for the number distributions were evaluated using the nuclear data that we obtained from MCNP-PoliMi. For neutrons, inclusion of the absorption does not affect the probabilities in a significant way. This behaviour is to be expected for the type of samples investigated in this work, since they were comprised of plutonium. The relevant fissile isotopes that could be investigated with this type of non-destructive assay (NDA) methods will in general show low absorption of neutrons.

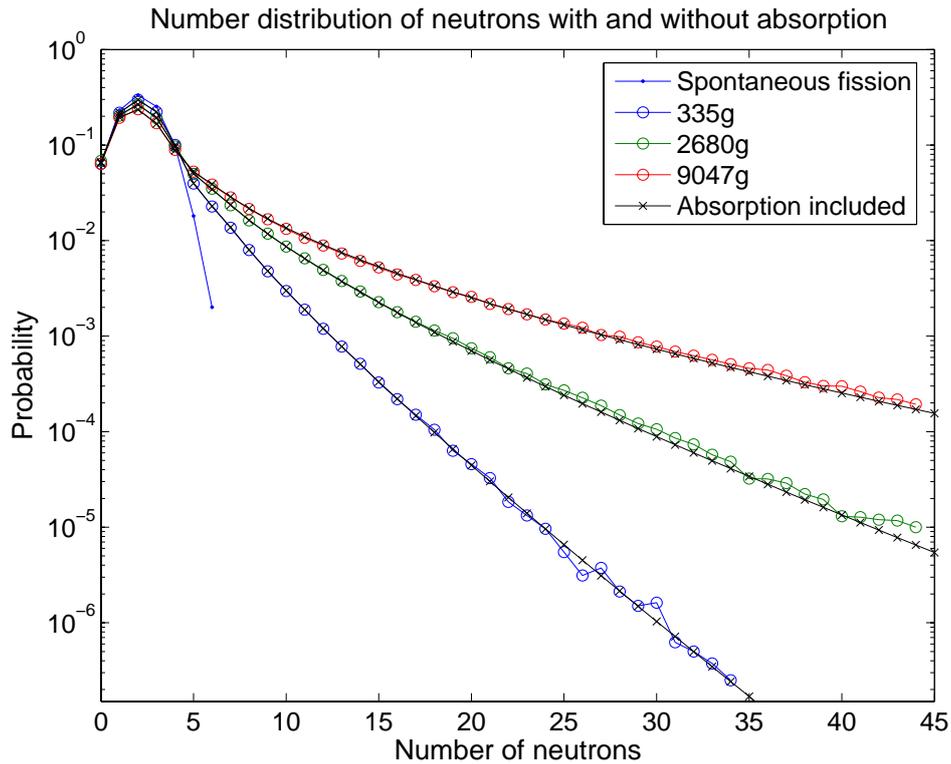


FIG. 1. The number distribution for neutrons calculated without absorption (MCNP-PoliMi) and with absorption (analytically). As can be seen the inclusion of absorption does not affect the neutrons much. Samples contained 80 wt% Pu-239 and 20 wt% Pu-240.

Fig. 1 shows that the absorption only changes the number distribution very slightly for neutrons in that the probabilities for very high numbers of neutrons escaping the target decrease a few percents in value.

In the case of gamma photons, the absorption will have a more complex influence on the number distribution. Since photons are generated by neutrons, their generation is influenced by how much the neutrons are affected by absorption, and their survival by how much the photons themselves are affected in form of self-shielding in the sample. Since plutonium is a heavy element, this means that the attenuation of gamma photons will be significant, as in all types of samples of high mass.

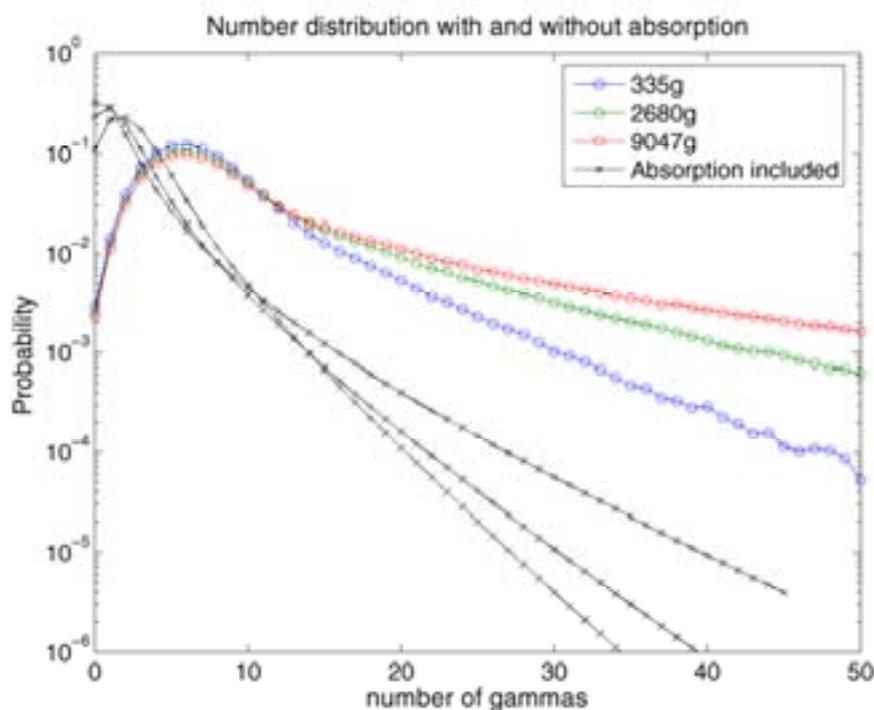


FIG. 2. The number distribution for photons calculated without absorption (MCNP-PoliMi) and with absorption (analytically). As can be seen the inclusion of absorption significantly reduces the number of photons leaking out from the sample. The samples contained 80 wt% Pu-239 and 20 wt% Pu-240.

In Fig. 2. the massive effect of the self-shielding is evident in all samples, with an increasing effect in heavier samples. A very notable thing is the fact that the curves with absorption included have reversed order for different sample masses, for high values of n , compared to the non-absorbing case. This means that the relative efficiency of neutrons and gamma photons depends on the sample mass. With low absorption, i.e. small sample mass, the gamma photons are favourable to observe and detect, but as absorption probability increases (or mass increases), multiplicity measurements with photons will be negatively affected. What regards neutrons, they are not particularly sensitive to absorption and can thus be detected with almost equal probabilities in the case of absorbing or non-absorbing samples. In the case where we have a certain material but different masses, then we notice that there is a turning point at a certain mass below which the gamma photons will have higher multiplicities as compared to neutrons, but above which it is the other way round. Thus for large samples, neutrons will be a far more efficient NDA-tool, while gamma photons are more efficient for small samples.

4. Conclusions

By introducing absorption into the models describing the number distribution of neutrons and gamma photons emitted from a fissile sample, one can obtain the actual number of escaped neutrons and photons which are available for detection. The results show that the absorption in a plutonium sample will have a much larger effect on photons as compared to neutrons. This means that there is a mass limit, above which neutrons become a better tool for NDA-studies for heavy samples. The change for factorial moments is very straightforward in the case of photons where a leakage probability, raised to

A. Enqvist et al.

the same power as the order of the factorial moment, decreases the numerical values of the factorial moments.

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Gamma-spectrometry for characterization of Pu-Be neutron sources

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Abstract. A NDA method for quantitative assay of Pu-Be neutron sources by HP-Germanium detector is described. Analyzing the structure of gamma spectrum in the 0-15000 keV energy range the value of around 1200 keV was obtained for the maximum energy of neutrons due to the effect of the interaction between neutrons and Germanium material via (n,γ) reaction. This effect and Compton tail of 4438 keV increase the background in the gamma spectra of Pu-Be neutron sources. Nevertheless, the commercial software MGA++ proved to be still reliable tool for analyzing the 0-600 keV energy range to derive the age and the isotopic composition of Pu material, although the part of its report about the existence of the Uranium is incorrect. Using three intensive gamma lines of Pu-Be, assuming far field assay of cylindrical sources the Pu content of the sources was determined with an uncertainty of up to 6%. In particular, the 375 and 413 keV lines were used for estimation of the Pu content and the 129 keV line for estimation of the dimensions of the sources. Combining this result with total neutron counting, the specific neutron yield can be verified. As a result of these measurements, it was shown that the claim of the supplier of the sources about two different sets of Pu-Be sources supplied to Hungary with different specific neutron yields is false.

1. Introduction

Plutonium-Beryllium (Pu-Be) neutron sources are an issue both for safeguards and illicit tracking because they contain nuclear material. During developing the NDA-methods for the quantitative assay of these sources, a single 0-600 keV gamma spectrum of each source, measured by a planar HPGe detector in the far field of cylindrical sample was used to estimate the Pu content, the Pu isotopic composition and age of Pu material [1]. Some reports by the MGA++ code [2] used to analyze the gamma spectrum contain an entry about the existence of U isotope in the Pu-Be sources. This fact required carefully investigating the results. The intrinsic calibration method was used to verify the report of the MGA++ code. The effect originating from interaction between the neutrons and germanium material on gamma spectra was investigated by analyzing the structure of the 0-1500 keV energy spectrum of Pu-Be sources measured by coaxial high-purity germanium detector. In this paper we also show how to reduce the measurement uncertainty to below 6% for estimation of the Pu content. An advanced version of a previously developed method [3] relying on the combination of gamma spectroscopy and neutron measurement, the “advanced combination method” or “advanced-CBM” was used for determining the specific neutron yield of Pu-Be neutron sources.

2. Gamma spectra of Pu-Be neutron sources

The 0-15000 keV energy range spectra of Pu-Be, Am-Be, pure Pu and background measured by 150 cm³ coaxial HP Germanium detectors are shown in Figure 1.a. The peak of 4438 KeV and its Compton tail can be observed clearly. It is well known that the ratio of the yield of the 4438 keV gamma-line, γ_{4438} , and of neutron output, N , is a constant of about 0.53. Therefore, the count rate of the 4438 keV line can also be used to derive the N of the Pu-Be sources by gamma

spectrometry [4,5] instead of using neutron counting. Interestingly, the region of the spectra above 4500 keV can be observed clearly too. The origin of the counts in this region can be identified as the interaction of neutrons with germanium via (n,γ) reaction. By this way the value of around 1200 keV for maximum energy of neutrons from (α,n) reaction was confirmed. This value was well known as determined by other method [6,7], however, this is the first time that it has been confirmed by pure gamma spectrometry. It has been established that the Compton tail of 4438 keV and effect of (n,γ) reaction increase the background of the 0-600 keV spectra of Pu-Be as well as Am-Be sources, by nearly two orders of magnitude with respect to the background in the spectrum of pure Pu.

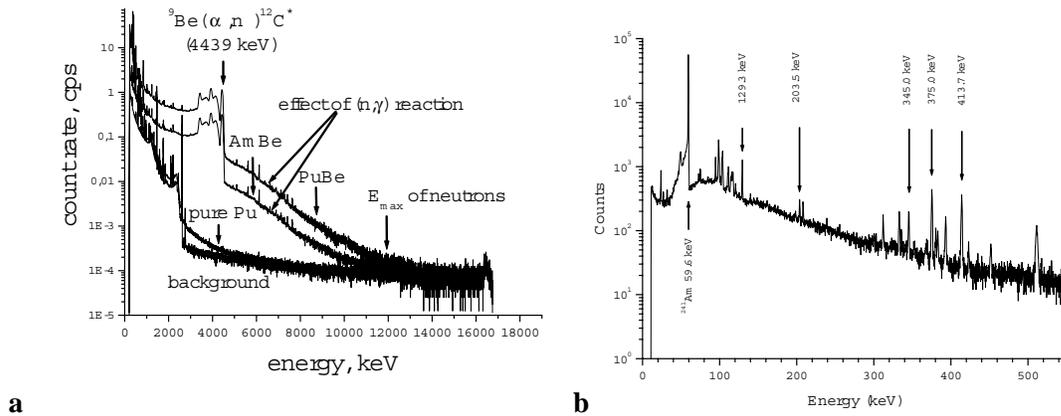


Fig. 1.a.-1500 keV gamma spectra of Pu-Be, Am-Be, pure Pu and background by 150 cm^3 coaxial HP-Ge detector; and Fig. 1.b. a typical 0-600 keV gamma spectrum of Pu-Be [1] by Planar HP-Ge detector.

Investigating all γ -peaks in 0-600 keV spectra, no gamma peaks created in an interaction between the neutrons and germanium coincidence with gamma lines coming from the decay of Pu and Am isotopes. This is very important from the point of view of applying the commercial software MGA++. The (n,γ) reaction and the Compton tail of 4438 keV mainly increase the background of the 0-600 keV spectra, which leads to decreasing the sensitivity of MGA++. Nevertheless, MGA++ still operates correctly the results given by MGA++ about the age and isotopic composition of Pu material is acceptable.

In some reports of MGA++ for Pu-Be neutron source with large neutron output, there was an entry about the presence of U in the sample. Carefully investigating these cases by using an intrinsic efficiency calibration curve constructed from the gamma peaks of Pu and Am isotopes show that the part of the report about the existence of ${}^{235}\text{U}$ was incorrect. However, the part of the report about age and isotopic composition concerning the Pu and Am isotopes was still correct.

3. Determination of Pu content by pure gamma spectrometry

3.1. Calculation model

As reported in Ref. [1], the total Pu content, M_E , can be determined from the count rate C_E of two intensive gamma lines 375 and 413 keV measured in the far field of a cylindrical sample as below:

$$M_E = \frac{C_E}{f_{239} G_E O_E} F_E(P) \quad (1)$$

where f_{239} is the ^{239}Pu isotopic composition, G_E ($(\text{gs})^{-1}$) is the specific gamma yield, O_E is the detector efficiency and F_E is the correction factor for absorption as a function of a set of parameters of the source P .

F_E is the product of a factor accounting for self-absorption, F_{IE} , and a factor accounting for absorption in the steel wall of the container, F_{2E} . Assuming that the gamma-rays are parallel to the detector axis, F_{IE} can be calculated using the far-field approximation for a cylindrical sample viewed along a diameter [8]:

$$F1_E = \frac{1}{2} \frac{\mu_{IE} D}{I_1(\mu_{IE} D) - L_1(\mu_{IE} D)} \quad (2)$$

while

$$F2_E = \exp(\mu_{ICE} d_C) \quad (3)$$

where

- μ_{IE} is the linear attenuation coefficient of the Pu-Be material,
- D is the diameter of the sample,
- I_1 is modified Bessel function of order 1 [9],
- L_1 is modified Struve function of order 1 [9],
- μ_{ICE} is the linear attenuation coefficient of the steel container taken from Ref. [9] and
- d_C is thickness of the container wall.

Furthermore, μ_{IE} and D can be represented from parameters of the source as well as from

$$\mu_{IE} = \mu_E \rho = \left(\frac{239}{239 + 9n} \mu_{PuE} + \frac{9n}{239 + 9n} \mu_{BeE} \right) \rho \quad (4)$$

$$D(\rho, R, n, M_{Pu}) = \sqrt[3]{\frac{4M_{Pu} * 239 + 9n}{\pi \rho R}} \quad (5)$$

where

- ρ is the density of the sample,
- μ_{PuE} and μ_{BeE} were the mass attenuation coefficient (cm^2/g) of Pu, and Be, respectively [10],
- n is the ratio of the number of Be atoms to the number of Pu atoms (PuBe_n), and 239 and 9 are the atomic mass numbers of Pu and Be, respectively,
- $R=H/D$ is the ratio of the height and the diameter of the Pu-Be material,

It means that F_E is a function of the set $P \{n, \rho, d_C, R\}$.

Estimation of M_E requires the values of P . The n , ρ and d_C are well known ($n=13$, $\rho=3.7 \text{ g/cm}^3$ and d_C from certification form), however, R was an unknown parameter in general. In ref. [1] assuming that the form of Pu-Be material is similar the cylindrical form of container, the R is

assumed to be equal to by ratio of the height and the diameter of inner container, which can be found in the certification form from the supplier. This assumption did not cause a large uncertainty of the final result [1]. By this way, the P became known and Pu content was derived due to Eq. 1. The final Pu content, M_{Pu} , was considered as average of two results by 375 and 413 keV:

$$M_{Pu} = \langle M_E(C_{375}); M_E(C_{413}) \rangle \quad (6)$$

3.2. Reducing the systematical error of F_{IE} -function.

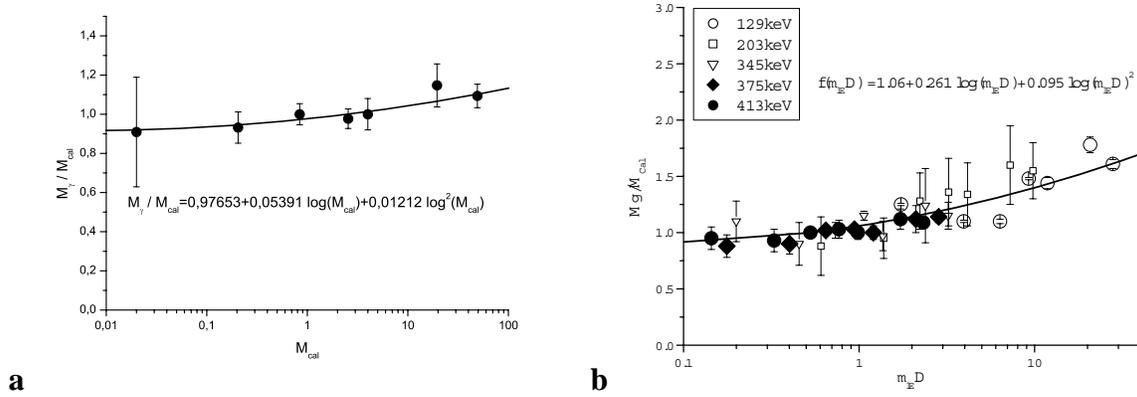


Fig. 2.a. The ratio of M_γ/M_{cal} versus M_{cal} . [1] and b- The ratio of M_γ/M_{cal} versus $\mu_{IE}D$.

Figure 2.a. shows the ratio M_γ/M_{cal} versus M_{cal} [1]. Here M_γ and M_{cal} are the Pu content obtained by the puregamma-spectroscopy [1] and by calorimetry [11], respectively. As reported in ref. [1], a bias of up to 12% can be observed between the two methods, however, it can be seen that the ratio M_γ/M_{cal} increases monotonically with M_{cal} reflecting the systematical error of the gamma method. The difference is bigger for larger values of M_{Pu} . This behaviour can be explained by the dependence of the systematical error of F_E on $\mu_{IE}D$ in formula 2. For developing this idea, the M_E by C_E of 129, 203, 345 keV lines were calculated, and the ratio M_E/M_{cal} of all lines was plotted versus $\mu_{IE}D$ (Fig. 2.b). The picture confirmed the systematical error of F_E on $\mu_{IE}D$ in formula 2. By fitting an empirical function one obtains:

$$f(\mu_{IE}D) = 1.06 + 0.261 \log(\mu_{IE}D) + 0.095 \log^2(\mu_{IE}D) \quad (7)$$

Consequently, using $F_E = F_{IE}F_{2E}f$ instead of $F_E = F_{IE}F_{2E}$, a better correction factor for absorption can be obtained. For example, using the 375 and 413 keV lines, with this correction factor the difference between M_γ and M_{cal} becomes less than 6% in a wide range of $\mu_{IE}D$.

3.3. Using the 129 keV for estimating R

After using the new function for self-absorption, however, M_E by 129 keV was still different from M_{cal} by more than the statistical error of the count rate (see data of 129 keV on Fig. 2b). It can be explained by the fact that the value of C_E of 129 keV strongly depends on R [1]. Using this aspect, a procedure for estimating the R was considered. In particular, an “effective R ”, R_{eff} , can be obtained by equating the M_E by 129 keV to M_{Pu} , obtained as the average of M_E by 375 and 413 keV, i.e., the value of R_{eff} can be derived from equation

$$M_E(C_{129}, R_{eff}) = \langle M_E(C_{375}, R_{eff}); M_E(C_{413}, R_{eff}) \rangle \quad (8)$$

In conclusion, a calculation model using the 129 keV for determining the value of R allows the estimation of the Pu content from the 375 and 413 keV lines with an error bellow 6%.

3.4. Estimating M_{Pu} for sources without a certification form

For these sources the parameter d_C was unknown. In this case we adopted a “usual” value, for d_C , then used the above procedure for estimating the Pu content. In our case a usual value is $d_C = 3.5 \pm 0.5$ mm. This assumption increases the error of the final result for merely few percents.

For a double container, the value of d_C was taken to be $d_C = 7 \pm 1$ mm. The error in this case was shown to be bellow 10%

For sources without any information, unknown parameter, the M_{Pu} was estimated by te following steps:

- Using the “infinite energy method” [12] as applied in ref. [1] for estimation of the M_{Pu-inf}
- Comparing M_{129} and M_{413} by Eq. 8 for deriving the effective thickness of the container, d_{C-eff} with adopted values of R_{eff}
- Changing d_{C-eff} and R_{eff} based on Eq. 8 for three gamma 129, 375 and 413 keV lines

The tests of this procedure by a computer program have shown that the error of the final M_{Pu} is bellow 10-15%. The systematical tests will present in detail elsewhere.

5. Monitoring the specific neutron yield of Pu-Be neutron sources by combination with neutron measurement: an advanced-CBM

The relation between the total neutron emission rate, N , and the total Pu content, M_{Pu} , can be expressed as below:

$$N = M_{Pu} \sum_i (f_i y_i) M, \quad (9)$$

where f_i and y_i are the isotopic fraction and the specific neutron yields of individual Pu isotopes, respectively, and M is the neutron self-multiplication factor.

In a previous work [12], the N was measured by a calibrated neutron counter, the isotopic composition, f_i , by gamma spectrometry. Correcting the M was performed by neutron coincidence counting and the specific neutron yields, y_i , were calculated from data adopted from literature and the total Pu content, M_{Pu} , was derived by using the relation expressed by Eq. 9. This was called as combination method, CBM.

Since M_{Pu} can be exactly estimated by gamma spectrometry, Eq. 9 can be considered for monitoring the values of the specific neutron yields, y_i . As the first result, in the ref. [1], it was shown that the specific neutron yields of the Pu-Be sources may be larger than the values used in the CBM by the factor of about 1.26 ± 0.1 . However, if using the M_{Pu} by gamma method corrected by equation 7, i.e., normalized by calorimetrical method, the specific neutron yields of ^{239}Pu and ^{240}Pu should be corrected by an additional factor of 1.13. Cosequently, the total factor is $1.26 \times 1.13 = 1.42$. The new set of the specific neutron yields is: ^{238}Pu : 2.89E7, ^{239}Pu : 8.86E4, ^{240}Pu : 3.25E5, ^{241}Am : 5.76E6 with unit of n/g.s. These values are now also used in Ref. [13].

The values of the specific neutron yield in Ref. [3] were derived based on bombardments of the different energy α -beams from accelerator on pure Be target in 2π -configuration. However, the α -particle of Pu isotopes bombard in 4π -configuration on complex PuBe_{13} target. The part of α -beam by Coulomb scattering on 2π -configuration may come back to air and does not create the neutrons via (α,n) -reaction. Beside that the specific neutron yields depend on the isotopic composition of the sources. For example, specific neutron yield of Am in Am-Be is about $8.7\text{E}6$ n/g.s., which is much more than that of Am in Pu-Be sources above. Therefore, there is a difference of the values between specific neutron yields for different configurations and also for different “types” of α -particle.

Using new values of specific neutron yield essentially improved the accuracy of the CBM. Precision of CBM is now about 6% [13]. Basis on the precision of the two methods, the relationship expressed by equation 9 is considered to verify the specific neutron yields. A definition of a new parameter k is:

$$k = \frac{N/M}{M_{\text{Pu}} \sum_i f_i y_i} \quad (10)$$

For example, if using the new set of specific neutron yields, the value of $k=1$ with error of 12% is typical for Pu-Be sources imported to Hungary. Applying equation 10 for Pu-Be neutron sources the same results k ($k=1\pm 0.1$) was received for all Pu-Be sources imported to Hungary. This means that the claim of the supplier, stating that the specific neutron yield of the sources imported to Hungary before and after 1978 is different, is false.

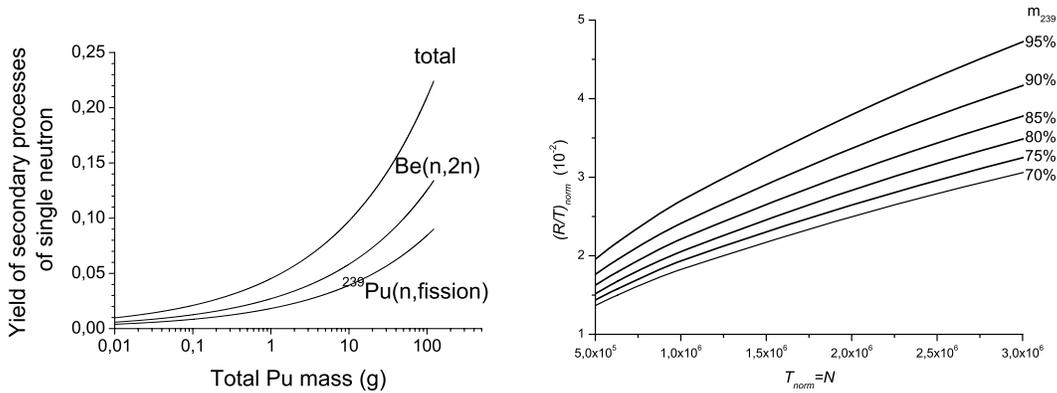


Fig. 3. Yield of secondary process of single neutron [14].

Fig. 4. The ratio $(R/T)_{\text{norm}}$ versus neutron output, N , with different specific neutron yield [15].

Note that in this paper the M was treated from the yield of secondary neutron of single neutron via $\text{Be}(n,2n)$ and $\text{Pu}(n,f)$ reaction as shown on Fig. 3 [14]. The value of M versus M_{Pu} can be expressed by an empirical equation.

$$M = 1 + 0.034 \log M_{\text{Pu}} + 0.0153 \log^2 M_{\text{Pu}} + 0.00283 \log^3 M_{\text{Pu}} \quad (11)$$

6. Relationship between pure gamma and pure neutron method.

As reported in [15], after the doubles and singles count rate (R_{norm} and $T_{norm} = N$, respectively) measured by neutron coincidence technique were normalized for parameter of neutron coincidence collar, a couple of physical parameters were received for Pu-Be neutron sources. It means that R_{norm} can be used for characterization of Pu-Be neutron sources as well as the neutron output $T_{norm} = N$. The couple $[(R/T)_{norm}, T_{norm} = N]$ can be used for monitoring the Pu-Be neutron sources by any coincide neutron collar relying on the pure neutron method according to a relationship plotted in Fig. 4 due to following equation:

$$(R/T)_{norm} = 1.787 \left(\frac{N/M}{a + b * f_{239}} \right)^{0.44 + 0.024 \ln \left(\frac{N/M}{a + b * f_{239}} \right)} \quad (12)$$

where $y = a + b * f_{239}$ was total specific neutron yield of the source with $a = (21.4 \pm 2.1) \times 10^5$ and $b = -(0.214 \pm 0.01) \times 10^5$ n/s-g Pu. The M can be represented by Eq. 10.

In case of using only gamma measurement, the value of N can be calculated by Eq. 9 and the $(R/T)_{norm}$ can be derived from an empirical equation [16]:

$$(R/T)_{norm} = 1.787 M_{Pu}^{0.44 + 0.024 \ln M_{Pu}} \quad (13)$$

By this way the values of the couple $[(R/T)_{norm}, T_{norm} = N]$ and M can also estimated from the results of the pure gamma method without of using any neutron collar and can be used for monitoring the sources by pure neutron method later.

Conclusion

For the first time, the gamma spectra in 0-1500 keV energy range of Pu-Be sources were measured by HPGe detector. The maximum energy of around 12000 keV of neutrons was obtained too. The effect of (n, γ) reaction and the Compton tail of 4438 keV line increase the background of gamma spectra of Pu-Be neutron sources, however, the Pu isotopic composition and age by reported by the commercial software MGA++ is still acceptable. In case the software reports the existence of U in the sample, the results must be verified by intrinsic calibration method carefully.

Using three intensive gamma lines of 129, 375 and 413 keV of ^{239}Pu measured in the far field of a sample assumed to be cylindrical, the Pu content was determined with an error of 6%. Combining these results with the neutron measurement, an advanced CBM was considered for verifying the specific neutron yield. As a first result, the existence of two sets of values for the specific neutron yields given by supplier for Pu-Be imported to Hungary must be rejected. Moreover, the couple of $[(R/T)_{norm}, T_{norm} = N]$, which is needed for the verification by pure neutron method, can be derived from the results of pure gamma measurement.

With error of about 6%, the Pu-Be sources assayed by gamma spectrometry can be used as reference sources for calibrating NDA methods in other locations, i.e., without the need of a costly and complex calorimeter. For inspection, it is enough if the inspector uses one of two

procedures: pure gamma or pure neutron measurement. Of course, if both methods are used, more information can be derived about the sources using the advanced CBM.

Acknowledgements

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Development of an XRF analyser with preliminary energy selection filter for screening environmental samples

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Abstract. A new X ray fluorescence (XRF) analyser for the measurement of the total quantity of uranium on environmental swipe samples is under development at the IAEA Clean Laboratory for Safeguards, a unit of the Safeguards Analytical Laboratory (SAL). The system is an energy-dispersive XRF analyser with a high-power 4 kW X ray tube excitation source and a preliminary energy selection filter (PESF). The filter is installed between the silicon lithium, Si (Li), detector and the sample and selects the energy range containing the uranium L_{α} lines while rejecting the X ray quanta with other energies.

This paper reports on the general concept of the system as well as its main technical features and operating parameters. Folding of the swipe sample as a measure to improve the U detection limit is investigated. The measured transmission function of the PESF shows that the filter increases the efficiency of the U L_{α} radiation detection by a factor of ~ 2 -times and suppresses the X ray quanta of other energies by a factor of ~ 200 times. The instrument prototype design is described.

1. Introduction

Screening of environmental swipe samples at the Safeguards Analytical Laboratory (SAL) is performed with an X ray fluorescence (XRF) scanning system TRIPOD [1]. This instrument measures the uranium (U) distribution on the swipe with a detection limit (DL) of 35 ng/cm^2 . The measurement time (T_m) for one swipe is ~ 4 hours. This long measurement time is the limiting factor in the total analysis time and sample throughput. The reporting of the DL as ng/cm^2 is another disadvantage because the required value is a DL for the whole swipe. Simply multiplying the DL [in ng/cm^2] for the total swipe surface (100 cm^2) gives an erroneously high estimate of the DL. These disadvantages were the reasons for the development of a new XRF analyser for screening without scanning the swipes and with a significantly shorter T_m .

2. Swipe folding

The typical swipe sample is a square $10 \text{ cm} \times 10 \text{ cm}$ piece of cotton packed in a plastic bag. Typically, what is visible by the detector in energy-dispersive and wavelength-dispersive XRF systems does not exceed a circle of 3 cm diameter ($\sim 28 \text{ cm}^2$), which does not cover the total whole swipe surface (100 cm^2). Folding of the swipe allows the making of a 'one shot' measurement of the whole swipe. The folded swipe surface is 2^n times smaller than the unfolded swipe ($n =$ number of times the

swipe is folded). After folding four times, the total swipe area is a square 2.5 cm x 2.5 cm, which is within the solid angle visible to the detector. Folding also has negative effects on the screening results, namely:

- (a) Absorption of the analytical signal ($U L_{\alpha}$ X ray line) in the swipe tissue; and
- (b) Scattering of the X rays from the swipe.

These effects were estimated with TRIPOD. It was found that one additional layer of swipe tissue absorbs about 6% of the $U L_{\alpha}$ X rays and increases by 16% the background under the $U L_{\alpha}$ peak.

Normally, the U distribution on the swipe is unknown and this introduces additional error in the case of folded swipes because of the absorption effect. The U distribution is variable from swipe to swipe, but it has some typical features due to the swiping procedure in use [1,2]. Figure 1 shows the average U distribution integrated over all swipes measured with TRIPOD in 2005. One can see that the U distribution has its maximum in the central area of the swipe.

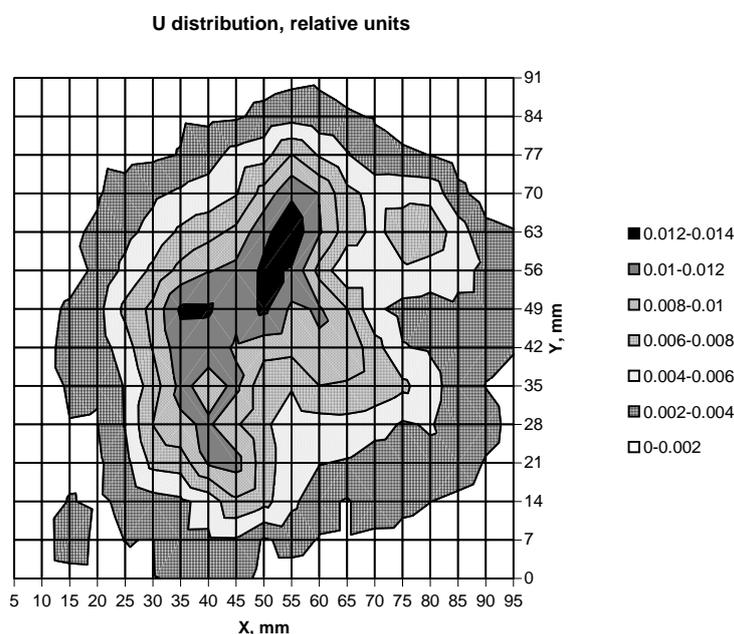


FIG. 1. Average U distribution on the swipes measured with TRIPOD in 2005.

Figures 2 and 3 show, respectively, the U distributions on a typical unfolded swipe and on the same swipe folded four times. The measurements were performed with TRIPOD. The swipes were folded as follows: first and second folding - square 5 cm x 5 cm; third folding - rectangle 2.5 cm x 5 cm; and fourth folding - square 2.5 cm x 2.5 cm.

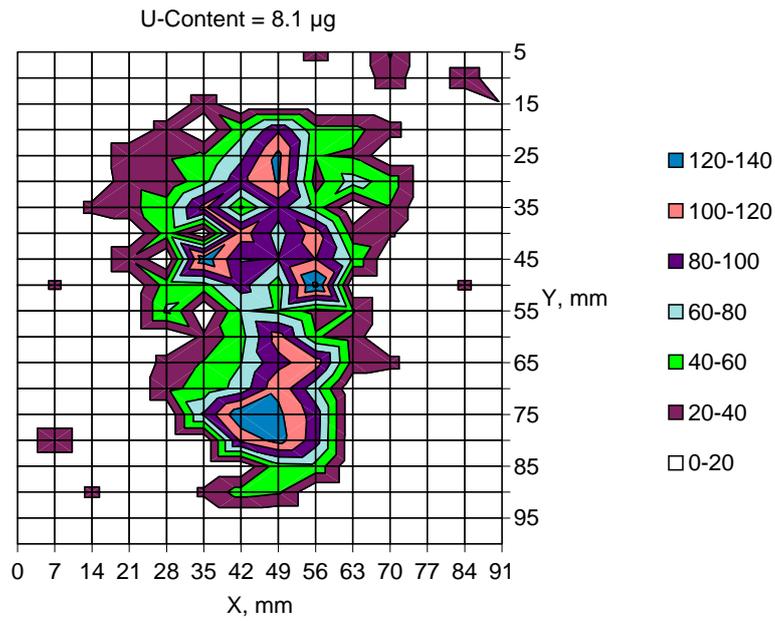


FIG. 2. U distribution on the unfolded swipe.

The U content was evaluated with one calibration curve in both cases. The unfolded swipe was measured in 280 measurement positions covering the total swipe surface. The folded swipe was measured in 40 positions covering an area of 2.5 cm x 2.5 cm.

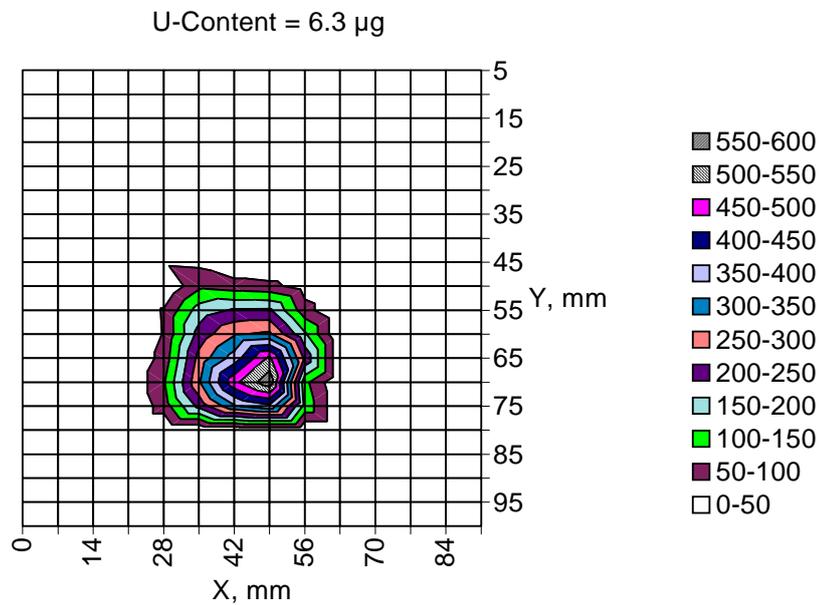


FIG. 3. U distribution on the folded swipe.

One can see (Figure 3) that the U distribution on the folded swipe has a maximum at one of the corners of the square. It results from the original U distribution on the unfolded swipe (Figure 2) and the selected folding method. The absorption of the U X rays in the swipe tissue gives a 25% underestimate of the U content in the folded swipe. This effect can be reduced by performing a calibration of the system with a reference swipe having a typical U distribution (Figure 1).

Folding of the swipe increases the effective U concentration per unit area of the sample. The U L_{α} line intensity would increase proportionally to the U concentration if there were no absorption and scattering effects. Having estimates of these effects (see above) one can compare the U DL for the folded DL_{fold} and the unfolded DL_{unfold} swipes assuming that the swipe is folded four times:

$$DL_{\text{fold}}=3*m*\text{sqrt}(B_{\text{fold}})/S_{\text{fold}} \approx 3*m* \text{sqrt} (B_{\text{unfold}}*1.16^{16})/[16*S_{\text{unfold}} / (1.06^{16})]=0.33*DL_{\text{unfold}}, \quad (1)$$

where m is the U mass and B and S are background and peak areas, respectively. It is assumed that U L_{α} X rays pass through an average of eight layers of the swipe tissue.

Measurements of the unfolded and the folded swipes with TRIPOD confirm this relation. The conclusion can be drawn that swipe folding improves the overall U DL.

3. Instrument prototype design

The main components of the prototype XRF screening instrument are: Si(Li) detector, the PESF, measurement chamber, X ray excitation system, control and spectrometer electronic modules and a personal computer. Figure 4 shows the general view of the central part of the instrument. The Si(Li) detector is in a vertical dipstick cryostat with a 25 μm Be window and 30 litre Dewar. It detects X rays coming from the PESF installed between the measured swipe and the detector. The PESF design is described in the next section. It is fixed inside a stainless steel cylinder and is not visible in Figure 4.

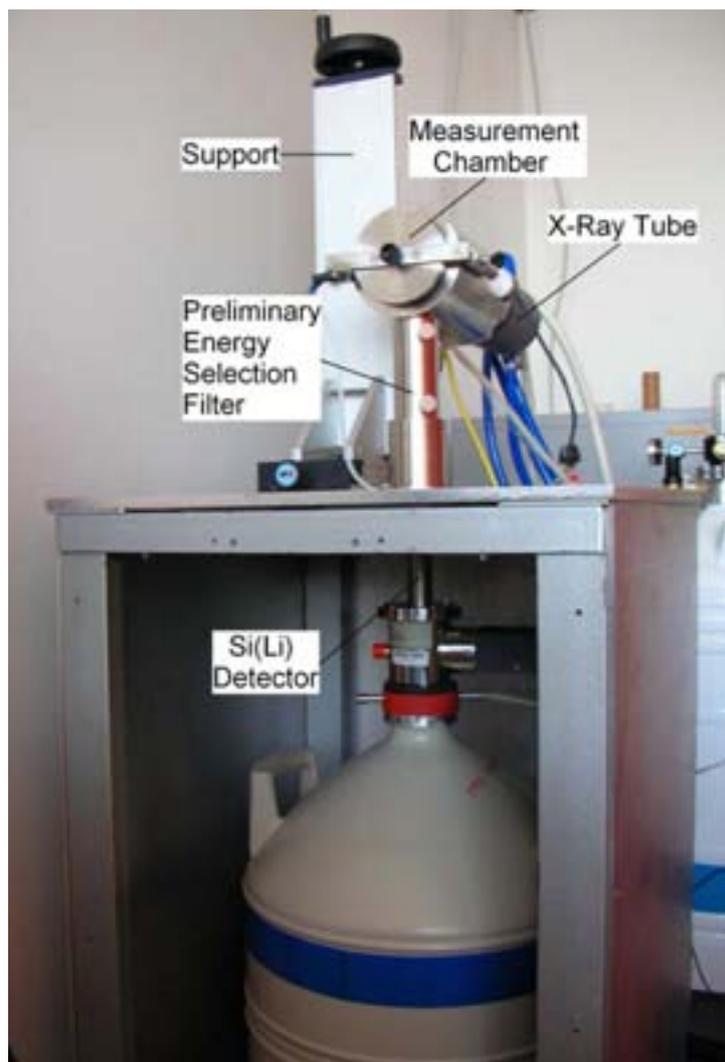


FIG. 4. General view of the central part of the XRF analyser.

The X ray beam from the X ray tube enters the measurement chamber through a Ag filter and irradiates the swipe at an angle of 45 degrees (Figure 5). The folded swipe is fixed in the sample holder which is attached to the entrance 'door' of the measurement chamber. The fluorescent X ray beam from the swipe passes the collimator and reaches the PESF.

The X ray excitation system consists of a 4 kW super sharp X ray tube with Rh target material, a high voltage generator and a cooling system. The X ray tube is embedded in a cylindrical stainless steel chamber (X ray tube chamber) coupled with the measurement chamber. The internal volume of the X ray tube chamber is ventilated with N_2 released from the Dewar. This protects the Be window of the tube from corrosion. A tungsten safety shutter is installed in front of the Be window.

The high voltage generator is fully operated from the personal computer. The cooling system has internal and external water circuits. The internal circuit is filled with de-ionised water. The external circuit is connected with a chiller.

The electronic control module checks the temperature and water flow in the internal and external circuits, the status of the measurement chamber entrance door (opened /closed) and the shutter. It blocks operation of the high voltage generator if the controlled parameters do not achieve safe conditions for the tube and the operator.

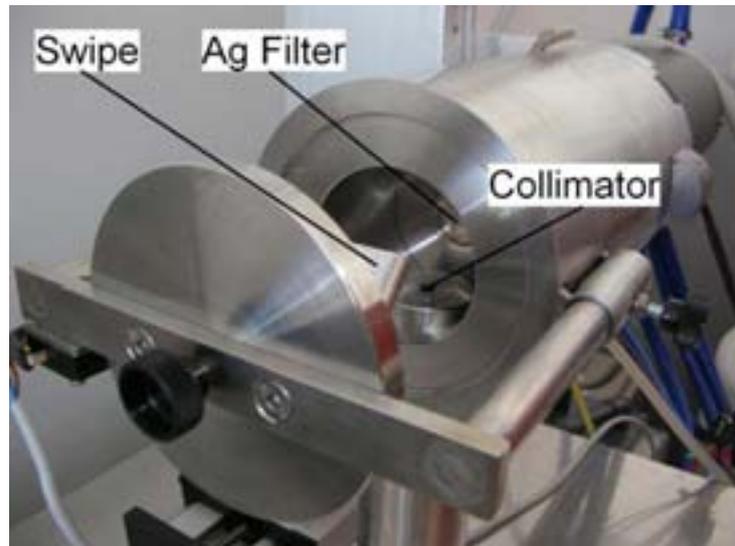


FIG. 5. Measurement chamber.

The spectrometer electronic modules process the detection of the fluorescent X ray quanta. High count rates can be processed with a digital signal processor or with a fast spectrometric amplifier and analogue to digital converter (ADC).

4. Preliminary energy selection filter

The PESF is a pyrolytic graphite Bragg reflector with a cylindrical shape (Figure 6). It selects the energy range containing the U L_{α} lines and rejects X ray quanta with other energies [3]. Two 'Beam Stops' prevent direct X rays emitted by the swipe to pass through the filter.

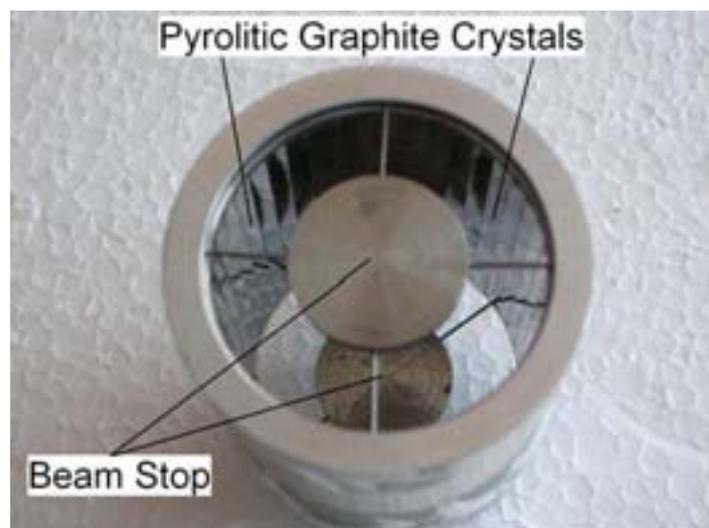


FIG. 6. Preliminary Energy Selection Filter (PESF).

The transmission function (I / I_0) of the PESF was measured with ^{241}Am and ^{55}Fe sources (Figure 7). The quantity I / I_0 is a ratio of the intensity of the X ray line measured with/without the PESF. One can see that the PESF depresses by ~ 200 times (or more) the X ray intensity outside an energy window at 12-16 keV ($E[\text{U } L_\alpha] = 13.6 \text{ keV}$) and increases it inside the window by up to a factor of 2. By installing the PESF, both the scattering radiation of the excitation source and the low energy 'tail' of this radiation in the spectrum due to the incomplete collection of the charge in the detector are reduced by several orders of magnitude. This improves significantly the U DL.

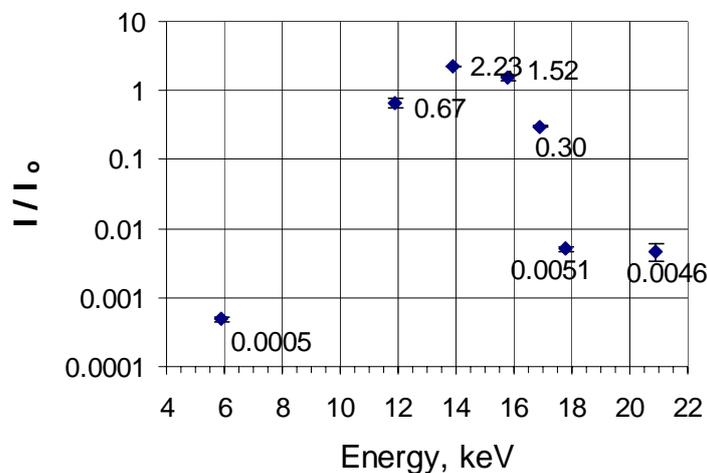


FIG. 7. Transmission function of the PESF.

5. Future activities

The next steps for the development of the XRF analyser system are:

- Measurement of the response profile;
- Selection of the measurement mode (static, rotation, etc.);
- Optimisation of the excitation parameters and detection geometry;
- Design and manufacture of a sample changer;
- Calibration of the system (using reference swipes, quality control (QC) samples); and
- Validation.

The response profile is a $\text{U } L_\alpha$ line intensity profile from a point source U sample, measured across a 5 cm wide swipe. This size corresponds to the measurement of a swipe folded two times. The profile depends on the area excited by the X ray tube and the solid angle of the system PESF + Si(Li) detector projected on the swipe. The measured data will provide information necessary to select the measurement mode. If the response profile is highly inhomogeneous across the swipe area, then a rotation of the folded swipe should be performed during measurement. Either symmetric or asymmetric rotation can be selected.

The excitation parameters to be optimised are: X ray tube HV, current, filter and collimation of the X ray beam from the tube. The following detection parameters should be optimised: PESF and Si(Li) collimators diameters and all distances (swipe – collimator – PESF – Si(Li) detector). The swipe folding method can be optimised using a typical U distribution on a swipe. The simple method presented in this paper leads to the 'concentration' of U at one of the corners of the folded swipe. A

D. Donohue et al.

better folding method should provide the optimum match between the maximum U concentration on the swipe and the maximum sensitivity of the analyser.

Two types of a sample changer can be used: a robotic arm or an electro-mechanical device. The measurement chamber design will have to be modified in either case. Additional shielding of the measurement area may be necessary.

The system will be calibrated with artificial reference swipes and a U point source sample. QC samples should have stable geometrical characteristics to avoid additional uncertainties caused by the swipe positioning. Validation of the XRF analyser will be performed according to the ISO 17025 guide.

6. Conclusions

A new XRF analyser for screening environmental swipe samples is under development at the IAEA Clean Laboratory for Safeguards. The system will provide an estimate of the total U content on the swipes. A high power X ray tube and a PESF allow the performance of fast screening measurements with a low U detection limit for the whole swipe. The prototype instrument has already been constructed and is under testing.

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Performance and validation of COMPUCEA 2nd generation for uranium measurements in physical inventory verification

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Abstract. A new instrumental version of COMPUCEA has been developed with the aim to provide a simplified and more practical instrumentation for in-field use. The main design goals were to eliminate the radioactive sources and the liquid nitrogen-cooled Ge detectors used in the 1st generation of COMPUCEA. This paper describes the major technical features of the 2nd generation of equipment together with typical performance data. The performance tests carried out during first in-field measurements in the course of physical inventory verification campaigns represent an important step in the validation of this new instrument.

1. Introduction

COMPUCEA (Combined Procedure for Uranium Concentration and Enrichment Assay) describes both a method and instrument used for verification measurements in uranium facilities [1, 2, 3]. The 1st generation of COMPUCEA used since several years utilised a K-edge densitometer with a mixed ⁵⁷Co/¹⁵³Gd radiation source and a HPGe detector for the determination of the uranium concentration, and passive gamma counting of the 186 keV photons from ²³⁵U with a HPGe well counter for the enrichment measurement. The respective in-field measurements during the physical inventory verification (PIV) campaigns in European Low-Enriched Uranium (LEU) fuel fabrication plants are carried out by analysts from the Institute for Transuranium Elements (ITU), Karlsruhe in support of Euratom and IAEA safeguards inspectors.

With the primary objective of producing a simplified instrumental variant better adapted to the requirements for in-field use, a 2nd generation of COMPUCEA equipment has been recently developed at ITU [4]. The primary design goals were (i) to eliminate the radioactive sources needed in the 1st generation, and (ii) to make use, if possible, of detectors which do not require cooling with liquid nitrogen. Both goals have been fully met for the uranium element assay part of COMPUCEA. For the enrichment assay part the investigations have shown that the use of a room-temperature detector is also possible at comparable measurement precision but with somewhat reduced measurement accuracy.

A prototype of the 2nd generation COMPUCEA equipment has been carefully tested and evaluated. It has been also used in some of the PIV campaigns during 2005/2006 for first in-field application in parallel with the existing 1st generation of equipment. This paper summarises the actual status of the project, which also constitutes a task in the support programme of the Joint Research Centre of the European Commission to the IAEA.

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2. Measurement methods and equipment used in 2nd generation

The layout and principle performance features of the new equipment have been recently described in detail [4]. Only the most salient features, and some modifications made since then, are therefore briefly summarised below.

2.1 Uranium concentration measurement

The majority of the samples to be analysed during the PIV campaigns are of solid form (powders, pellets), for which the COMPUCEA measurement requires a prior sample dissolution. Because of this additional step of sample preparation, an assay with COMPUCEA represents more a kind of radio-analytical analysis involving careful analytical procedures (like quantitative dissolution, solution density measurements, quantitative aliquoting etc.) than a mere spectrometry measurement.

For the uranium concentration measurement we have chosen L-edge densitometry as a practical alternative to the K-edge densitometry measurement applied in the previous COMPUCEA instrument. With the choice of the L-edge technique the two basic design goals, namely to eliminate the radioactive sources and the liquid nitrogen-cooled HPGe detector, could be realized. The new COMPUCEA equipment for the uranium concentration measurements is based on the following hardware components as shown in Fig. 1:

- A compact X-ray generator with maximum ratings of 30 kV and 100 μ A (model Eclipse-II from Amptek Inc.) with a Ag target transmission tube and associated controller as radiation source;
- A Peltier-cooled 10 mm² x 0.5 mm Si Drift Detector system with associated preamplifier/amplifier and power supply (model AXAS-SSD10 from KETEK GmbH) for high pulse rates, offering an energy resolution of 142 eV at 5.9 keV at a pulse shaping time constant of 0.5 μ s Gaussian equivalent;
- A digital signal processor (model DSA 1000 from Canberra);
- A shielding/collimation assembly with exchangeable sample adapters for alternative L-edge densitometry and XRF measurements.

The total weight of this part of the new COMPUCEA, including the shielding for the X-ray tube, amounts to about 6 kg (without Laptop), which meets the design goal of true portability.

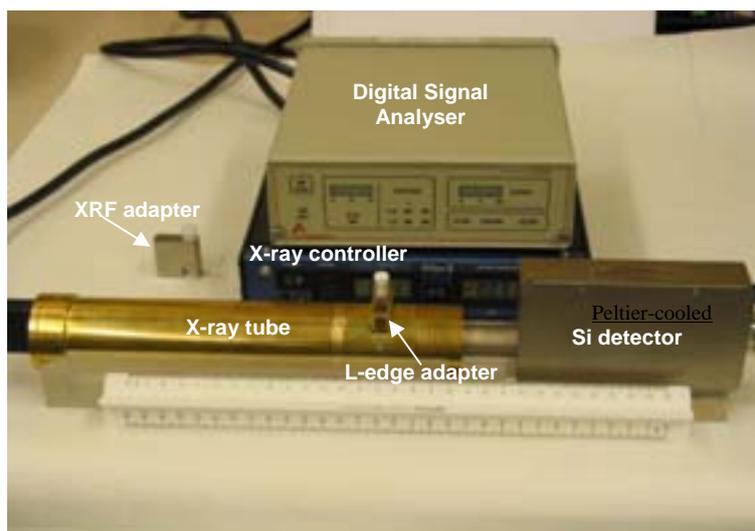


Fig. 1. Photograph of the L-edge/XRF densitometer for the new COMPUCEA.

The principle measurement configurations for the L-edge densitometry (and optional XRF measurements) are schematically outlined in Fig. 2. For the L-edge densitometry measurement a collimated straight-through beam from the X-ray tube is passing through a precision quartz cuvette containing the uranium measurement solution towards the Si drift detector. A second X-ray beam is directed towards a Ti/Ge fluorescence target to produce characteristic Ti and Ge K X-rays in the L-edge spectrum for energy calibration and instrument control (Fig. 3).

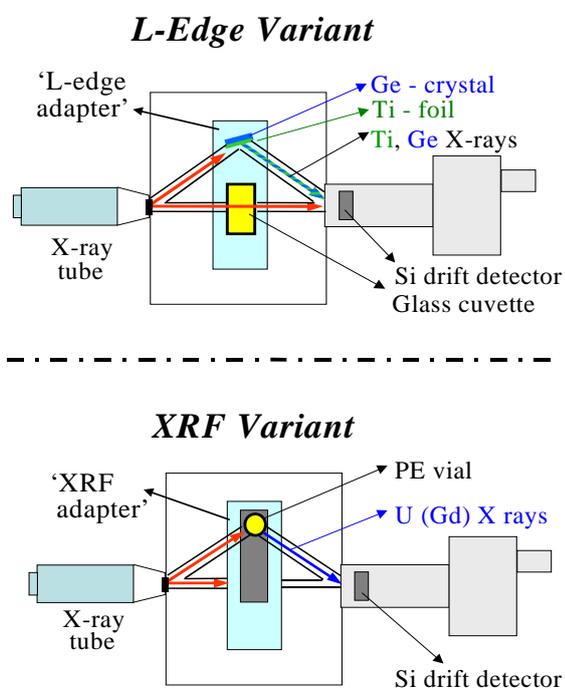


Fig. 2. Irradiation assembly for alternative L-edge densitometry and XRF measurements.

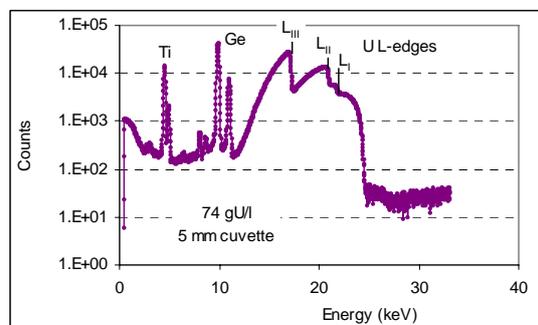


Fig. 3. L-absorption edge spectrum from a uranium solution.

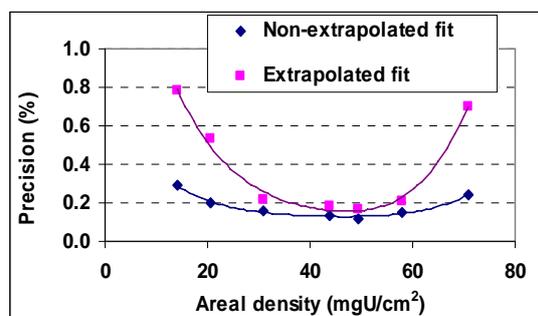


Fig. 4. Measurement precision as a function of uranium areal density (counting time 2000 s).

The path length of the measurement cell depends on the typical uranium concentration level to be measured. The precision curves given in Fig. 4 indicate that optimum measurement precision for the absorption measurement at the L_{III} absorption edge is obtained for an areal uranium density in the range of about 30-60 mgU/cm^2 . On the other hand, the adopted practice in the COMPUCEA measurements is to establish dissolved sample solutions with uranium concentration levels around 200 mgU/ml as primarily needed by the parallel ^{235}U enrichment measurement for improved measurement precision. Given this concentration level, the path length of the cuvette for the L-edge measurement has to be reduced to about 2 mm for optimum precision. Since available disposable cuvettes of this path length are only accurate to about 0.5 %, we have finally chosen the option of using a fixed flowing-through cell with a path length of 2 mm for the L-edge measurements. With the choice of a fixed cell the uncertainty of the actual path length will not matter, and we can maintain in this way the so far adopted practical procedure of using the same sample solution for the parallel concentration and enrichment measurements. Before this modification, during the initial test measurements, disposable 5 mm cuvettes with a path length accuracy of 0.2% were used, requiring a separate sample dilution step to bring the uranium concentration down to about 60-100 mgU/ml as needed for optimum measurement conditions with a 5 mm cell.

The new design also offers the option for separate XRF measurements as indicated in the bottom part of Fig. 2. This option has been added to provide measurement capabilities for the control and quantitative determination of relevant matrix constituents such as gadolinium, which is usually present in some of the uranium samples to be analysed at notable concentrations (up to 10 wt%). A fairly good knowledge of this particular additive appears desirable in order to allow for eventual bias corrections to be made on the L-edge densitometry assay result for uranium. The XRF option has been principally tested, but full performance data and working procedures still need to be established.

2.2 ^{235}U enrichment measurement

The ^{235}U enrichment measurement in COMPUCEA is based on the counting of the 186 keV gammas from ^{235}U from a defined amount of uranium in solution form in a defined counting geometry as illustrated in Fig. 5. The standard detector used so far is a 110 cm³ HPGe well detector with a well diameter of 16 mm in the detector cap. The measurement procedure requires accurate aliquoting of 2.5 ml of the sample solution into a cylindrical plastic vial for counting in the well detector. The accurate amount of uranium in the measurement sample is calculated from the known uranium concentration of the solution sample determined in the parallel K-edge densitometry measurement, from the solution density obtained from an additional density measurement, and from the accurately controlled weight of the sample aliquot. At the typical uranium concentration of 200 mgU/ml this means an amount of approximately 0.5 g of uranium used for the enrichment measurement, which provides a source strength for the 186 keV gammas from ^{235}U of roughly 220 photons per sec and percent enrichment. For the given counting configuration with a measured photo peak detection efficiency of 27 % at 186 keV for a 2.5 ml solution sample this results in a counting rate of 50-60 counts per sec and percent enrichment for the 186 keV line.

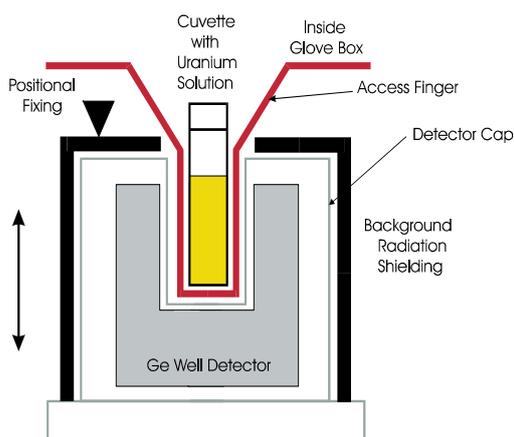


Fig. 5. Principle counting configuration for the enrichment measurement.

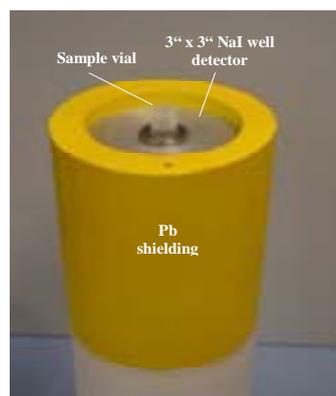


Fig. 6. Shielded NaI well counter with sample vial for enrichment measurement.

The proven principle procedure for the enrichment measurement is also retained in the 2nd generation of COMPUCEA. However, investigations into the possible replacement of the liquid-nitrogen-cooled Ge detector by a room-temperature NaI detector were carried out (Fig. 6). To this end comparative test measurements were performed with a 7.6 x 7.6 cm NaI well detector with a well diameter of 16 mm and a measured resolution of 6.5 % at 662 keV and 9.0% at 186 keV. A measurement example taken from a 4.4% enriched uranium sample is shown in Fig. 7. The given NaI detector provides a more than two times higher detection efficiency (58% for the 2.5 ml sample) compared to the Ge well detector,

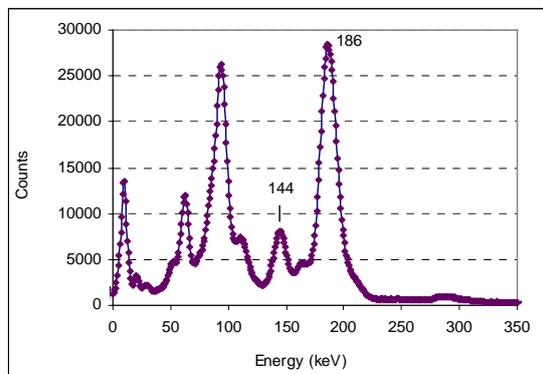


Fig. 7. NaI gamma spectrum from a 4.42% enriched sample.

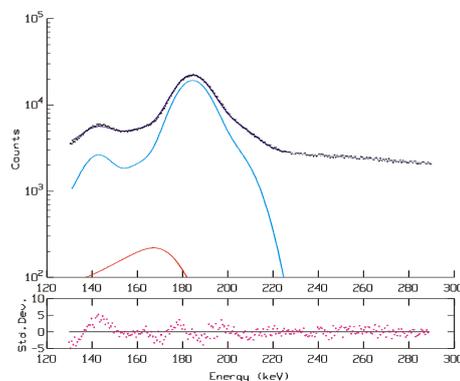


Fig. 8. Example of a response fit with the NaIGEM code (by courtesy of R. Gunnink).

but suffers of course from the much poorer energy resolution. For an accurate determination of the net 186 keV peak area it is proposed to use the method of response function fitting such as provided, for example, by the NaIGEM (NaI Gamma Enrichment Measurements) code, which has proved to provide nearly bias-free net 186 keV peak areas over the full range of enrichment from natural to highly enriched [5]. An example for a spectrum fit with the NaIGEM code is shown in Fig. 8.

3. In-field evaluation of the L-edge densitometer

The new L-edge densitometer for uranium concentration measurements, as well as the NaI well counter for enrichment measurements, were used during some recent PIV campaigns in parallel with the so far applied COMPUCEA equipment for field tests and further performance evaluation.

3.1 L-edge densitometer calibration

Although the COMPUCEA instruments are fielded in a pre-calibrated status, it has become a standard practice to perform an actual calibration on site before each PIV measurement campaign. For this purpose Euratom keeps at each facility a set of well-characterised sintered UO₂ pellets with different enrichment values, ranging from natural to 4.4 % ²³⁵U enrichment. Four of the UO₂ pellets are normally taken to prepare 4 reference solutions with U-concentrations ranging from about 160-220 gU/l, and with enrichments of 0.72, 1.9 and 4.4 wt% ²³⁵U. For the preparation of the calibration solutions the same procedure is strictly followed as for the preparation of the PIV samples, i.e. dissolution of the given amount of sample (typically 5-8 g) into an adjusted volume of 8 molar nitric acid with subsequent dilution with water to establish the desired uranium concentration at a fairly constant acidity level of 3 molar HNO₃.

Fig. 9 displays the calibration results from an in-field calibration of the L-edge densitometer. For an assumed linear instrument response, calibration in this context means the determination of a single calibration factor $k = \Delta\mu (L_{III}) \cdot d$, where $\Delta\mu (L_{III})$ is the change of the total photon absorption cross section (in cm²·g⁻¹) of uranium across its L_{III}-edge, and d the path length of the measurement cell. The evaluation of the uranium concentration from the measured ratio of photon transmission across the L_{III} edge at 17.17 keV follows our proven analysis procedure adopted for K-edge densitometry with an X-ray continuum [6]. In this approach the photon transmission as a function of energy, T(E), is measured

relative to a blank spectrum from a nitric acid solution of representative molarity, and then linearised in a representation $\ln(1/T)$ vs $\ln E$. Linear least-squares fits to the respective data on both sides of the absorption edge determine the photon transmission at energies slightly displaced from the absorption edge ('non-extrapolated fitting mode'), or directly at the absorption edge energy ('extrapolated fitting mode'). Fitting intervals ranging from 15.5-16.7 keV, and from 17.6-18.8 keV were chosen for the evaluation of the transmission ratio across the L_{III} edge.

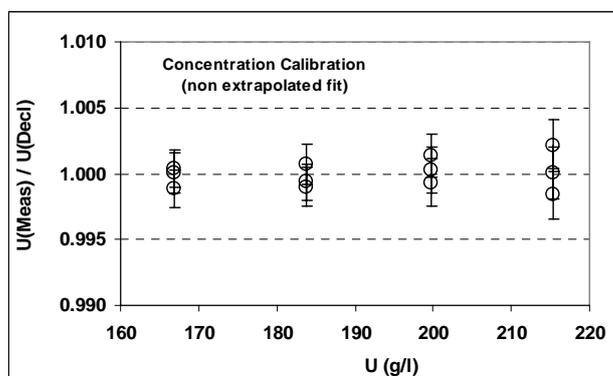


Fig. 9 L-edge calibration data obtained from the non-extrapolated fitting mode.

For the given range of concentration the calibration data suggest perfect linearity, with deviations less than 0.03 % from linearity. The pooled results from a total of 12 calibration runs performed on the 4 calibration samples yielded standard deviations of 0.107 and 0.114 % for the non-extrapolated and extrapolated fitting results.

3.2 Reproducibility

During a PIV campaign one of the calibration samples (with a uranium concentration of 167 gU/l) has been repeatedly measured in the course of the week for a test of the reproducibility of the measurements. From previous extensive studies performed at ITU with a removable 5 mm cell we could already demonstrate a long-term instrument stability of better than 0.1 % [4]. The present investigation therefore rather aimed at an evaluation of the measurement situation encountered with the newly introduced flowing-through cell. The procedure so far adopted for a measurement is to manually press a sample volume of about 2 ml through the flowing-through cell and its connecting piping tubes. The volume of the 2 mm quartz cell itself is 120 μ l. Between two measurements the cell and the connecting tubes are flushed with water and dried with air.

The results from the repeat measurements are plotted in Fig. 10. The standard deviation obtained for the results from 12 repeat measurements is 0.20 %. This value is slightly larger than the statistical measurement precision of 0.15 % for a single run, suggesting an extra average uncertainty component of about 0.15 % associated with the sample handling. This will be further investigated.

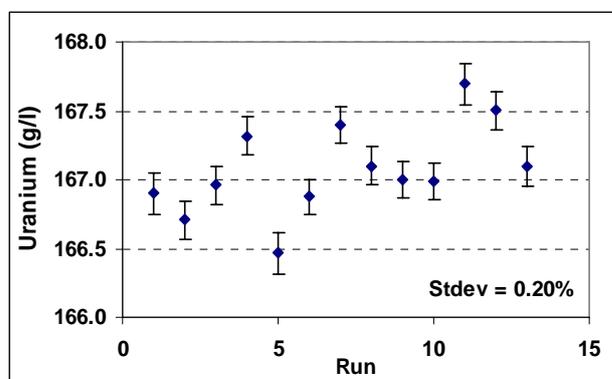


Fig. 10. Reproducibility of repeat measurements made with the flowing-through cell.

3.3 Accuracy

A test for the accuracy of the uranium measurements was made from a set of PIV measurements performed on pellet samples. For this type of verification samples normally reliable declared values for the uranium content are available for comparison as long as the pellets do not contain Gd. The absence of Gd is easily verified from a comparison of the non-extrapolated and extrapolated fitting results for the uranium concentration as illustrated in Fig. 11. The comparison of the two results provides a powerful diagnostic tool. The data also prove that the extrapolated analysis mode is practically insensitive to matrix effects. The indicated percentage bias in Fig. 11 for the non-extrapolated results refers to a uranium reference concentration of 200 gU/l. For comparison the figure also indicates the sensitivity of the previous COMPUCEA measurements made with the $^{153}\text{Gd}/^{57}\text{Co}$ gamma pair of 103/122 keV at the K-absorption edge of uranium.

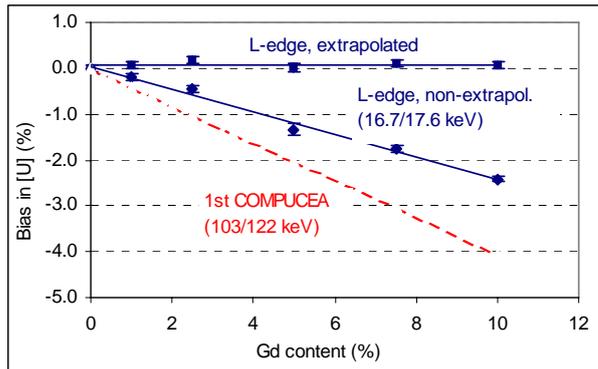


Fig. 11. Impact of the presence of Gd on the uranium assay.

The PIV results for the uranium concentration determination on a set of 10 UO_2 pellets are shown in Fig. 12. All of them were free from Gd as could be verified from a comparison of the results from the non-extrapolated and extrapolated fitting analysis. The average deviation from the declared theoretical value (88.14 wt% of U) obtained for this set of measurements was 0.04% with a standard deviation of 0.14%. The latter is consistent with the typical counting precision of about 0.14-0.16% for an individual measurement.

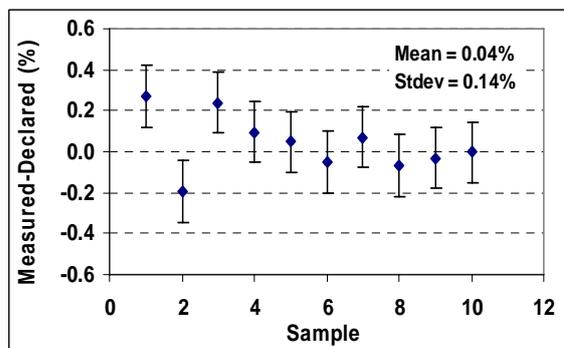


Fig. 12. Uranium assay results for a set of ten PIV samples (UO_2 pellets).

4. In-field evaluation of the NaI well detector for the ^{235}U enrichment determination

The room-temperature NaI well counter, which has been evaluated as an alternative to the HPGe well detector, had identical well dimensions (diameter 16 mm times 40 mm depth). This allows the same vial to be counted in both detectors for comparison measurements. In the case of the NaI detector a thin-walled titanium sleeve (0.5 mm) has been inserted into the well to absorb the uranium L X-rays in order to avoid possible contributions from pulse summing of L X-rays and 186 keV gammas to the to the high-energy side of the 186 keV peak.

4.1 Calibration

The same reference solutions as prepared for the calibration of the L-edge densitometer were also used for the calibration of the enrichment measurement. The purpose of the calibration is (i) to establish the calibration factor converting the measured 186 keV counting rate into the corresponding ^{235}U enrichment, and (ii) to determine the dependence of the measured 186 keV rate on the uranium concentration. This dependence is slightly non-linear because of the gamma self attenuation in the sample (Fig. 13).

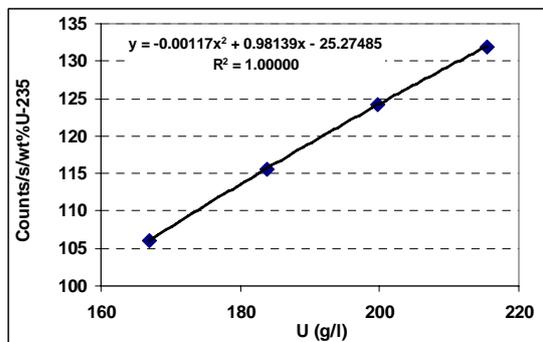


Fig. 13. Dependence of the 186 keV counting rate on the uranium concentration.

4.2 Performance

The performance evaluation for the enrichment determination with the NaI well detector bases so far on measurements during a single PIV campaign, where comparative measurements with the NaI and the routinely used HPGe detector were performed. Fig. 14 shows the results for a set 15 enrichment determinations made on this occasion. The typical measurement precision for the NaI detector measurements as reported by the NaIGEM analysis code is shown in Fig. 15 as a function of the enrichment (normalised to a counting time of 1000 s).

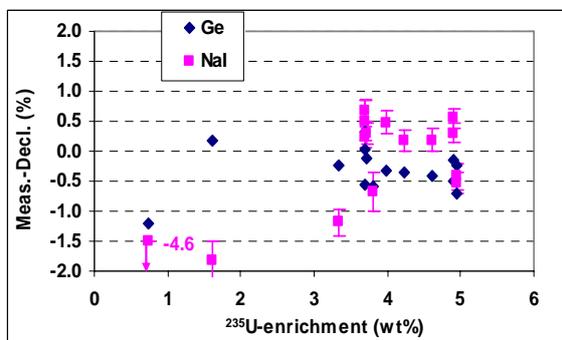


Fig. 14. Percentage difference between measured and declared ^{235}U enrichment.

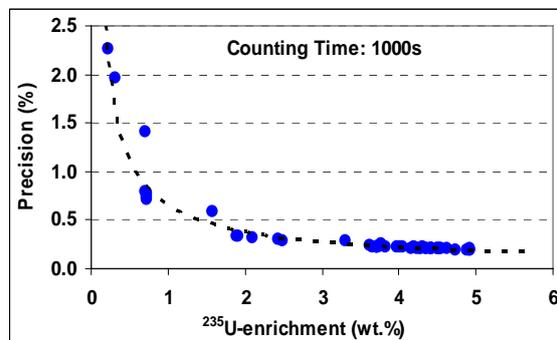


Fig. 15. Measurement precision obtained with the NaI well detector.

The results obtained with the low-resolution NaI well detector are quite encouraging, although the measured enrichment values tend to be increasingly biased with decreasing enrichment, where the peak-to-background ratios for the 186 keV line become poorer and hence the correct assessment of the background more critical. Omitting the data point for the lowest measured enrichment (U-nat), we calculate for the remaining 14 results shown in Fig. 14 the following average differences and standard deviations between measured and declared results:

NaI detector:	-0.10 ± 0.74
HPGe detector:	-0.26 ± 0.30

Some further improvements of the NaI measurement results may be achieved through the application of corrections for temperature effects, which are known to exist in some way but have not been corrected for so far, and through refined spectrum analysis procedures especially for spectra obtained

from very low enrichments below 1%. On the other hand, a completely different alternative worth to pursue is now opening through the availability of new types of scintillator materials like $\text{LaBr}_3(\text{Ce})$, which offer more than 50% better energy resolution compared to NaI together with other attracting features such as a zero temperature coefficient [7].

5. Conclusion

At this stage the new developed 2nd generation of COMPUCEA equipment for the uranium element assay has fully proved its compliance with international target values for measurement performance for this kind of radiometric measurement techniques. The developed L-edge densitometer is easy to operate and has shown a high degree of reliability during the past field applications. Some further tests will be carried out to complete and to conclude the validation procedure in order to enable its accepted routine use during PIV campaigns from 2007 onwards.

For the enrichment measurement part of COMPUCEA, which is well established for the measurements with a HPGe well detector, the alternatively examined use of a room-temperature NaI scintillation detector shows acceptable results, although one has to realize that in terms of measurement performance a low-resolution NaI detector will never become a full substitute for a high-resolution Ge detector. In this case a balance between desirable performance on the one hand and simplicity of equipment on the other hand will have to be made. Nonetheless, further investigations into the measurement performance achievable with the better performing new scintillator materials will remain on the agenda. For the time being the enrichment measurements with the 2nd generation of COMPUCEA will continue with the HPGe well detector as first choice.

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Quantification of Cerenkov light for safeguard applications

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Abstract. The International Atomic Energy Agency (IAEA) safeguard inspectors currently exploit the Cerenkov light to perform attribute verifications of spent nuclear fuel assemblies stored under water. For these verifications, dedicated non-destructive assay equipment can be used to visualise and record images of the UV light produced by the radiation originating from the spent fuel assembly. It is advanced that the qualitative information obtained from these observations could be used to distinguish between different fuel irradiation and cooling history, thus providing appropriate tools for reaching further conclusions on the measured spent fuel. Consequently, the IAEA investigated the quantification of the UV light source term for a variety of irradiation and cooling histories of UO₂ fuel using ORIGEN-ARP software. Additionally, the present study looked at the similarity or potential difference between fuel and dummy rods in terms of UV photon production. Results of this investigation are presented.

1. Introduction

The IAEA safeguard inspections verify nuclear materials in the member state civilian nuclear fuel cycle installations to check their compliance within the framework of the non-proliferation treaty and/or other nuclear safeguards agreements. These verifications include the control on nuclear spent fuel stored in cooling ponds. For these assemblies stored under water observation of the light produced by Cerenkov Effect [1] represents a very convenient mean to establish the presence or absence of nuclear material (Gross Defect verification). Two Non-Destructive Assay (NDA) devices are routinely used in inspections to image the Ultraviolet (UV) Cerenkov light. Within the IAEA these are referred as the Improved Cerenkov Viewing Device (ICVD) [1] and the Digital Cerenkov Viewing Device (DCVD) [1]. Approximately 70% of spent fuel verifications are performed on the basis of the UV Cerenkov glow observation.

Over the last 5 years, improvement to the Cerenkov viewing devices has positively impacted the method in which inspectors utilize the Cerenkov Effect in safeguards applications. The DCVD records a digital image (512x512 pixels) of the spent fuel and allows the relative quantification of the light intensity at the pin level. This quantification could provide reliable information in evaluating possible diversion scenarios, based on differences in UV source glow. The discussions related to these diversion scenarios were the driving factors to the IAEA effort for quantification of the source term of UV photons emitted by spent fuel. The IAEA investigated, within suitable boundaries, the quantification of the source term definition for a variety of irradiation and cooling histories using ORIGEN-ARP software. In addition to UO₂ fuel in Zircalloy-2 cladding, dummy rods of Depleted Uranium, Natural Oxygen and Lead were also investigated.

2. Cerenkov Light Production

Cerenkov light originates from the charged particles traveling through a transparent medium faster than the speed of light. This induces UV photon production due to the Cerenkov effect [2]. In the case of spent nuclear fuel placed underwater, secondary electrons originating from gamma ray Compton scattering are probably the main contributor to the Cerenkov light intensity. Two additional sources of charged particles could potentially be investigated in terms of their relative contribution to the Cerenkov light intensity. Firstly (n, γ) reactions in the water for which secondary gammas could also induce charged particle by Compton scattering and secondly beta particles managing to escape the fuel pin and its cladding with an energy larger than the threshold required to emit light by Cerenkov Effect. For water having a refraction index of $n=1.3$, this correspond to an electron travelling faster than $230609615 \text{ m.s}^{-1}$, thus having an energy greater than 0.2887 MeV . Table 1 lists the primary photon emitters and their respective half-lives, gamma energies and recoil electrons

Table 1. Primary Photon Emitters.

Element	Isotope	Half-life of parent	Major Gamma Energies (MeV)	Recoil energy of electron (MeV)
Barium/Lanthanum	$^{140}\text{Ba}/^{140}\text{La}$	13d	1.60	1.21
			2.52	2.10
Zirconium/Niobium	$^{95}\text{Zr}/^{95}\text{Nb}$	64d	0.72	0.42
			0.76	0.45
Cerium/Praseodymium	$^{144}\text{Ce}/^{144}\text{Pr}$	284d	0.70	0.40
			1.49	1.11
			2.19	1.78
Ruthenium/Rhodium	$^{106}\text{Ru}/^{106}\text{Rh}$	373d	0.51	0.25
			0.62	0.34
			1.05	0.71
Cesium/Barium	$^{137}\text{Cs}/^{137\text{m}}\text{Ba}$	30.2a	0.66	0.37

3. Previous Cerenkov Light Quantification

From the previous paragraph, one must infer that the intensity of UV photon production is directly related to the inventory of gamma ray emitting radioisotopes present in the spent fuel. Initial study like the one of Dowdy at al. [3] only included the contribution from 6 fission products as shown in Table 1. Additionally, Cerenkov light originating from spent fuel is generated over wavelength covering both the UV and visible region (180 – 600 nm) with a maximum around 480 nm. This gives it its characteristic “Blue” Glow. Other Cerenkov light source term quantification estimates like Chen et al [2] and Rolandson [10] have been based on results obtained from depletion computer code for nuclear spent fuel. Depending on the version of the code used increased complexity in the number of transuranic fission and number of fission product being considered. These allowed for a given reactor and fuel type to generate depending of assumptions on the irradiation intensity and duration to produce an exhaustive inventory of most radionuclide formed in the nuclear spent fuel. Of course they also account for the decay of these nuclides during irradiation, down time and cooling down periods. Table 2 outlines previous publications and their respective methodologies used. The beta emission was also neglected in our calculations but it is envisaged to include it in further work.

Table 2. Previous Methodologies.

<p>1979, Dowdy, et al. [3]</p> <ul style="list-style-type: none"> • Only thermal fissions of ^{235}U are included in calculations of fission products • 6 photon emitters considered • Only Cerenkov photons in the range of 400-600 nm is of interest • Neglected electron emission 	<p>1989, Chen, et al. [2]</p> <ul style="list-style-type: none"> • Gamma activity of spent fuel of CANDU fuel calculated using LATREP AND CANIGEN • Only direct fission products from both ^{235}U and ^{239}Pu included • Summed photon emitters and translated to UV photon production rate • Neglected electron emission 	<p>1994, Stig Rolandson, [10]</p> <ul style="list-style-type: none"> • Used ORIGEN2.2 to for 0.45MeV-3.0MeV bins to find 6 primary photon emitters • Summed primary photon emitters and translated to UV photon production rate • Neglected electron emission
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4. IAEA Cerenkov Light Evaluation

4.1. Proposed Methodology

In the present work the ORIGEN-ARP software was used to generate the inventory of gamma ray emitters and the corresponding energy spectrum. This information once splitted over a number of energy bins was used to derive the mean number of UV photons produced.

4.2. Software for isotopic inventory calculations

ORIGEN-ARP is a software package developed by Oak Ridge National Laboratory to perform spent fuel characterization, to calculate isotopic inventory, radiation source terms, and decay heat of specific spent fuel assemblies after irradiation and decay. The version utilized was release 2.00, copyright 2002. Users must input descriptive values related to the initial spent fuel composition, the fuel type, irradiation and decay times and neutron and gamma energy groups. Listed in Table 3 are the input parameters for the initial fuel composition used in this investigation. Calculations were performed with an 8x8 BWR library with a moderator density of 0.4323g/cc.

Table 3. Input Parameters for ORIGEN-ARP Calculations.

Enrichment U^{235} (wt%)	2	
Basis Unit	1 MTU	
Composition	UO₂ Fuel	Mass (grams)
UO ₂ Fuel, 2% enriched	<i>Uranium</i>	10^6
	U^{235}	20000
	U^{238}	980000
	O^{16}	138952.16
	Cladding	
	Zircaloy	50835.52
	Nb	513.49

4.3. Calculation of UV Photons from ORIGEN-ARP Output

ORIGEN-ARP is capable of producing an output indicating the gamma spectrum, based upon the user's selected energy bins in the units of MeV/sec as well as photons/sec. For this application, the energy bin structure provided in Table 4 was utilized, and a new structure was added to the ORIGEN-ARP photon spectrum library. The Maximum value of 3.0 MeV used in energy bin structure corresponds to the maximum energy at which photons undergo Compton scattering within a water medium. Using the aforementioned energy bin structure, an estimated UV Photon

production rate was calculated for each of the energy bins as indicated in Table 4. The ORIGEN-ARP photon spectrum in the units of MeV/sec was then multiplied and summed to tally the expected UV photon production rate.

Table 4. Selected Energy Bins for ORIGEN-ARP Software.

Lower Bound	Upper Bound	Mean Energy	Photon Production per Energy Bin
0.512	0.60	0.56	0.29
0.60	0.70	0.65	4
0.70	0.80	0.75	10
0.80	1.00	0.90	22
1.00	1.20	1.10	41
1.20	1.33	1.27	60
1.33	1.44	1.39	75
1.44	1.50	1.47	86
1.50	1.57	1.54	96
1.57	1.66	1.62	108
1.66	1.80	1.73	124
1.80	2.00	1.90	151
2.00	2.15	2.08	181
2.15	2.35	2.25	210
2.35	2.50	2.43	241
2.50	3.00	2.75	296

4.4. UV Source Term Results

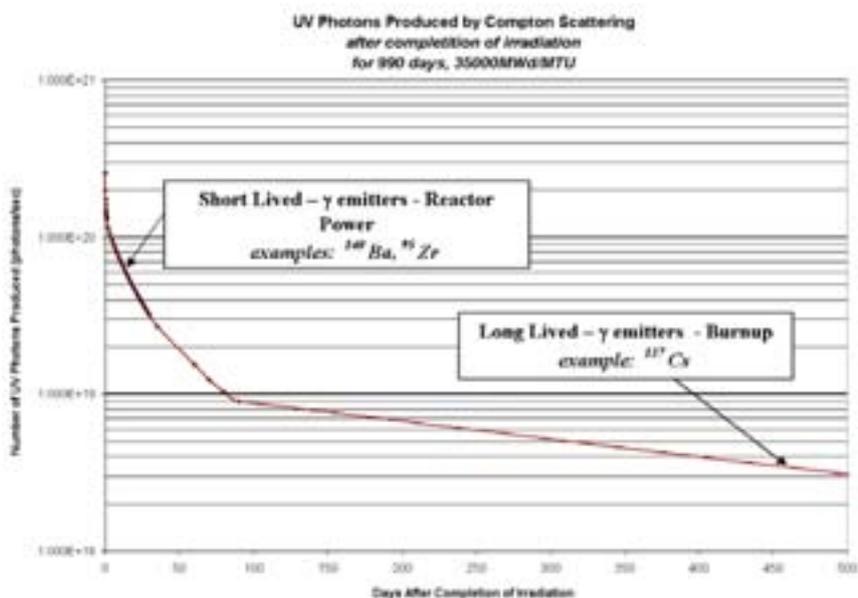
The general trend observed in the UV photon production rate can be summarized as follows:

- an initial high intensity term following the decay of short lived gamma rays;
- a long term component following the concentration of long lived gamma emitters.

The initial term intensity is in direct relationship with the power level of the reactor in the last few weeks prior to fuel discharge.

The second term is in direct relation with the achieved burnup of the fuel throughout its irradiation period. The results obtained within this exercise follow the behaviour of previous calculations in literature and are thus comparable. However, the UV photon production rate in the present study was calculated with specific emphasis for shorter cooling times and long cooling times (greater than 20 years). Figure 1 illustrates the two different areas of the graph and highlights the isotopes that are primarily responsible for the Cerenkov Light during that time interval.

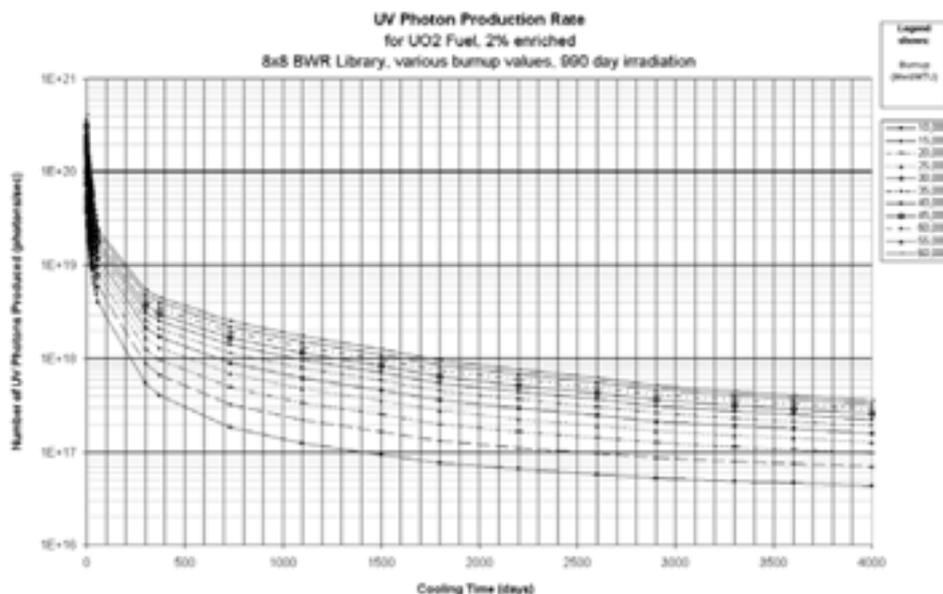
Figure 1. UV Photon Production Rate for 990-days, 35000 MWd/MTU.



4.5. Impact of Burn up on Photon Production Rate

As the burnup of the fuel increases, so does the respective UV photon production rate, as seen in Figure 2. Although, it is observed at shorter cooling times, the disparity between the different burnup values are less pronounced than at shorter cooling times, again reiterating that at shorter cooling times, the UV photon production rate is primarily indicative of the reactor power and not necessarily the burnup of the fuel.

Figure 2. UV Photon Production Rate for a variety of burnup values from 0 to 4000 days.



4.6. Application of Results to Safeguards Scenarios

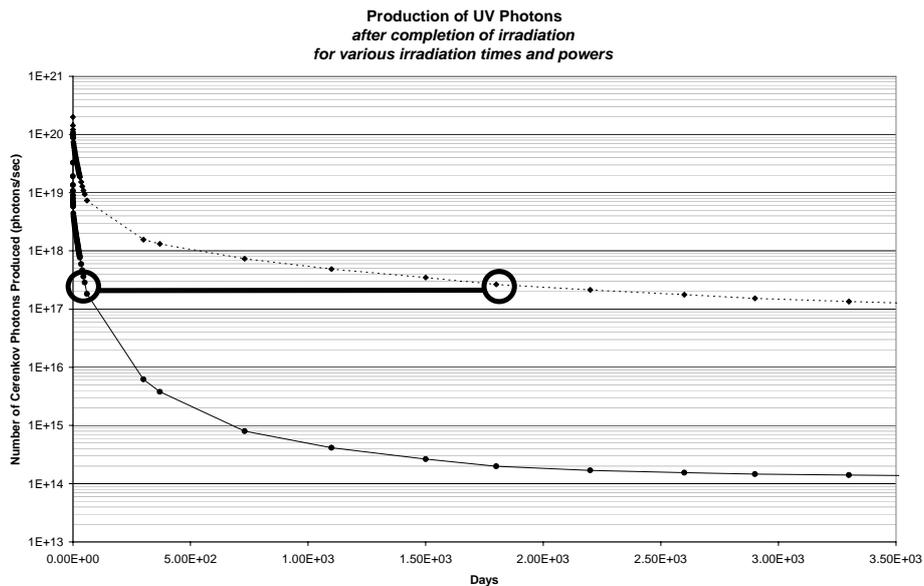
Using the calculated information, two different spent fuel scenarios were addressed and the results compared. In this sense, the applicability of Cerenkov light in evaluating different diversion

scenarios is tested, and discussed. In the first scenario, a surveillance system failed on a spent fuel pond, rendering a time span of thirty days without information on the entirety of the spent fuel in the pond. In a different scenario, a surveillance system failed on a portion of a spent fuel pond for seven days. An assembly must be verified to have the same irradiation and cooling history as the other fuel bundles. The burn up of this fuel bundle is known.

Verification of Irradiation Time

When trying to analyze the UV photon rate of extremely short cooled fuels without information related to burn-up of the fuel, information about the cooling and irradiation history cannot be contrived from a single viewing of the Cerenkov light. Previous calculations of UV photon source terms have found that there are three parameters that contribute to the term including irradiation time, power level and decay time. With these three parameters, there is a possibility to observe the same absolute UV photon intensity for fuel elements that have very different irradiation/cooling time histories, as highlighted in Figure 3, showing a comparison between UV source term intensity from fuel that was irradiated for 7 days at 10 MW and cooled for 60 days versus fuel irradiated for 730 days at 55 MW and cooled for 2220 days. As the graph illustrates, the rate of change of the UV photon production rate can be indicative of whether the fuel is recently discharged or not.

Figure 3. Production of UV Photons for UO₂ fuel irradiated for 7 days at 10 MW and 730 days at 55 MW.



Verification of Cooling Time

When observing fuel assemblies with identical burnup, the approximate cooling time of the assemblies can be compared against one another. As shown in Table 5, for a burnup value of 35000 MWd/MTU, the Cerenkov Light at 66 days of cooling time is approximately 15 times brighter than would be seen at 980 days cooling time. This feature allows to compare light intensities of assemblies of similar burnup and infer their relative cooling time difference

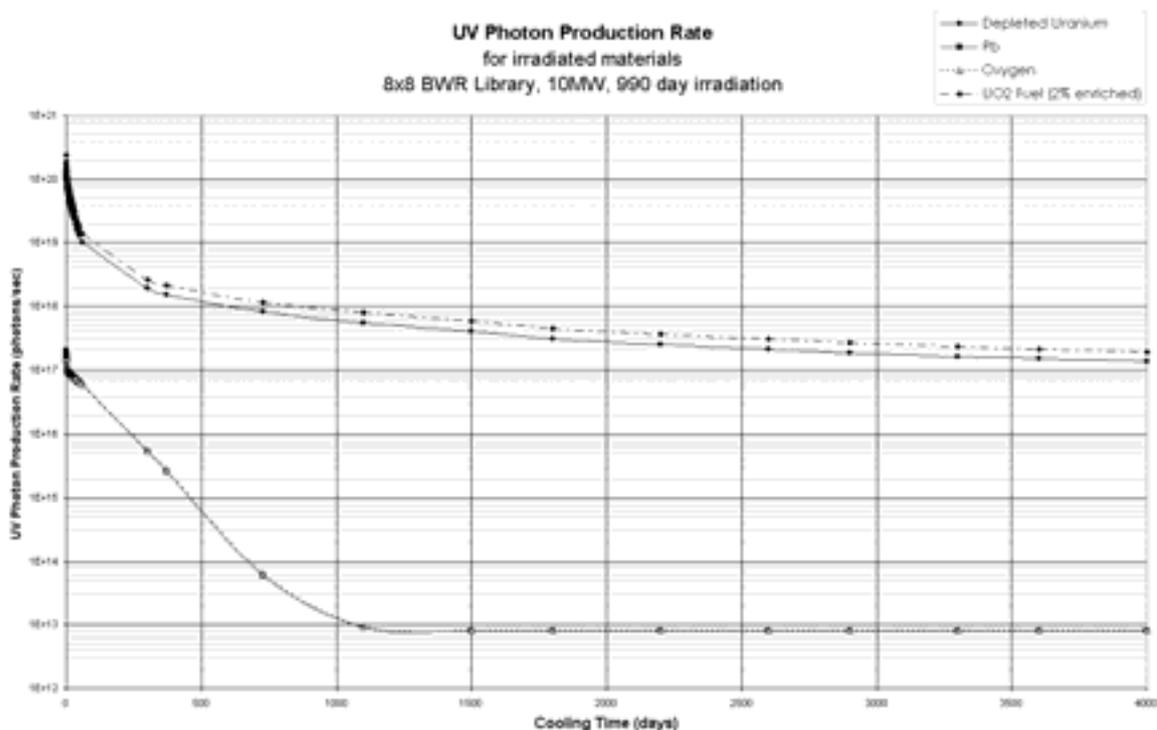
Table 5. Cerenkov Light intensity for various irradiation scenarios.

Days Irradiated	Burnup (MWd/t)	Cooling Time (days)						
		66	270	610	980	1300	1700	3500
990	10000	3.55E+18	6.57E+17	2.42E+17	1.38E+17	9.16E+16	6.76E+16	3.39E+16
990	15000	5.31E+18	1.04E+18	4.12E+17	2.42E+17	1.62E+17	1.2E+17	5.61E+16
990	20000	7.09E+18	1.47E+18	6.13E+17	3.67E+17	2.48E+17	1.83E+17	8.13E+16
990	25000	8.9E+18	1.93E+18	8.43E+17	5.11E+17	3.48E+17	2.56E+17	1.09E+17
990	30000	1.08E+19	2.42E+18	1.1E+18	6.72E+17	4.59E+17	3.38E+17	1.39E+17
990	35000	1.27E+19	2.95E+18	1.37E+18	8.46E+17	5.79E+17	4.25E+17	1.69E+17
990	40000	1.46E+19	3.49E+18	1.66E+18	1.03E+18	7.05E+17	5.17E+17	2.01E+17
990	45000	1.66E+19	4.06E+18	1.96E+18	1.22E+18	8.36E+17	6.12E+17	2.32E+17
990	50000	1.86E+19	4.64E+18	2.27E+18	1.42E+18	9.71E+17	7.1E+17	2.64E+17
990	55000	2.06E+19	5.22E+18	2.58E+18	1.62E+18	1.11E+18	8.09E+17	2.96E+17
990	60000	2.27E+19	5.82E+18	2.9E+18	1.82E+18	1.25E+18	9.09E+17	3.28E+17

Dummy Rod Calculations

In addition to calculating the UV photon production rate of spent fuel (UO₂), the UV photon production rate was also calculated for dummy spent fuel assemblies with UO₂ replaced by Lead, Oxygen (hollow) and Depleted Uranium, encapsulated in a Zircaloy-2 Cladding. Results are given in Figure 4 and show that immediately after irradiation, the UV photon rate is approximately the same for UO₂ fuel irradiated at the same power level of 35 MW for a 990-day continuous irradiation and cooled for 4000 days and as for an Oxygen or Lead filled dummy rod irradiated at the same power level and irradiation time and with a 10 day cooling time. However, in contrast to Lead and Oxygen, the UV photon production rate for UO₂ fuel enriched at 2% and Depleted Uranium is difficult to distinguish throughout the cooling time period.

Figure 4. UV Photon Production Rate for various fuel types irradiated at 10 MW for 990 days.



5. Conclusion

Cerenkov glow from spent fuel stored under water is commonly used as an attribute tester. The present investigations aimed at establishing a methodology for quantifying the UV photon intensity originating from the Cerenkov Light. Additionally this information was used to investigate the possibility of differentiating irradiated fuels with different irradiation and cooling time histories or “dummy” fuel elements.

The ORIGEN-ARP software was used to generate the isotope inventory according a specific irradiation and cooling pattern, and the UV photon intensity was calculated by considering only energetic gamma rays capable of producing secondary energetic electrons by Compton effect. The UV photon production rate was calculated for a variety of irradiation and cooling times, specifically for very short cooling times and very long cooling times. Two specific scenarios were explored and the possibility of using Cerenkov Light to identify assemblies with different characteristics (burn-up, cooling time...) was investigated. This study indicated that, by successive measurements of the Cerenkov Light, information about the irradiation and cooling history could be gained. It therefore confirmed the possibility to infer whether the fuel has been directly removed from the reactor or cooled for a long period by such measurements. Cerenkov Light intensity comparison was also shown to provide the mean to perform cooling time evaluation for assemblies of equivalent burn-up.

Calculations performed for dummy rods of oxygen (hollow) and lead were shown to exhibit Cerenkov Light emissions that can easily be identified as originating from short-lived nuclides. Similar calculations for a DU assembly showed very similar light intensity as UO₂ fuel at 2% enrichment and therefore may not be detectable by simple DCVD observation. The results from this investigation will influence the methods in which the ICVD and DCVD are used during inspection.

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Safeguards and environmental measurements using Compton suppressed Ge detectors

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Abstract: The Compton suppressed Ge detector has been a powerful tool in basic nuclear science research. The suppression mechanism helps one “see” low-intensity peaks that are otherwise hidden under the Compton continuum. It also improves statistical uncertainty because of lower background under the gamma-ray peak. We are using this method to significantly advance the currently available safeguards and environmental sample analysis techniques that use gamma-ray measurements. Our preliminary studies using a not completely optimized system show both positive and negative results in using the Compton suppression. More careful setup (in both electronics and geometry) and study (using broad range of assays) are needed.

Compton suppression system

Recently we have acquired a Compton suppressed system shown in Photo 1 that consists of a 25% safeguards type HPGE detector from ORTEC (energy resolution: 850eV@122keV) and a BaF₂ shield from SCIONIX. Conventional NIM electronics were used for fast-timing signal processing. A CANBERRA ACCUSPEC card was used for data acquisition. In the future, we will be using the ORTEC DSPEC+ for Ge signals and the NIM electronics to generate Compton suppression gate signals to the gate input of the DSPEC+ to obtain full advantage of digital signal processing. We are using plutonium standards PIDIE#1, CRM-137 and uranium standards CRM-U005,U010,U500 and U750 for intercomparison studies. The MGA++ suite was used for data analysis.

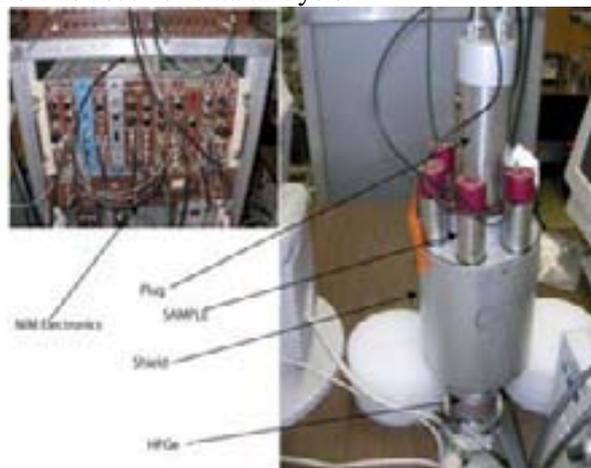


Photo 1. The Compton suppression system, the plug is used to suppress the backscattered photons (i.e. high energy electrons around the Compton edge).

MGA++

The gamma-ray multi-group analysis suite MGA++^{1,2,3} developed at the Lawrence Livermore National Laboratory is widely used for non-destructive plutonium and uranium gamma-ray assay for isotopic information. This plutonium/uranium isotopic analysis suite is unique in that it de-convolutes the complicated, 100-keV x ray and gamma-ray region to obtain the ratios of the Pu/U isotopes. As a result, for example, MGA++ can determine the relative abundance of the plutonium isotopes with accuracy better than 1% using a high-resolution planar germanium detector in a few minutes of counting time. For a confirmatory measurement MGA++ can provide plutonium isotopic analysis within a few seconds of counting time.

DATA with and without COMPTON SUPPRESSION

Study 1: Unknown sample

Figure 1 shows a comparison of the gamma-ray data from a plutonium solution sample that was taken with the same detector with and without the Compton suppression (CS) mechanism. The solution contains about 300 ng of Pu and was counted for 24 hours. The Compton continuum has been reduced significantly. Table 1 shows the area error of the selected peaks. The results clearly show that reduction in the background that improves the peak-area errors. The lack of Compton suppression below 60-keV is due to the geometry of the suppression shield.

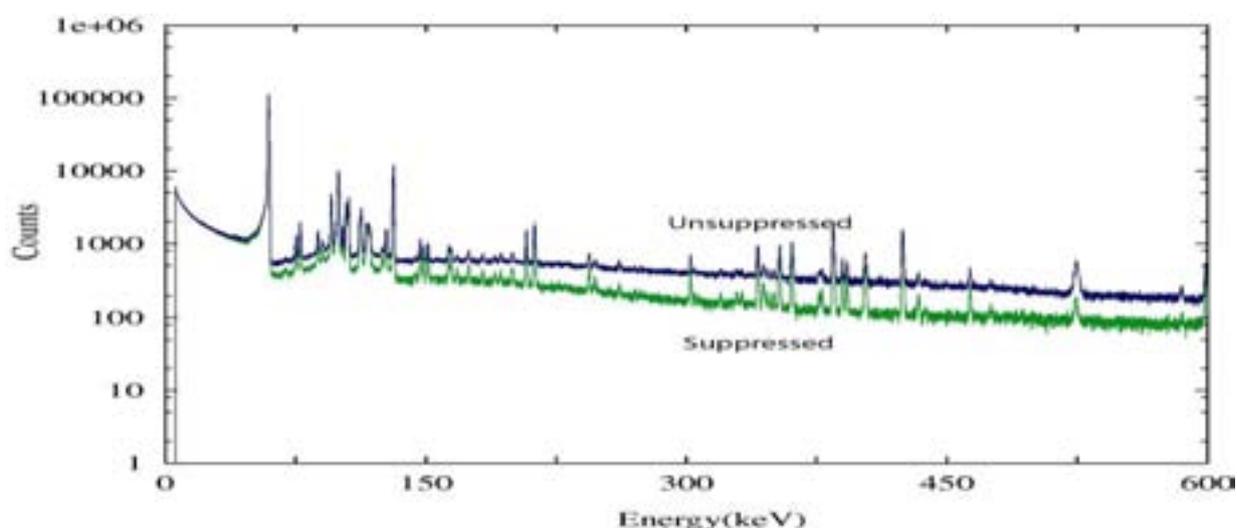


Figure 1. Comparison of the gamma-ray data with and without Compton suppression.

Table 1. Comparison in peak area error.

Energy (keV)	Nuclide	Peak Area Error (%) With CS	Peak Area Error (%) Without CS
129	Pu-239	0.5	0.5
203	Pu-239	1.9	2.1
208	Pu-241	1.6	1.8
375	Pu-239	1.3	1.9
414	Pu-239	1.3	1.5

Both data sets were analyzed using MGA++. The results are shown below. The CS data show a slight degradation in the energy resolution, the ^{238}Pu has smaller % error for the CS data, however, the CS data have a significantly larger ^{241}Pu % error.

MGA Results:

(Compton Suppressed) CS

Pu g/cm2 = .0020 CD g/cm2 = .917 FWHM at 122 keV = 856 eV
 QFIT = 1.03 at 208 keV = 949 eV
 NQFIT= 1.002

	WT.PCT.	%ERR
Pu238 =	.00697	9.39
Pu239 =	89.33430	.10
Pu240 =	10.59105	.83
Pu241 =	.06768	3.90
Pu242 =	.0000	(10)
Am241 =	.19213	.94
U235 =	1.8803	17.34

(No Compton Suppressed) NCS

PU g/cm2 = .0020 CD g/cm2 = .945 FWHM at 122 keV = 848 eV
 QFIT = 1.00 at 208 keV = 926 eV
 NQFIT= 1.000

	WT.PCT.	%ERR
Pu238 =	.00488	13.10
Pu239 =	89.68882	.09
Pu240 =	10.24325	.79
Pu241 =	.06307	1.72
Pu242 =	.0000	(10)
Am241 =	.18465	.89
U235 =	2.2047	18.14

Study 2: Pidie#1

We have performed measurements using PIDIE#1 plutonium standard, a Cd absorber was placed between source and the detector to reduce the rate of the 59-keV gamma-rays, however, there was no Cd absorber between the source and the shield . The deadtime is about 15%, and the PIDIE source is about 3cm to the front of the HPGe detector. The MGA analysis results for the CS and NCS data are shown below with declared date on January, 1, 1988.

MGA Results:

CS

PU g/cm2 = 1.3278 CD g/cm2 = 1.821 FWHM at 122 keV = 903 eV

at 208 keV = 983 eV

ISOTOPIIC ANALYSIS AT

	MEAS. DATE		DECLARED DATE		SPECIFIC POWER (MILLIWATTS/GM)
	WT.PCT.	%ERR	WT.PCT.	%ERR	
Pu238 =	.01083	13.31	.01256	11.44	.07128
Pu239 =	93.90833	.12	93.78310	.12	1.80889
Pu240 =	5.97482	1.90	5.97559	1.90	.42321
Pu241 =	.08177	1.49	.20456	.78	.00698
Pu242 =	.0242	(10)	.0242	(10)	.00003
Am241 =	.34248	1.05	.22757	1.64	.25989
TOTAL = 2.570 +/- .50%					

NCS

PU g/cm2 = 1.1928 CD g/cm2 = 1.791 FWHM at 122 keV = 918 eV
 at 208 keV = 997 eV

ISOTOPIIC ANALYSIS AT

	MEAS. DATE		DECLARED DATE		SPECIFIC POWER (MILLIWATTS/GM)
	WT.PCT.	%ERR	WT.PCT.	%ERR	
Pu238 =	.01287	11.37	.01493	9.77	.08474
Pu239 =	93.67731	.12	93.55097	.12	1.80441
Pu240 =	6.20212	1.80	6.20281	1.79	.43931
Pu241 =	.08234	1.54	.20598	.79	.00703
Pu242 =	.0253	(10)	.0253	(10)	.00003
Am241 =	.34985	1.01	.23428	1.56	.26755
TOTAL = 2.603 +/- .50%					

It is interesting to note that for the PIDIE#1 measurement, the CS results show better energy resolution and better agreement with the declared Pu isotopics (%):

Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Am-241
0.01	93.76	5.99	0.20	0.03	0.23

Study 3: CRM-137

Because of building limits on nuclear materials, we are not allowed to bring other PIDIEs into the room where the CS system resides. We used liquid form of about 200 ng of CRM-137 standards for another plutonium isotopic study. The ²³⁹Pu is about 77.5% in the CRM-137, a 60 mil Cd absorber was placed between the source and the detector. Data were collected for 24 hours. The results are shown in the following table:

	Pu-238 (%)	Pu-239 (%)	Pu-240 (%)	Pu-241 (%)	Pu-242 (%)
Declared	0.268	77.549	18.792	2.168	1.225
CS	0.253	78.069	18.795	2.096	0.785
NCS	0.294	77.523	19.068	2.257	0.856

Study 4: CRM-U005,U010,U500,U750

The MGA++ results for these standards are summarized in the following table. The results are mixed: For low enrichments standards, the CS data is not as accurate as the NCS data, however, the CS results for the 50% ²³⁵U, U500 is better. The huge discrepancy for both CS and NCS results for U750(75% ²³⁵U) standard is mostly due to the inadequate resolution of the HPGe detector (850eV @ 122 keV,

MGA++ requires 550eV@122 keV) and partially due to the algorithms used for intrinsic efficiency determination in the MGA++ code.

	²³⁸ U	²³⁵ U	²³⁴ U
U005 declared	99.5 +- 0.0003	0.500+-0.0003	0.00334+-0.00007
U005 CS	99.3 +- 0.21	0.664 +- 0.21	0.004 +- 0.002
U005 NCS	99.35 +- 0.21	0.640 +- 0.21	0.003 +- 0.002
U010 declared	98.991+- 0.001	1.0038 +- 0.001	0.0054+- 0.00005
U010 CS	98.903 +- 0.024	1.092 +- 0.024	0.005 +- 0.002
U010 NCS	98.926 +- 0.023	1.062 +- 0.023	0.012 +- 0.002
U500 declared	50.06 +- 0.05	49.42 +- 0.05	0.511 +- 0.0008
U500 CS	48.249 +- 0.505	51.146 +- 0.505	0.604 +-0.061
U500 NCS	47.396 +- 0.513	51.984 +- 0.513	0.621 +- 0.063
U750 declared	23.86+- 0.024	75.54 +- 0.025	0.5938 +- 0.0009
U750 CS	15.787 +- 1.104	83.467 +- 1.104	0.747 +- 0.075
U750 NCS	16.182 +- 1.95	83.044 +- 1.095	0.774 +- .078

Future Studies:

We will use smaller, higher resolution HPGe detectors in conjunction with the CS shield for safeguards accountability studies. For example, assay of fresh separated Pu and U samples where some weak peaks can be identified quicker and more accurately using the CS method. We will also explore the cases where the current CS system may be useful such as ^{232,233}U mixtures, shielded U/Pu samples, and fissile material in environmental samples.

Conclusions:

Our preliminary studies show that an optimized CS system could be a powerful tool for non-destructive gamma-ray spectrometry. For example, in one of our measurement using the ^{233,232}U mixture, we have observed significant error reduction in the CS results. We are currently modifying the MGA++ suite for such ^{232,233}U isotopic analysis. We have also identified areas (both hardware and MGA++) that need improvement to realize the goal of improved gamma-ray spectrometry for safeguards and environmental sample analysis.

ACKNOWLEDGEMENTS\

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Superconducting ultra-high energy resolution Gamma spectrometers for nuclear safeguards applications

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Abstract. Superconducting Gamma-ray spectrometers offer an order of magnitude improvement in energy resolution over conventional germanium or scintillator detectors, and can improve the precision of non-destructive isotope analysis accordingly. We are developing spectrometers based on bulk tin absorbers and molybdenum-copper multilayer sensors operated at the transition between their superconducting and normal state. They are cooled to their operating temperature of ~ 0.1 K in an adiabatic demagnetization refrigerator, and have achieved an energy resolution between 50 and 90 eV FWHM at 100 keV. We discuss the performance of the instruments in the context of nuclear safeguards applications.

1. Introduction

Gamma (γ) spectrometry is widely used to determine the isotopic composition of radioactive materials [1]. Line intensities of characteristic γ -rays can provide information about isotope ratios to infer sample composition, age, and processing history. High-precision measurements tend to be based on lines with similar energies to reduce systematic errors from variations in self-absorption and detection efficiency. Germanium detectors are typically used, since they combine high energy resolution to separate the emission from different isotopes with high absorption efficiency required to measure weak emission lines from dilute samples. They can determine isotope ratios with an error of 1% or better, despite the fact that some of the emission lines can be affected by spectral interferences. For improved precision, spectrometers with better energy resolution would therefore be desirable in nuclear safeguards and verification applications that rely on measuring small differences in isotopic composition or on verifying isotopic composition with low uncertainty.

Cryogenic γ -ray spectrometers operating at temperatures of $T \approx 0.1$ K offer an order of magnitude improvement in energy resolution over conventional Ge detectors [2]. They typically consist of an absorber attached to a sensitive thermometer, both weakly thermally linked to a cold bath. A γ -ray with energy E_γ will increase the temperature of an absorber with heat capacity C by an amount E_γ/C proportional to the γ -ray energy, which can be measured with the attached thermometer before both absorber and thermometer cool back down to the bath temperature through the weak thermal link. The energy resolution ΔE_{FWHM} of cryogenic spectrometers is fundamentally limited by thermodynamic fluctuations to

$$\Delta E_{\text{FWHM}} \approx 2.355 \sqrt{k_B T^2 C} \quad (1)$$

and can be well below 100 eV FWHM for operation at $T \approx 0.1$ K for pixel volumes of order 1 mm^3 and thus small heat capacities C [3].

The Advanced Detector Group at Lawrence Livermore National Laboratory is developing γ -ray detectors based on bulk superconducting Sn absorbers coupled to sensitive Mo/Cu superconducting-to-normal transition edge sensors (TESs) [4]-[6]. We are also developing refrigeration and readout technology for user-friendly detector operation at ~ 0.1 K. These detectors have achieved an energy resolution between ~ 50 and 90 eV FWHM for energies below 100 keV, and are thus ideally suited for precise non-destructive analysis of nuclear samples. We discuss the spectrometer design, compare their performance with conventional Ge detectors for uranium analysis, and discuss their potential use in nuclear safeguards.

2. Technology

The spectrometers are based on measuring the increase in temperature upon photon or particle absorption with a sensor operated at the steep resistive transition between its superconducting and its normal state, typically referred to as transition-edge sensors (TES) or superconducting phase transition (SPT) detectors. Our sensors consist of a multilayer of superconducting molybdenum and non-superconducting copper, where the thickness ratio of the two materials is chosen to produce a multilayer with a desired intermediate critical temperature of ~ 0.1 K. The sensors are photolithographically patterned on top of a freestanding $0.5 \mu\text{m}$ silicon nitride membrane that provides the weak thermal link to the silicon substrate with thermal conductance G_{TES} . For increased absorption efficiency, a bulk absorber is attached to the Mo/Cu sensor, whose size (and thus heat capacity C_{absorber}) is chosen by a tradeoff between detection efficiency and desired energy resolution (equation 1). For operation in the ~ 100 keV energy range with an energy resolution below 100 eV FWHM, we typically use 0.25 mm thick tin absorbers with an area between $1 \text{ mm} \times 1 \text{ mm}$ and $2 \text{ mm} \times 2 \text{ mm}$ (figure 1).

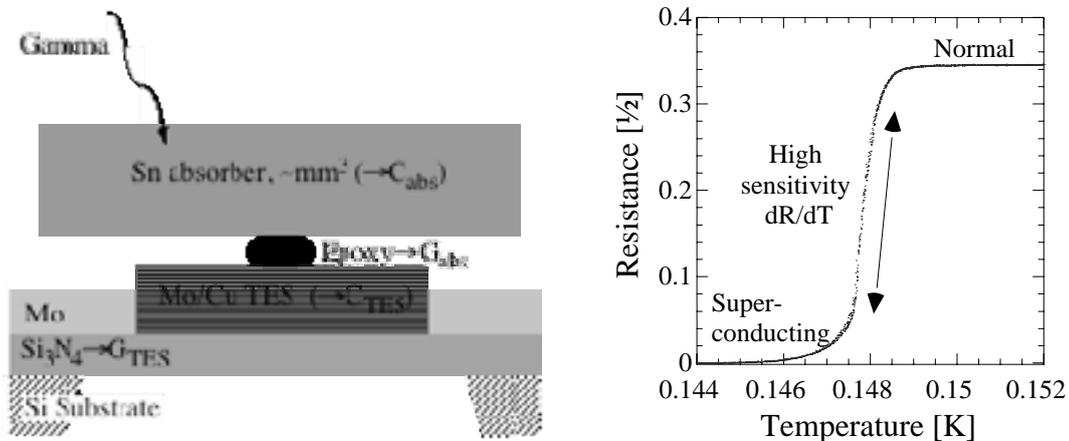


FIG. 1. Schematic of superconducting transition edge sensor (not to scale): Absorption of a Gamma-ray with energy E increases the absorber temperature by E/C_{absorber} . This is measured as a resistance change of the superconducting thermistor, and the detector temperature returns to the base temperature of the silicon substrate with a decay time $\tau \approx C_{\text{absorber}}/G_{\text{total}}$.

The detectors are operated in a two-stage adiabatic demagnetization refrigerator (ADR) (figure 2). The ADR uses a nested design with liquid nitrogen and liquid helium to precool to 77 K and 4.2 K, respectively. It then attains a base temperature of 70 mK by isothermal magnetization and adiabatic demagnetization of two paramagnets. ADRs are compact, reliable, and their cooling cycle has been automated. ADRs are also commercially available, including versions that replace the cryogenic liquids with a mechanical pulse tube cryocooler so that the full cooldown from room temperature can be achieved at the push of a button.

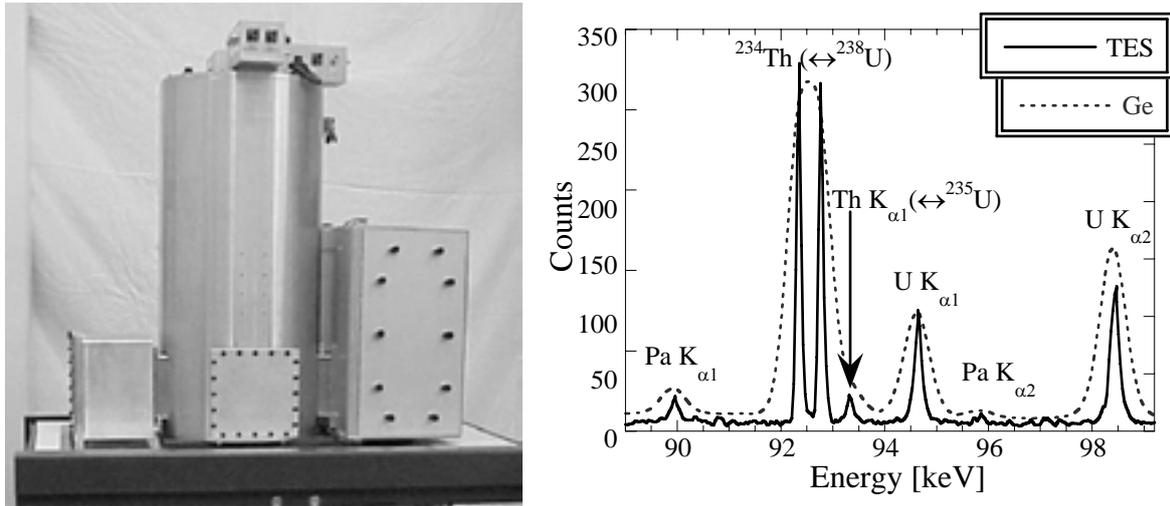


FIG. 2. (Left) Cryogenic spectrometer with detector cold finger. (Right) Gamma-spectrum of a low-enriched uranium sample in the 90 keV region used for precision measurements of uranium enrichment. A spectrum of the same sample from a planar Ge detector is included for comparison.

The spectrum of a low-enriched uranium sample (Figure 2) illustrates the advantage superconducting detectors offer for the analysis of nuclear samples. The 90 keV region of the spectrum is of interest for the analysis of uranium-containing samples, as it can be used for precise measurements of enrichment based on emission lines with very similar energies [1, 8-10]. The Gamma-rays at 92.4 and 92.8 keV originate from the ^{234}Th daughter of ^{238}U , the $\text{Th K}_{\alpha 1}$ X-ray emission at 93.35 keV predominantly arise from the decay of ^{235}U , and their ratio provides a direct measurement of enrichment in samples at least ~ 200 days old where ^{234}Th and ^{238}U are in saecular equilibrium.

These emission lines show significant overlap when analyzed with a Ge detector, and the increased statistical error can limit the analysis in cases of very high or very low enrichment when a weak line lies in the wings of a much stronger line with similar energy. The superconducting TES detectors used in this experiment have an energy resolution of about 90 eV FWHM at 92 keV, and can separate the lines of interest completely above the Compton background. This not only reduces the statistical error from line overlap, but also the systematic error from Compton background subtraction.

Note the difference in linewidth for the nuclear Gamma-ray and the electronic X-ray emission lines. The finite core hole life time adds about ~ 90 eV in quadrature to the observed linewidth, and the Lorentzian line shape of the X-ray emission differs noticeable from the detector-limited Gaussian line shape of the nuclear emissions.

3. Applications in Nuclear Safeguards

The IAEA and other agencies routinely use Gamma-spectroscopy to assess the composition of spent nuclear fuel or swipe samples from nuclear facilities to determine the starting composition of nuclear materials, types of enrichment, burn-up and sample age, and to compare the results to the declarations of the inspected facility. Nuclear safeguards can benefit from superconducting Gamma detectors whenever higher energy resolution can reduce the error of the isotope ratio measurements because of reduced line overlap or improved efficiency corrections and background subtraction. This is not only the case for

measuring Uranium enrichment at 92 keV, but also, for example, in precision measurements of Pu isotopes at 100 keV using multi-group analysis (MGA) [11]. Higher precision speeds up the analysis and allows detecting smaller discrepancies, or better attribution of a sample to its origin in nuclear forensics. It also helps detecting weak lines above background and thus measuring minor isotopes such as ^{242}Pu directly, rather than inferring them through model-based correlations with other isotopes [12]. While mass spectrometry ultimately provides higher sensitivity, the automation and ease of operation makes Gamma spectroscopy the preferred tool for initial inspection of a large number of samples.

To illustrate the potential of superconducting detector technology for nuclear safeguards, we have examined a nuclear swipe sample similar to those typically encountered in nuclear monitoring, using both Ge detectors and superconducting TES detectors.

Figure 3 shows the γ -spectrum of the sample, taken with a large coaxial Ge detector (left), and with a planar Ge detector for highest resolution in the low-energy region (right). The prominent $^{154,155}\text{Eu}$, ^{95}Zr and $^{134,137}\text{Cs}$ emission lines indicate that the sample consist mostly of fission products, in which the short-lived nuclei have already decayed. In addition to the full energy peaks at 1.173 and 1.333 MeV, pair production by the energetic ^{60}Co emission gives rise to single and double escape peaks 511 and 1022 keV lower in energy. The strong ^{60}Co γ -rays and the ^{137}Cs line at 662 keV also produce sum peaks if their arrival times cannot be distinguished by the pile-up rejection.

The feature at ~ 511 keV is likely due to the capture of a single annihilation γ from pair production in the detector housing, rather than to the ^{208}Tl emission at 510.8 keV, since the second ^{208}Tl -line at 583.2 keV is not observed. The ^{40}K emission at 1461 keV is caused by the natural background, and only the feature at 1765 keV can so far not be attributed to a particular decay. Most spectral features are well resolved by Ge detectors, except for a host of lines at very low energies, not an uncommon situation in γ -spectroscopy.

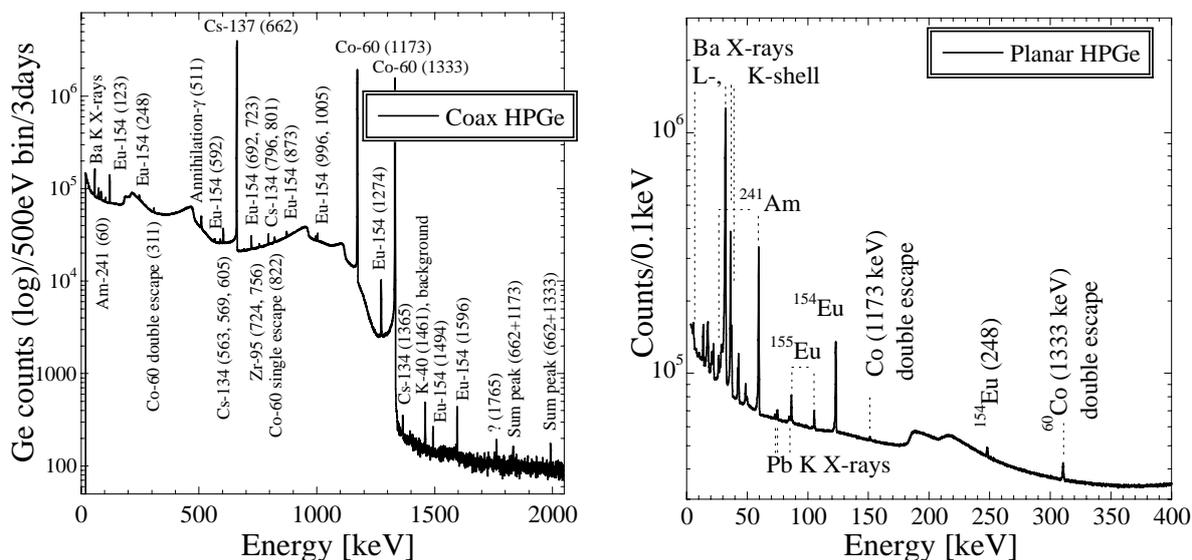


FIG. 3. Spectrum of a sample with fission products typical in nuclear safeguards applications. The energy resolution of conventional Ge detectors is sufficient to resolve higher-energy lines, and the need for superconducting ultra-high resolution detectors mostly arises for energies below ~ 100 keV.

Unfortunately, this is often also the region with the most intense γ -lines, and thus the region on which many analysis routines are based [1, 10] and where minor isotopes would easiest to detect. The detection of dilute or weakly radiating isotopes is further complicated by the high Compton background at low energy. Superconducting TES detectors can provide the resolution required to detect and quantify such isotopes in the presence of strong emissions with similar energy or high Compton background, and are thus a complementary technology to high-purity Ge detectors.

Figure 4 shows an enlarged view of the low energy region of the γ -spectrum, taken both with a TES detector and with a planar Ge detector. Several TES spectra have been added for improved statistics, which reduces the overall energy resolution to ~ 120 eV FWHM. This is still more than sufficient to separate all γ -rays and X-rays. The differences in the observed lines mostly arise from the different materials used in the detector construction and shielding. They give rise to different escape lines (Ge vs Sn from the absorber) and different secondary X-ray emissions (Pb from shielding vs Au from surface plating).

Within the statistics of these experiments, the line shape of TES γ -detectors follows a Gaussian function (Figure 4, right). This is due to the lack of a loss mechanism for the signal-carrying phonons, which are the final form of excitation after γ -absorption in thermal devices such as TESs, provided the only thermal path to the cold bath goes through the superconducting TES sensor (Figure 1).

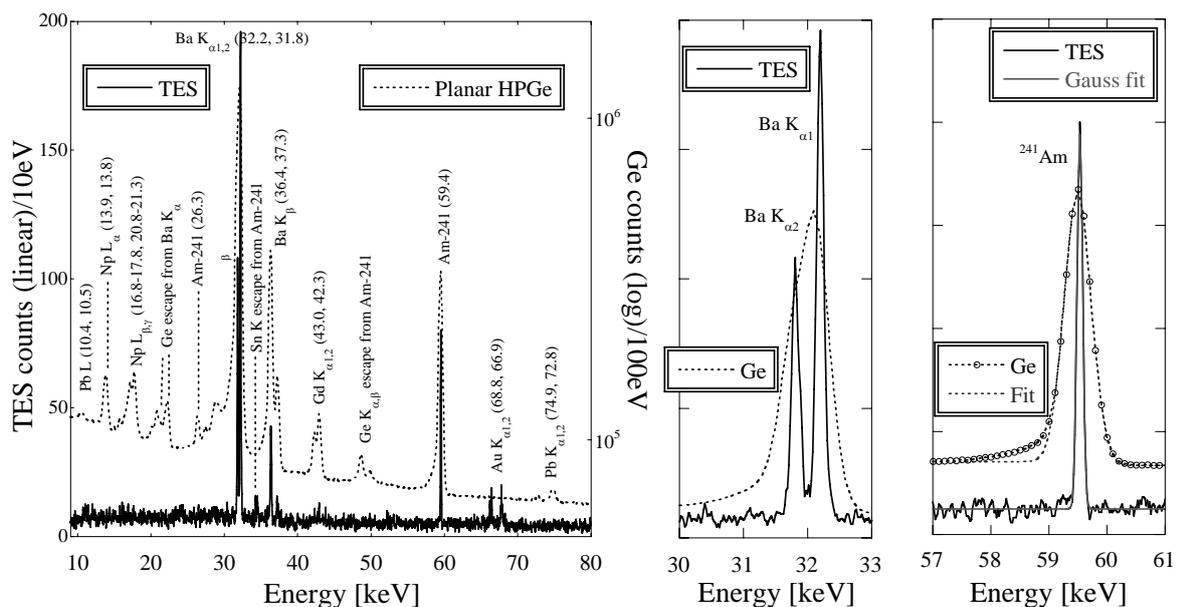


FIG. 4. (Left) Low-energy region of the fission spectrum from a superconducting TES and a planar Ge detector. Note the different escape lines and secondary X-rays due to different detector materials. (Right) High energy resolution not only allows the separation of closely spaced lines, but also improves the detection of lines above the Compton background.

These spectra show both the promise and the current challenges for superconducting detector technology. Their high energy resolution effectively removes all line overlap from the spectra, and for a fixed number of counts improves the peak-to-background ratio for detecting weak lines. In fact, the limiting resolution can be adjusted as needed by varying the absorber size and thus its heat capacity. High resolution can also reduce systematic errors through better

background subtraction between peaks, especially if the line shapes remain Gaussian for improved statistics [11].

The current challenge for TES technology remains to increase the sensitivity of the detectors and reduce the statistical error of the spectra. Since the volume of each pixel is limited by the need for high energy resolution according to equation (1), current research and development focuses on the development of pixilated detector arrays [13, 14].

4. Summary

Nuclear safeguards can benefit from improved precision in non-destructive isotope analysis to determine starting composition, types of enrichment, burn-up and age of nuclear materials and compare it to declared values. In this context, superconducting transition edge sensor (TES) γ -ray detectors are a promising technology, since they offer an order of magnitude improvement in energy resolution over current high-purity Ge detectors. This reduces statistical errors from line overlap and systematic errors from background subtraction and efficiency corrections, especially important for low energy γ -rays around ~ 100 keV for MGA-type analysis. We are developing detectors based on Sn absorbers and Mo/Cu multilayer TESs, which have, depending on design, achieved an energy resolution between 50 and 90 eV at 100 keV. They can increase the precision of non-destructive isotope analysis, identify minor isotopes in complicated mixed-isotope samples, and reduce errors in verifying compliance with international treaties.

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Autoradiography using optically stimulated luminescence for verification and safeguards applications

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Abstract: New storage phosphors for ionizing radiation detection have been under development at the Pacific Northwest National Laboratory (PNNL) since the mid-1980s. The technology, known as either optically stimulated luminescence (OSL) or storage phosphors, provides a more sensitive alternative to thermoluminescent dosimeters and silver halide-based x-ray film. Current PNNL research is focused on development of collimators that can be placed in front of the OSL image plates that sufficiently distort the visible image so that no size or shape information could be gained from the image itself without detailed knowledge of the collimator and a special mathematical unfolding of the image. To date, our research has shown that this masking, known as coded-aperture imaging (CAI), is potentially the most powerful. The goal is to develop a CAI design that can measure important attributes of a radioactive object without revealing any sensitive information.

Introduction

The Pacific Northwest National Laboratory (PNNL) has been developing new storage phosphors for ionizing radiation detection since the mid 1980s [1,2,3]. The materials are inorganic-based phosphors that uniquely store ionizing radiation damage. The technology is known as either optically stimulated luminescence (OSL) or, alternatively, storage phosphors [4]. The phosphors can be used as a more sensitive alternative to thermoluminescent dosimeters or silver halide-based x-ray film. In addition to widespread application in personnel dosimetry, the phosphors can be fabricated into large two-dimensional sheets and can replace x-ray film in most applications. OSL phosphors are analyzed by illumination with light and emit a luminescence that is proportional to the ionizing radiation exposure. Raster-scanning across a two-dimensional plate can determine an image of the ionizing radiation field. Advantages over x-ray film include a wider dynamic range, higher sensitivity, and the elimination of chemical processing. Our primary development client has been the U.S. Department of Energy (DOE), but PNNL has transferred the technology to industry, resulting in two R&D 100 awards and numerous patents. Our current research is focused on developing special collimators that can be placed in front of the OSL image plates that sufficiently distort the visible image so that no size or shape information could be gained from the image itself without detailed knowledge of the collimator and a special mathematical unfolding of the image. This type of masking is known as coded-aperture imaging (CAI).

Collimator Development Using OSL Image Plates

OSL imaging plates are superior to silver halide-based x-ray film because of their higher quantum efficiency, much wider dynamic range, reusability, and no chemical development. Some smaller-format commercial OSL film readers based on BaFBr:Eu are reasonably priced and easily can be taken to the field for imaging measurements. Our measurement goals were to determine if the measured object was radioactive and consistent with the inspected party's declaration of nuclear materials and processes without revealing sensitive information about the object. Physical inspection of the object or intrusive radiation measurements would likely not be allowed in most scenarios.

Pinhole collimators initially were considered for safeguards and dismantlement applications but were rejected for two reasons. First is the low sensitivity of the pinhole technique. The ratio of the open area of the pinhole to the overall area of the collimator is a good estimate of how many gamma rays will reach the image plate. The second reason is the quality of the image; the pinhole technique is too accurate dimensionally when measuring a sufficiently strong gamma source. For these reasons, we initially decided to use a venetian-blind-type collimator with around 1 cm spacing. This collimator can be seen in Figures 1 and 2.

A second, two-plate collimator design was tested. Both the first and second plate had two holes placed to image an extended radioactive object. The conceptual design is shown in Figure 3; the constructed collimator is pictured in Figure 4.



Figure 1. Venetian-blind collimator measurement setup.

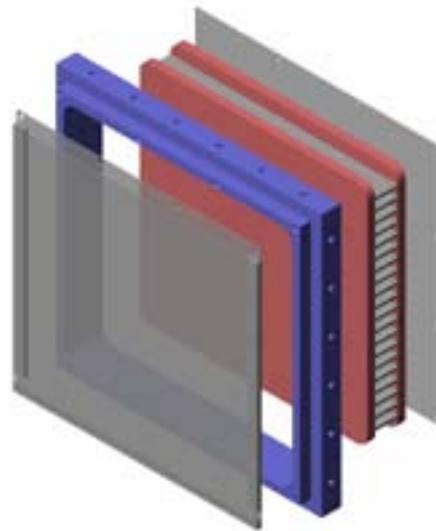


Figure 2. Exploded view of venetian-blind collimator.

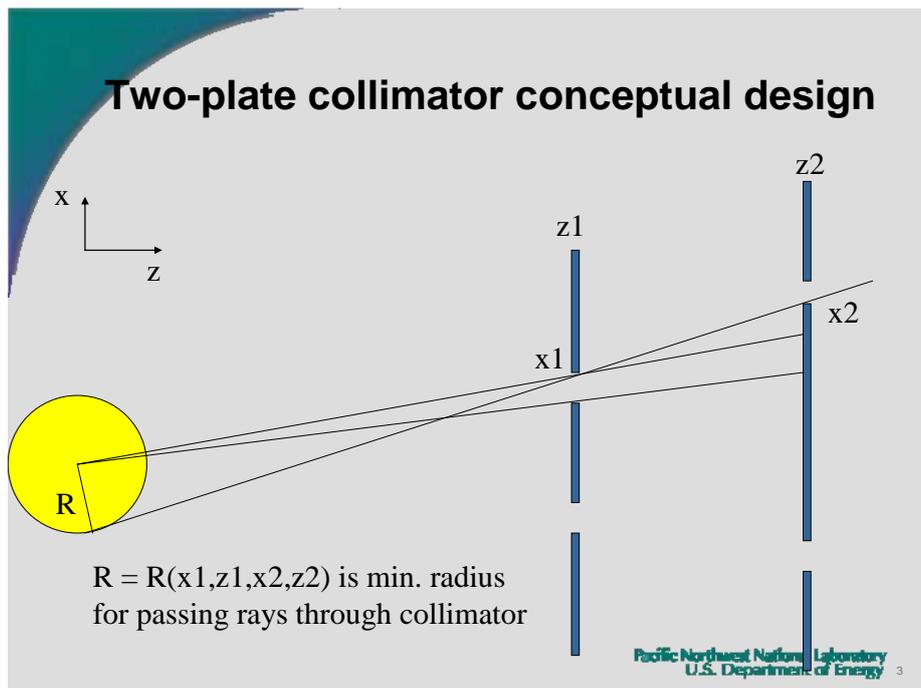


Figure 3. Two-plate collimator conceptual design.



Figure 4. As-built two-plate collimator.

To test these new collimator designs, we modeled aluminum hemispheres with a cobalt-57 gamma source to optimize gamma-ray scatter. The goal was to produce extended gamma-ray sources to test our collimator designs without the complexity of measuring sensitive components. We built 3-in., 4.5-in., and 6-in. aluminum hemispheres to house a cobalt-57 gamma point source. A camera stand was used to adjust the height of the aluminum hemispheres. The hemispheres as built are shown in Figure 5. The black plastic object in Figure 5 is to hold the cobalt-57 source for a point source measurement.

The aluminum hemispheres being used with a collimator in a measurement configuration are shown in Figure 6.



Figure 5. Aluminum hemispheres.



Figure 6. Using the hemispheres in a measurement scenario.

This most recent approach in collimator design for detecting sources of varying geometry is the most versatile and promising technique to date. Pinhole-based collimators can be unpractical in some situations due to their reduced throughput. The venetian-blind collimator can reveal sensitive information in some circumstances and produces images that are difficult to interpret in measurement situations with a great deal of scattered radiation. For these reasons, we have begun to develop coded-aperture image (CAI) techniques. The advantages of CAI are very extensive, especially in comparison to pinhole collimators. Coded apertures do not suffer from the reduced throughput issues that pinhole collimators do. For nearly any coded aperture, the open area of the collimator is roughly always 50%. This allows for a much greater throughput, resulting in increased efficiency and decreased exposure time of the imaging plate to obtain a useful image. Another interesting feature of CAI is that a source at any position in the aperture field of view will place a unique pattern on the detector positioned behind the aperture. This unique pattern, when mathematically reconstructed, provides the location of the source within the field of view, giving CAI the ability to not only detect sources but also to provide information on where the source is located. Our initial CAI is pictured in Figure 7.

The initial CAI array is based on the principle of a modified Uniform Redundant Array (mURA). The CAI is built from 3-mm tungsten and is an example of a 5 x 5 repeating mURA structure. Results from MCNP5 code modeling of this array pattern with a cobalt-57 gamma point source and 4.5-in. and 6-in. hemispheres fitted with a cobalt-57 source are shown in Figures 8, 9, and 10.

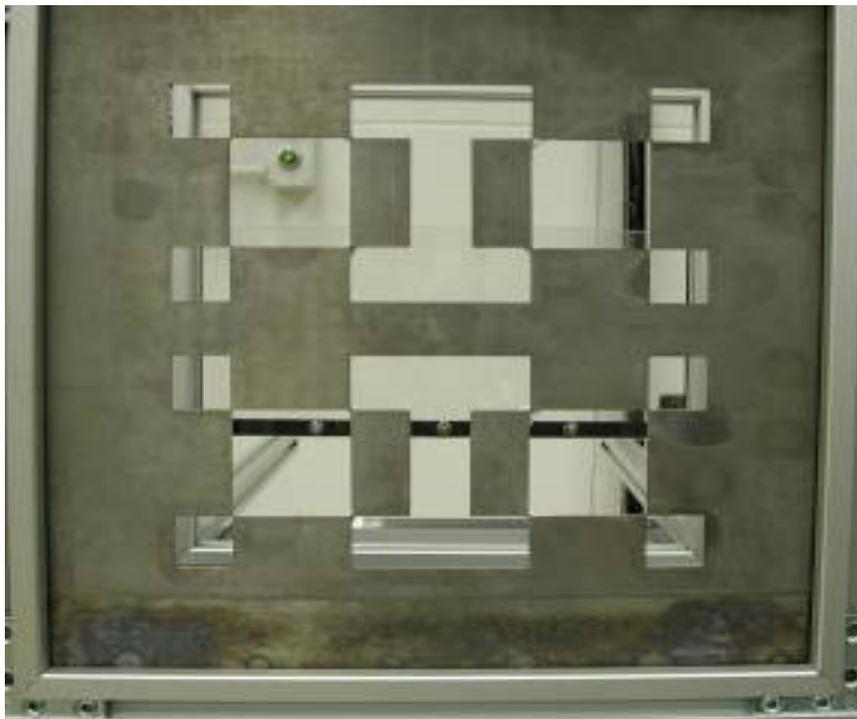


Figure 7. Initial coded aperture image array.

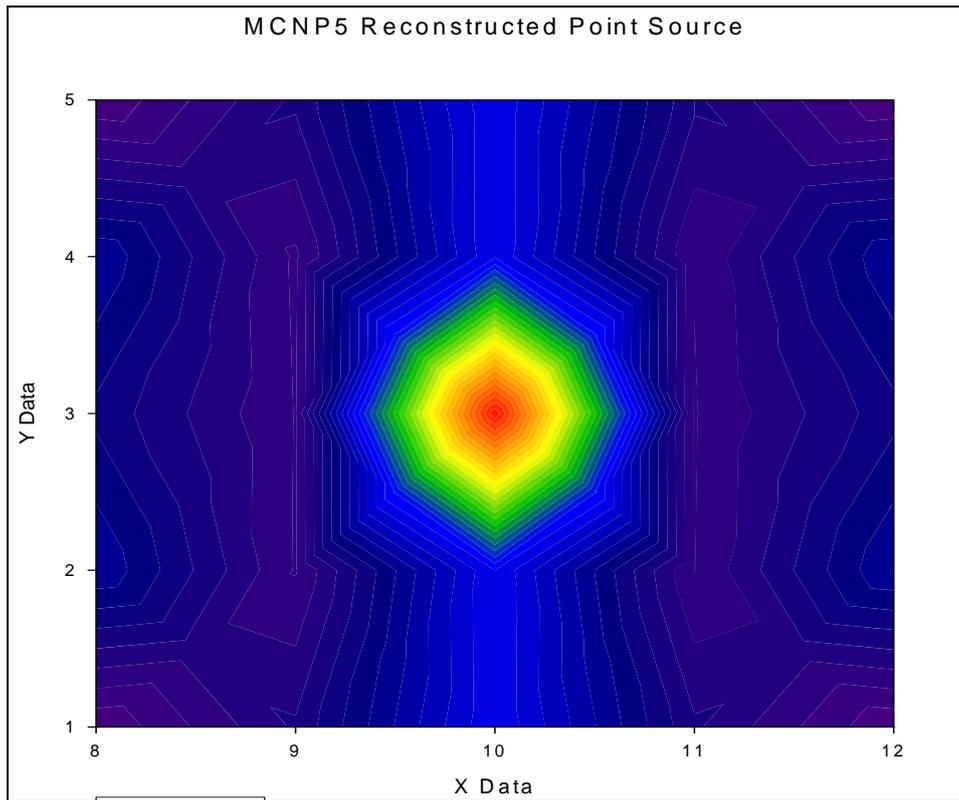


Figure 8. Reconstructed cobalt-57 point source.

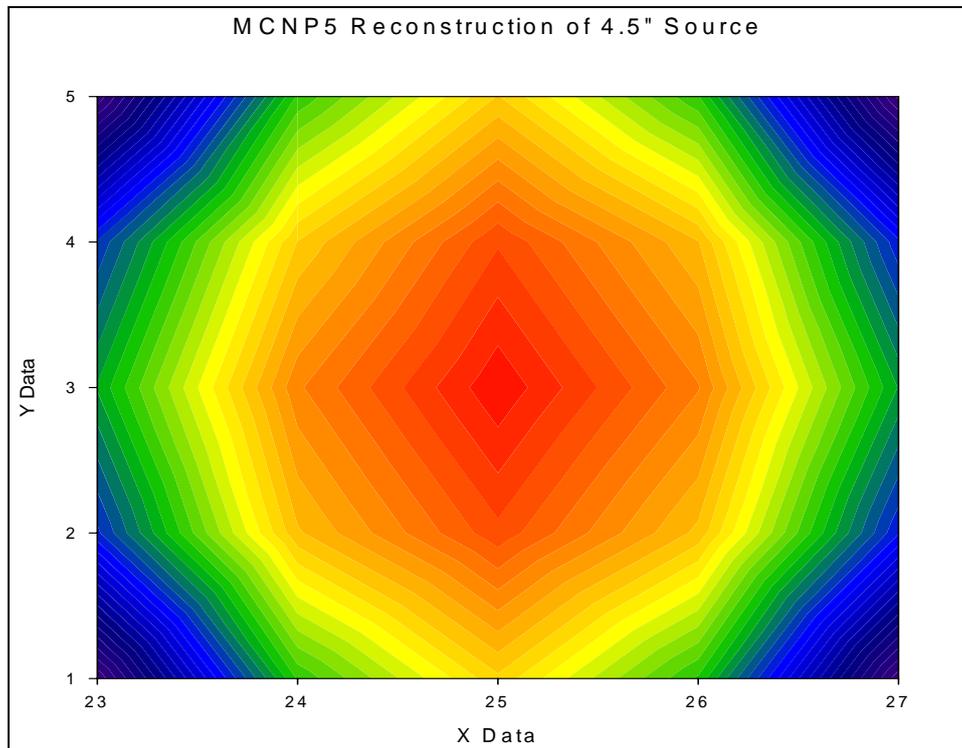


Figure 9. Reconstructed 4.5-in. aluminum hemisphere with cobalt-57 source.

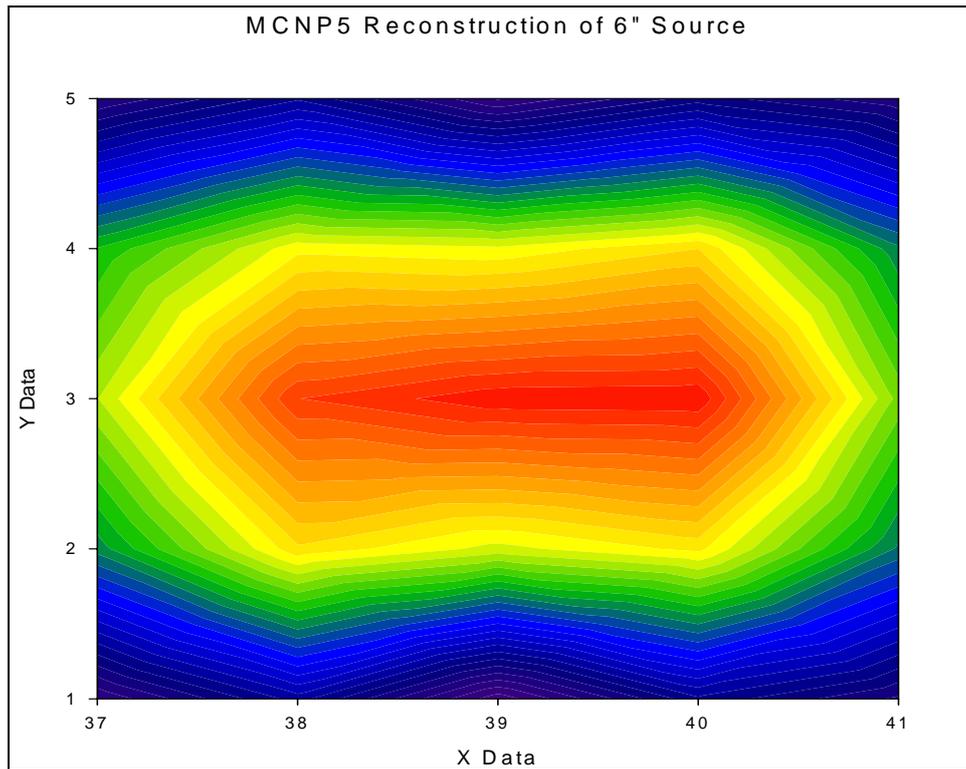


Figure 10. Reconstructed 6-in. aluminum hemisphere with cobalt-57 source.

Conclusion

Several collimator designs tested at PNNL for safeguards and dismantlement applications using OSL imaging demonstrated that potentially the most powerful is the CAI array design. The goal is to develop a CAI design that can measure important attributes of a radioactive object without revealing any sensitive information.

ACKNOWLEDGEMENTS

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Simulation smuggling study for shielded sources of ^{235}U , ^{238}U , ^{232}Th , ^{137}Cs and ^{60}Co by using γ -detection

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Abstract. This work deals with gamma detection of shielded sources of ^{235}U , ^{238}U , ^{232}Th , ^{137}Cs and ^{60}Co by using HpGe and NaI detectors through their main characteristic energies; 185.7 keV, (63, 1001), (583 and 911), 661.6 and (1173 and 1332) KeV respectively. These radioactive sources were individually hid in containers of aluminum (Al), iron (Fe) and copper (Cu) of different thicknesses; 3, 6, 9, 12, 15, 18 mm. Thin containers of lead (Pb) were also used with the same sources. The attenuation factor (I/I_0), the ratio of radiation intensity with and without container or shield, was calculated for all the investigated γ -energies. The relationship between the attenuation factor and the studied thicknesses of the containers was given, depicted and discussed. Detection of the mixed sources shielded by different materials of different thicknesses was also carried out by using PDR-77 survey meter. The radiation levels were found to be within the natural background at 65cm distance from the shielded sources. Telescopic arm carrying 2x2" NaI and radiometer was also used. Generally, It can be concluded that smuggling of radioactive materials of low radioactivity can be done without detection by hiding them in containers or within scrap of Cu, Fe or even Al rather than using the conventional heavy lead shield.

1. Introduction

The identification of shielded radioactive or nuclear sources by gamma spectrometry is still an unsolved problem. Hand-held devices used by custom officers at borders suffer from this limitation (1).

In the present time, smuggling and illicit trafficking of nuclear and radioactive materials are targeted by international groups and networks. Hundreds of smuggled nuclear and radioactive materials have been seized and currently recorded in the illicit trafficking database bank (ITDB) of the International Atomic Energy Agency (IAEA). As a matter of fact, these materials can be used for manufacturing of dirty or radiological dispersal device (RDD) for sabotage and malicious purposes.

After decomposition of the Former Soviet Union in the beginning of 90's, many nuclear (e.g. high enriched uranium (HEU), low enriched uranium (LEU), plutonium) and radioactive materials (e.g. natural uranium (NU), depleted uranium (DU), thorium,...) have been stolen and smuggled, illegally transported to another state, through several international borders. The theft and smuggling of nuclear materials can pursue different goals (2). One is commercial, that is, resale to third party with the purpose of obtaining personal financial profits. Another is terrorist, namely the malevolent use of stolen nuclear materials for terrorism or blackmail (2).

There are hundreds of thousands of curies in radioactive sources are currently available for potential use in a terrorist attack. This level of radiation has the potential to cause fatalities and significant disruption of the economy and society (3).

As of December 2003, the IAEA's illicit trafficking database contains 540 confirmed incidents involving nuclear and other radioactive materials, 182 of them with nuclear material, which have occurred since 1 Jan. 1993 (4) (see Fig. 1).

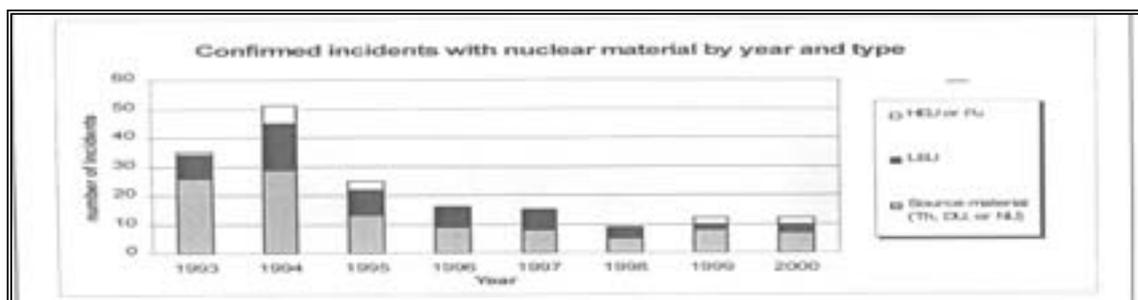


FIG. 1. Trafficking incidents with nuclear and source nuclear materials.

This study concerns with simulation by gamma detection for smuggling of radioactive materials hid or shielded by cheap scrap materials like Al, Fe or Cu rather than lead. Efficiency of detection of the investigated shielded radioactive sources is also a target of the present work.

2. Experimental work

Containers of different materials and thicknesses of Al, Fe and Cu were manufactured in our workshop to carry out this work. The thicknesses of the containers were 3, 6, 9, 12, 15, 18 mm. Thin lead (Pb) containers of different thicknesses were also used. Gamma spectrometry based on HpGe and NaI detectors was used to carry out the analyses. The HpGe spectrometer is Canberra type of 25% efficiency and 1.95 FWHM at 1332 keV of ⁶⁰Co. The scintillation detector used in this work is Ortec type of 3" x 3" NaI crystal. The gamma spectrometers were energetically calibrated before measurements. The radioactive materials of uranyl nitrate powder containing natural ²³⁵U and ²³⁸U, ²³²Th (30nCi), ¹³⁷Cs (1.99μCi) and ⁶⁰Co (1.99μCi) were used in this simulation study.

The main gamma transitions to identify the investigated γ-emitter sources are given in Table 1.

Table 1. The main gamma energies of the analyzed radioactive materials.

Sources	Energy (keV)
²³⁵ U	185.7
²³⁸ U (²³⁴ Th, ^{234m} Pa)	63.3 & 1001
²³² Th (²⁰⁸ Tl, ²²⁸ Ac)	583 & 911.2
¹³⁷ Cs	661.6
⁶⁰ Co	1173.2 & 1332.5

The attenuation factor (I/I_0) of the investigated gamma rays was calculated using the following equation (5):

$$I = I_0 e^{-\mu x} \quad (1)$$

or $I / I_0 = e^{-\mu x} \quad (2)$

Where I_0 is the incident γ -rays intensity, I is the intensity observed at thickness (x) and μ (cm^{-1}) is the linear attenuation coefficient (5).

Detection and inspection of the mixed sources shielded in Al, Fe and Cu containers of different thicknesses were also carried out by using RADIAC Meter, PDR-77 survey meter. The surevymeter was calibrated by using standard field calibrator; Model FC-1 APTEC containing $8\mu\text{Ci}$ of ^{137}Cs . Three meter telescopic arm carrying 2×2 " NaI and Thermo FH40G-L10 Radiometer was also used to detect and monitor of the shielded sources.

3. Results & Discussion

The gamma attenuation factors of the investigated gamma energies based on HpGe detector measurements for different materials and thicknesses were calculated by the equation no. 2 and given in the following figures from 2 to 5.

It can be observed that at 18mm thickness of Al container, more than 77% and 79% of the 63 and 185.7 keV gamma lines of ^{238}U (^{234}Th) and ^{235}U were attenuated. While the slightly energetic gamma line 1001 keV of ^{238}U ($^{234\text{m}}\text{Pa}$) was attenuated by 64% only at the same thickness of Al (Fig. 2).

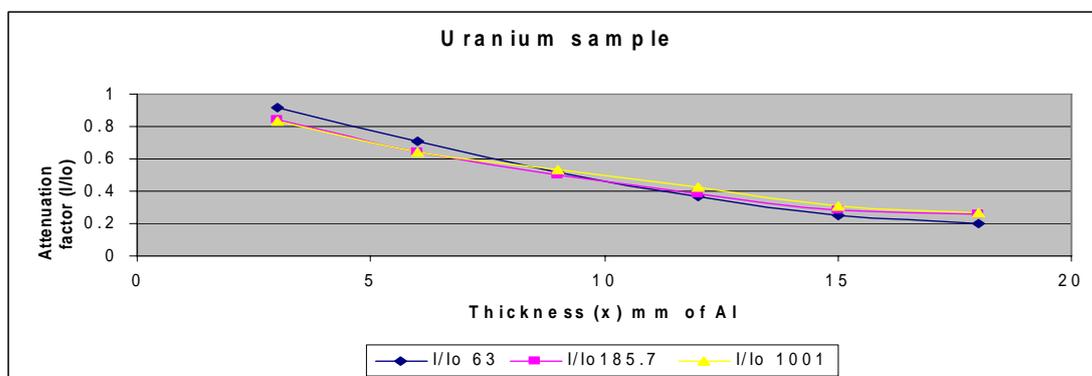


FIG. 2. The attenuation factor of uranium gamma lines at different Al thicknesses.

The low energy gamma line 63keV of ^{238}U (^{234}Th) was completely attenuated at 9mm thickness of Cu and Fe containers. While the gamma lines; 185.7 and 1001 keV, of ^{235}U and ^{238}U were attenuated at the 18mm thickness by 99% and 93% respectively. This may lead to the fact that uranyl nitrate powder can be hidden, shipped using Fe or Cu scrap or even by slightly thicker Al containers and then passed through monitoring gates at borders without noticeable detection. It was also observed that at 18mm of Al container, 70% of ^{232}Th activity, 64% of ^{137}Cs activity and 58% of ^{60}Co activity were hindered (Fig. 3).

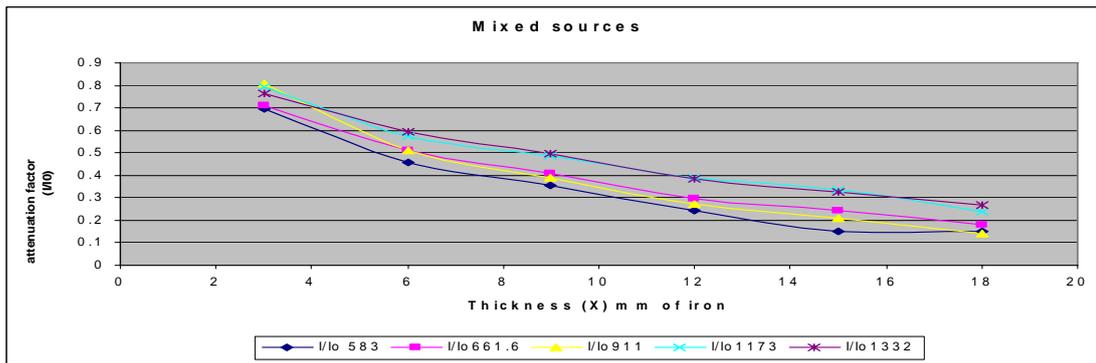


FIG. 3. The attenuation factor of Th, Cs, and Co γ -lines at different Al thickness

The attenuation percentage of $\sim 85\%$ of ^{232}Th , 84% of ^{137}Cs and 76% of ^{60}Co were observed by using Cu and Fe container at the same thickness (figs. 4,5). The variation observed in the attenuation percentage is mainly attributed to the difference in the emitted gamma energies and material of the containers used.

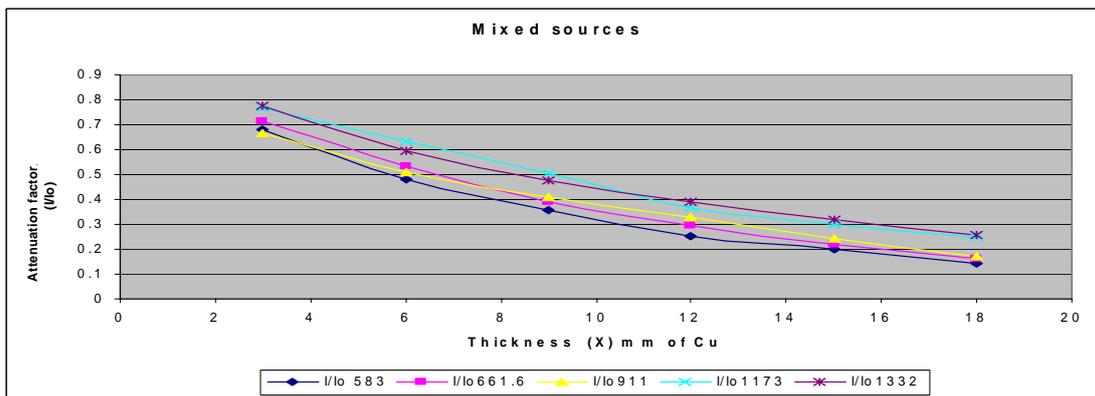


FIG. 4. The attenuation factor of Th, Cs and Co γ - lines at different Cu thicknesses

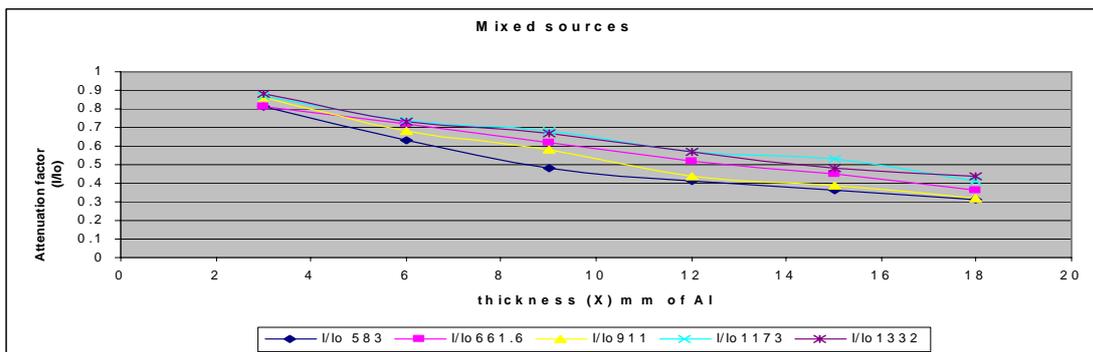


FIG. 5. The attenuation factor of Th, Cs and Co γ - lines at different Fe thicknesses.

By using thin containers of Pb, the results show, as expected, very high attenuation percentage ($>98\%$) for the low energies 63 keV and 185.7 keV of ^{238}U (^{234}Th) and ^{235}U . While, the energetic 1001 keV of ^{238}U ($^{234\text{m}}\text{Pa}$) was attenuated by $\sim 78\%$ at 4.7mm thickness (see Fig. 6).

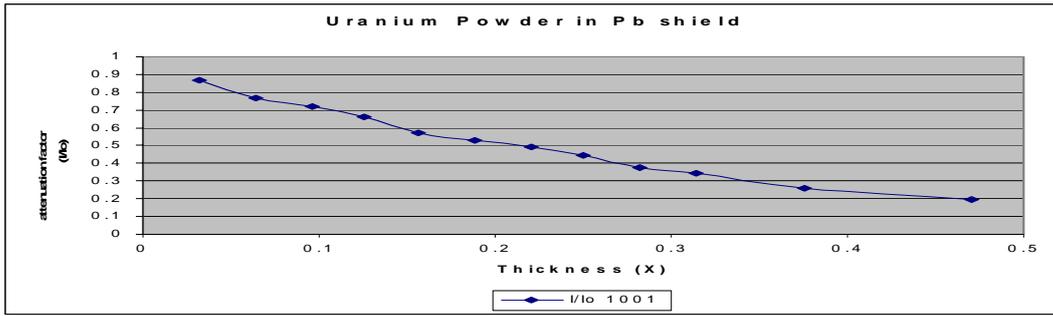


FIG . 6. The attenuation factor of ^{238}U (^{234m}Pa) γ - line at different Pb thicknesses.

The results obtained by using high efficiency NaI detector show overlapping between the multi-gamma lines of the investigated mixed sources. Many of the studied γ -lines were disappeared or could not identified due to Compton effect and the bad resolution characteristic of NaI detector.

The calculated attenuation of the gamma rays of ^{137}Cs and ^{60}Co at different thicknesses of Pb based on HpGe detector analysis is shown in figure 7. It was observed that at 4.7mm thickness of Pb, $\sim 92\%$ of 661.6 keV of ^{137}Cs and $\sim 83\%$ of 1173 and 1332 keV gamma energies of ^{60}Co were attenuated (Fig. 7).

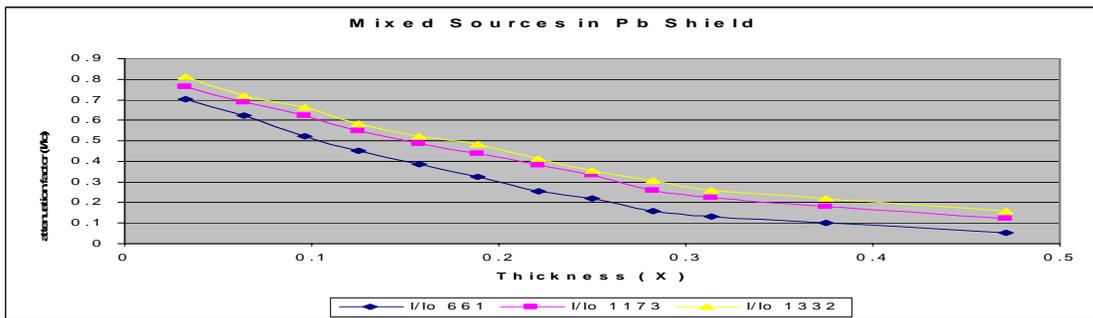


FIG. 7. The attenuation factor of Cs and Co γ - lines at different Pb thicknesses.

The comparison of attenuation percentage (%) of the studied gamma transitions of the investigated shielded mixed sources at 18mm is given in Figure 8.

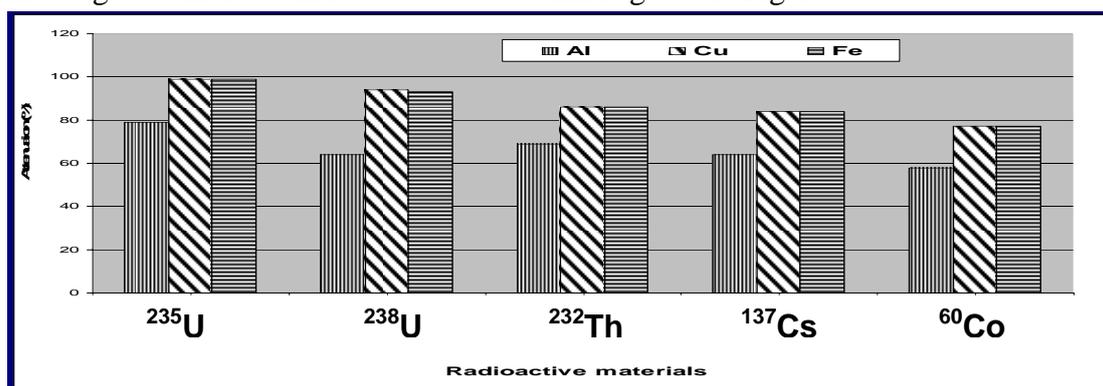


FIG. 8. Comparison of the attenuation percentage of the studied materials.

This comparison shows clearly that Cu and Fe containers are more effective to be used for radioactive materials smuggling than Al at the same thickness.

Generally, It can be concluded that radioactive materials of small activity can easily be smuggled by hiding them in containers or scrap of Cu or Fe or even Al rather than using the conventional heavy high atomic number lead.

Inspection or gamma detection of the sources at surface of the containers (shield) was carried out by PDR77 surveymeter. The surveymeter readings at the surface of the containers and the corresponding attenuation percentages of radiation levels for different container's thicknesses are given in Table 2.

Table 2. Results of Surveymeter Detection of the Shielded Mixed Sources.

Container's thickness	Average	surveymeter	readings
	Attenuation	percentage (%)	
	Fe	Cu	Al
Without container	0.99 mR/h		
3 mm	0.759 (23%)	0.701(~29%)	0.865(~ 13%)
6 mm	0.603 (39%)	0.601(39%)	0.719(~ 27%)
9 mm	0.536 (~45%)	0.53(~ 46%)	0.633(36%)
12 mm	0.395 (60%)	0.381(~ 62%)	0.562(43%)
15 mm	0.344 (~65%)	0.322(~68%)	0.47(53%)
18 mm	0.283(~71%)	0.246(75%)	0.409(~ 59%)

It was found that the radiation levels due to the mixed sources shielded by Cu or Fe or Al containers of 18mm thickness attenuated by 75%, 71% and ~59% respectively. While ~80% was obtained at 4.7mm of Pb. This means that slightly thicker containers of Cu or Fe or Al can be used to totally eliminate the radiation levels emitted from the smuggled radioactive sources to the undetectable levels.

The effect of distance from the shielded mixed sources on the measured radiation levels at different thicknesses and materials was carried out (Fig. 9).

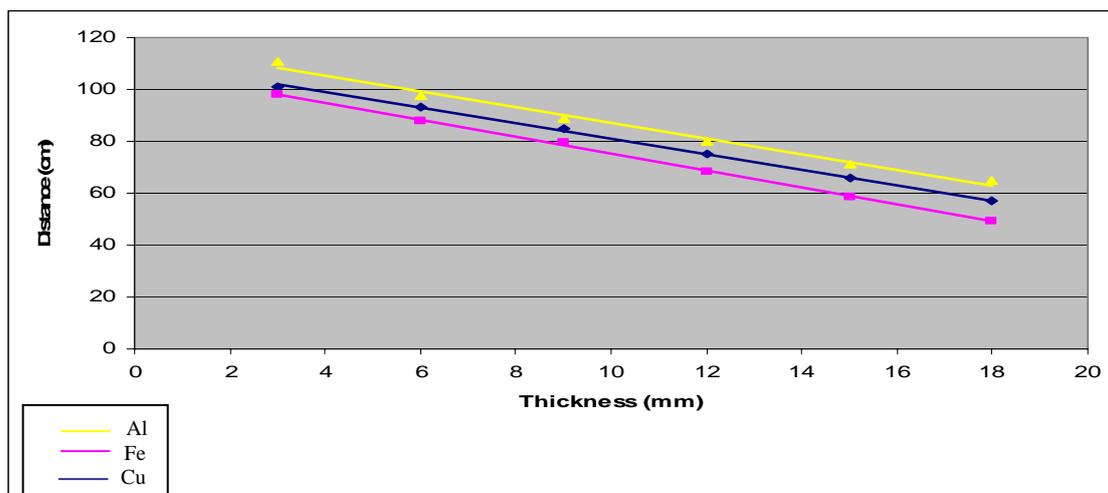


FIG. 9. The relationship between the container thickness and the distance from the source.

The radiation levels of the mixed sources were found to be undetected at 50 and ~65 cm distance from the surface of 18mm of Fe, Cu and Al containers respectively.

There were no changes in the radiation levels more than the background by using NaI mounted on 3 meter telescopic arm for detection of the shielded sources

4. Conclusion

Many alternative scrap materials such as Fe, Cu and Al can also be used as containers for smuggling of radioactive materials with low detection probability. Due to the gamma lines overlapping, it was not easy to detect and identify many of the analyzed gamma transitions of the studied mixed sources ^{235}U , ^{238}U , ^{232}Th , ^{137}Cs and ^{60}Co by NaI detector. It was observed that ~76 - 100% of the investigated γ -lines attenuated at 18 mm thickness of Cu and Fe containers. While 57 - 79% was obtained in the case of Al container at the same thickness. The radiation levels due to the shielded sources were completely vanished to the undetectable level at distance of 50 - ~65cm from the sources. These results indicate that development of the hand-held and gates monitoring counters at borders should be considered to avoid further nuclear smuggling operations.

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Prototype tomographic partial defect tester

Project status update

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Abstract. Passive Gamma Emission Tomography (PGET) has been extensively studied during the past years. The feasibility study and the tests performed show that the device is capable of detecting missing pins inside a PWR assembly. A construction project of the prototype device was enabled as IAEA Joint Support Program task: "Prototype Tomographic Spent-Fuel detector system" (JNT A 1510). The participants to the task are: the Finnish Support Program to the IAEA Safeguards (FIN SP, construction of mechanical structures and coordination), Hungarian SP (system theory, analysis and control software, detection system specification and testing), Swedish SP (hosting the final testing and arranging additional analysis of the data) and the European Commission SP (facilitating the test campaign). The IAEA has defined the user requirements for the device. The IAEA expects that the device is capable of verifying fuel with a minimum burn-up of 15 GWd/t and a cooling time up to 40 years. Measurement time per assembly should be less than 10 minutes. The analysis software should automatically determine whether pins are missing or replaced by other than irradiated fuel material. Although dedicated training will be developed to train IAEA inspectors, no specific expertise should be needed to operate the equipment. Delivery of the detector arrays will take place in spring 2007. In the meanwhile, the mechanical components will be manufactured and the software written to the extent possible. After the delivery of the detector array the software will be finalized, and the detection system will be tested at a Co-60 facility, and the device will be assembled in final form. According to the project plan the final testing will take place in late 2007 at Ringhals NPP with spent fuel assemblies. Then the device will be ready for the IAEA. The PGET will be the first device for single rod verification by the IAEA, therefore greatly enhancing IAEA's verification capability.

1. Introduction

Verification of the contents of spent nuclear fuel assemblies is one of the basic safeguards measures routinely carried out by authorities and inspectorates. The basic objective is to gain assurance that the operator declared data concerning isotopic contents and mass are correct. In addition to correctness, also completeness of the data needs to be verified to gain assurance that no material is missing. With introduction of the Integrated Safeguards (IS), increased cooperation between the State systems (SSAC) and inspectorates may offer more flexibility in carrying out the actual measurements. This

doesn't, however, change the basic need to create the knowledge first before its continuity can be maintained.

Verification measurements are carried out on different levels depending on the need. Gross defect level measurements result in a conclusion whether the assembly verified is completely missing or replaced with a dummy. If a higher level of assurance of the lack of diversion is needed, partial defect level verification may be necessary. The IAEA definition of the meaning of a partial defect has been for years limited by the sensitivity of the methods available to reveal defects. Missing or replacement of 50 % or more of the irradiated fuel rods in a spent fuel assembly has been the defined level for a partial defect verification method.

The limited power of the verification methods available for the IAEA to use for spent fuel measurements in field conditions has been known for years. The limiting factor in developing such methods has been technical in nature. No physical or technical principles have been known to allow the development of a practical method for field use by the IAEA. After years of testing and developing different potential methods, as requested by the IAEA, the only passive method available to have real detecting power seems to be the method based on the use of passive gamma emission tomography. Earlier studies have shown the capability of Passive Gamma Emission Tomographic Verifier as a partial defect tester [1]. The device is capable of detecting missing pins inside a PWR assembly, and when approved, it will enhance the safeguards verification capability substantially.

2. Method

The basic idea of the passive gamma emission tomography is mapping of the emitted radiation by imaging techniques. The emitted radiation is detected using a directionally sensitive detector-collimator system followed by an image reconstruction. The reconstructed image gives a rod-to-rod distribution of the gamma emitter concentration of the object. Replacement or missing of irradiated fuel rods can be detected by visual or by computer supported evaluation of the image.

The high sensitivity to reveal missing of irradiated fuel rods is explained by the following facts:

- No need for a reference data set because the activity map provides an inherent rod-to-rod comparison of fission product gamma activities.
- The effect of a single missing rod is very low in one orientation, usually lower than the noise level. The image reconstruction process uses, however, all scanned data for calculating each image point. Noise and statistical fluctuations in different data sets are uncorrelated and the averaging effect improves the signal to noise ratio.

3. Test results

Both BWR 8x8 and PWR 17x17 type assemblies has been measured with experimental setups in 1999 and in 2001. The test arrangements have not been optimal; however, the results show the capacity of the method. The theoretical simulation tools have been developed and verified using the test results. Simulations predict that the detection of a single inner missing pin of 17x17 type assembly is detected with high (> 96%) probability in optimal geometry and reasonably low noise conditions. At peripheral positions the detection is certain. The simulations will also be used in design work of the prototype.

4. Establishment of the prototype construction task

The encouraging results were noted in Concerted Technical meeting arranged by the Agency in March 2003. The meeting recommended building a prototype of the device. However, due to the high price of the detector array the project could not start immediately.

Finally, in April 2005 the IAEA was in the position to decide to finance the construction of the prototype. As a consequence a construction project of the prototype device was enabled as IAEA Joint Support Program task: "Prototype Tomographic Spent-Fuel detector system" (JNT A 1510). The participants to the task are: the Finnish Support Program to the IAEA Safeguards (FIN SP, construction of mechanical structures and coordination), Hungarian SP (system theory, analysis and control software, detection system specification and testing), Swedish SP (hosting the final testing and arranging additional analysis of the data) and the European Commission SP (facilitating the test campaign).

5. IAEA requirements for the prototype

The IAEA has defined the user requirements for the device. The IAEA expects that the device be capable of verifying fuel with a minimum burn-up of 15 GWd/t and a cooling time up to 40 years. Measurement time per assembly should be less than 10 minutes. The analysis software should automatically determine whether pins are missing or replaced by other than irradiated fuel material. Although dedicated training will be developed to train IAEA inspectors, no specific expertise should be needed to operate the equipment.

These criteria were taken as a design basis for hardware and software. Also the procurement specifications of the detector arrays were formulated so that these criteria can be met. However, for cost reasons, the detector array specifications were eased. For low burn-up and long cooling time PWR assemblies, increased measurement time may be needed.

6. Present status and schedule:

The procurement order of the Detector array was placed in 1st March 2006. Delivery of the detector arrays will take 12 months. The hardware has been designed and construction will start at the end of 2006.

The exploded view of the designed detector head is presented in Figure 1. The device will be toroidal in shape, and the measurement position will be inside the torus. Its dimensions will allow the measurement of BWR and PWR fuel assemblies including the VVER 1000 and the PWR 17x17 assemblies.

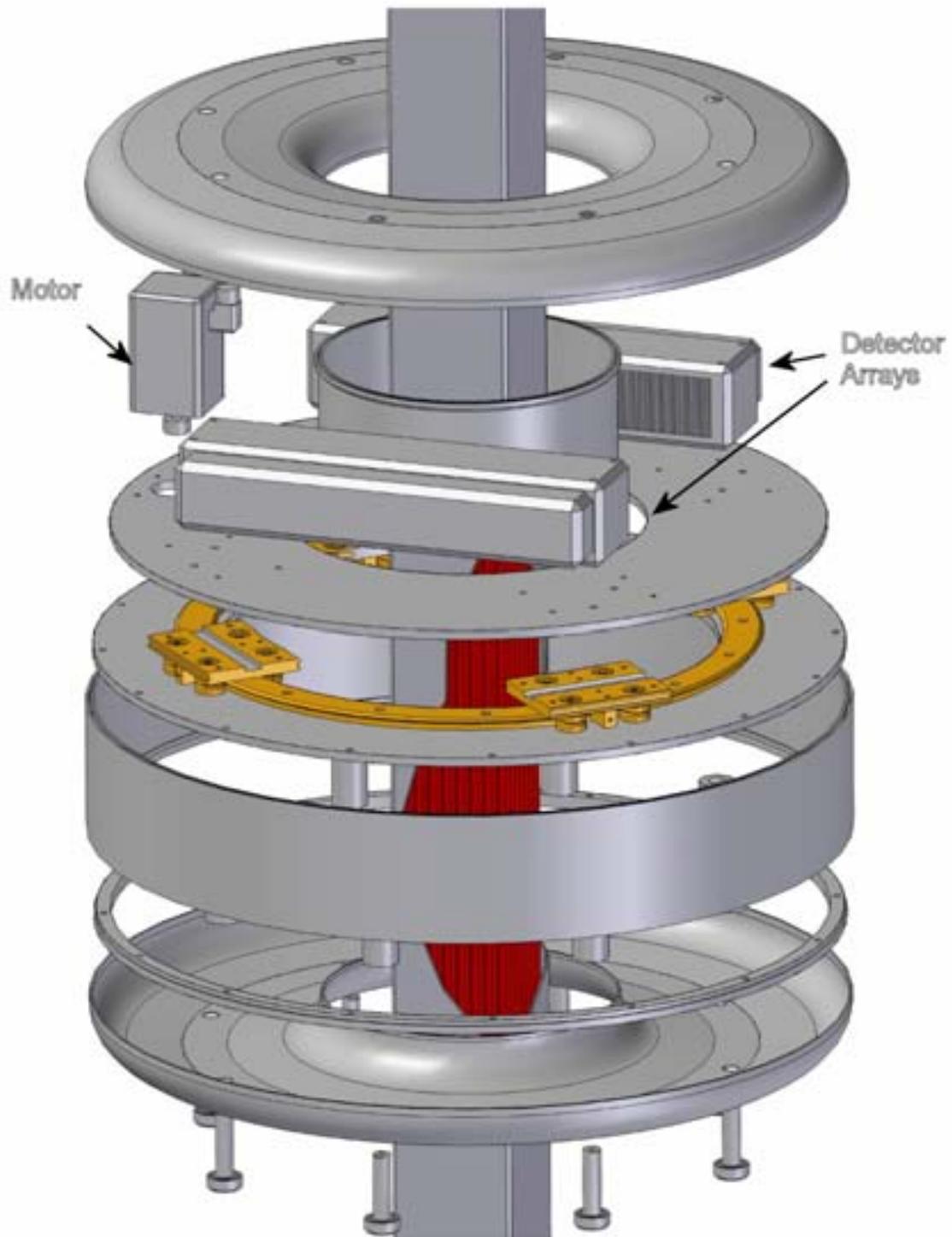


FIG. 1. Exploded view of designed passive emission tomographic verifier. BWR spent fuel is placed inside the toroidal verifier. Cabling is not shown.

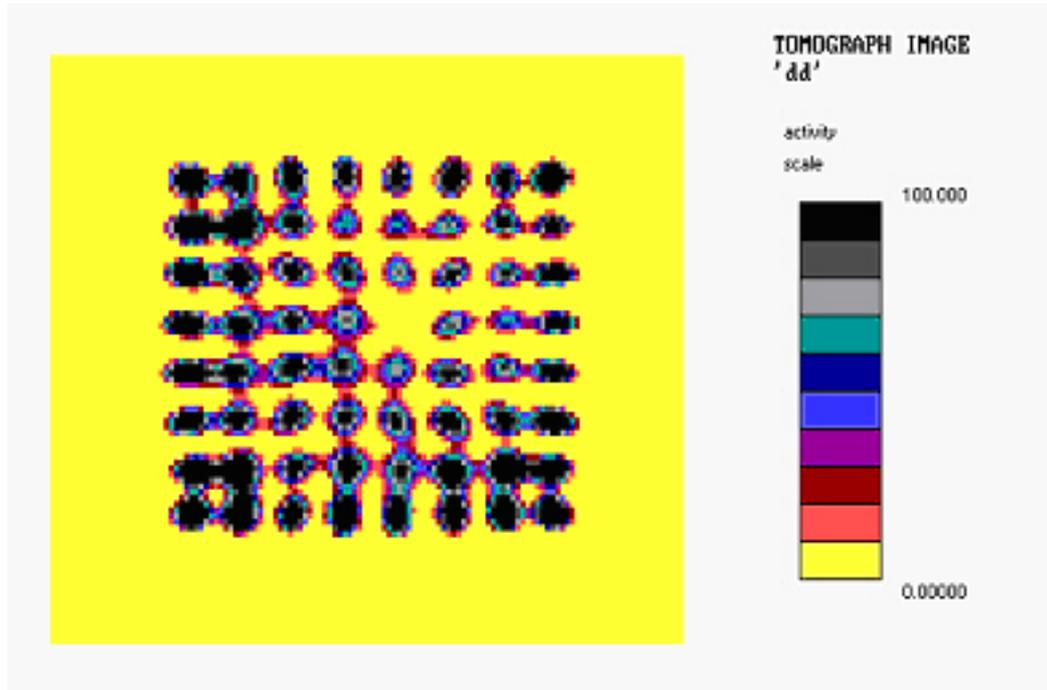


FIG. 2. Calculated image of 8x8 BWR fuel assembly.

Also the development of the software has begun. Part of the software must be written after the delivery of the detector array, because not all communication parameters are known. The requirements for the software are high. It should be user-friendly, but still facilitate all necessary set-ups for maintenance and testing purposes. Data collection, internal motor controls and signal processing will be included in the package. Finally, the software will interpret the image and report the results and store the results and raw data into an internal database. A view of an early test version is provided in Figure 2.

After delivery the detection array and acquisition system will be tested in STUK with a strong Co-60 source. After acceptance testing the device will be assembled in final form. According to the project plan the final testing will take place in late 2007 at Ringhals NPP with spent fuel assemblies.

The construction of the prototype is a big effort. Quality manuals for hardware and software development have been prepared in order to make sure all details will be taken care of. In 2007 the timetable will be tight. By the end of 2007 the device will be ready for the IAEA. The PGET has a potential to become the first device for single rod verification by the IAEA, therefore greatly enhancing IAEA's verification capability.

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Analysis of U-particles by fission track-etch method and quadrupole ICP-MS

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Abstract. Track-etch method in combination with thermal ionisation mass spectrometry is commonly used for the analysis of particles in environmental samples in nuclear safeguards. At VTT track-etch method was used for separation of U-particles from swipe materials and the extracted uranium containing particles were analysed by using ICP-MS after track-etch separation. Procedure was previously used for testing the separation and analysis of uranium containing particles. In this work the method was used to identify U-doped glass particles in IAEA reference samples to find out the reliability of the procedure. The results show that with the precision reached with quadrupole ICP-MS it is possible to separate particles of NU, LEU and HU.

1. Introduction

Track-etch methods are used to identify uranium containing particles of interest for further isotopic analysis by thermal ionisation mass spectrometry. In general, the procedure consists of extracting particles from swipe material by ultrasoneration, collecting of particles in a Collodion (cellulose-nitrate) film or similar on a solid state nuclear detector and irradiation with neutrons, followed by examination of etched neutron induced fission tracks on the detector, and locating and picking of particles under a microscope. The most common etchable solid state nuclear detectors are the plastic detectors CR-39 (polyallyldiglycol carbonate, $C_{12}H_{18}O_7$), Lexan and Makrofol (bisphenol-A polycarbonates, $C_{16}H_{14}O_3$), which are insensitive to light charged particles, X-rays and gamma rays. Various ways to prepare the detector and to locate the particles of interest have been described.

The method developed at VTT use Makrofol as detector. Particles are collected in Collodion and the mixture is spread on the detector to form a thin film. Irradiation is carried out in a Triga Mk II reactor in Otaniemi. Sample preparation was performed earlier by ashing the filter samples and collecting the ashing residue in Collodion [1,2]. This work describes the use of the method to analyse the IAEA glass reference samples. In this work particles were extracted by ultrasonification with ethanol from the swipes and with Collodion. The mixture was spread on Makrofol detectors, where it formed a thin film. Irradiation was carried out in Triga Mk II reactor in Otaniemi. After etching the detectors the fission tracks could be examined under a microscope and the particles were picked up and dissolved for the analysis with ICP-MS.

2. Experimental

2.1. Sample preparation

The samples from the IAEA were received unknown. The five cotton pieces (10x10 cm) were packed in double plastic bags, each of which had a code number.

The samples were cut off the cotton swipes in the clean room. A strip of 2 cm was further cut for two pieces and these pieces (2x5 cm) were placed in tubes with 7 ml of ethanol. The samples in ethanol

were ultrasonerated for 3 minutes. The samples were centrifuged and ethanol was removed carefully with a pipette. Collodion was added so that Collodion to ethanol ratio was one to ten and pipetted on the Makrofol sheets as a thin layer. Next day Collodion in ethanol-mixture (1:1) was added to cover the sheets as a thick layer and were left to dry up overnight.

The sheets were irradiated for one hour in the neutron flux $1.2 \times 10^{12} \text{ cm}^{-2}\text{s}^{-1}$ in the Triga Mark II reactor in Otaniemi, Espoo. The Collodion film was separated from the Makrofol in hot water after marking the specimens. The Makrofol were etched in 6.5 M KOH for 15 min and glued to the microscope slides. The collodion films were placed back on the Makrofol sheet according to the marks.

The fission tracks were examined under a microscope and the Collodion piece with the uranium particle was cut out from the film with a razor knife (Fig. 1). The Collodion pieces were picked up into 0.5 ml polyethylene vials. The vial was filled with acetone to dissolve the Collodion. Acetone was evaporated under an infrared lamp and 7 drops of conc. HNO_3 was added to burn the residue of possible organic material. After the evaporation of HNO_3 the glass pearls were dissolved in five drops of conc. HF, which was evaporated. The same was repeated with five drops of HNO_3 . Then 150 μl of 5% HNO_3 could be added and the sample analysed with ICP-MS.

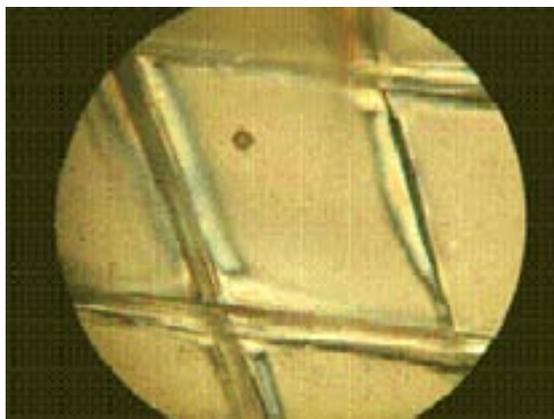


Fig. 1. Separation of the particle.

2.2. ICP-MS-analysis

The analysis of the separated particles was performed by an ICP-MS (VG Plasma Quad 2+) with quadrupole mass separator. The small amount of sample solution with low uranium concentration makes the use of microconcentric nebulizer (Cetac MCN-100) obligatory. By using the natural uptake the sample volume 150 μl was enough for 3 parallel measurements for each sample. To measure the total amount of uranium in particles, uranium standard solutions were measured in each sample series.

The nitric acid used for dissolution of particles is free from uranium and the reagent blank containing acetone and acid contains $<0.5 \text{ pg U}$.

The spectra of separated particles (Figs 2 and 3) show a rather poor statistics. The three parallel measurements of 3 pg particle of 20% enriched uranium (NIST uranium oxide) gave isotopic ratio 0.22 ± 0.04 235/238 when the given value is 0.25125. The analysed 3% enriched particles were bigger. The isotope ratio of 38 pg particle calculated from three parallel measurements was 0.028 ± 0.001 and given the value 0.03143. The small amount of counts in 235 peak is the limiting factor in the analysis.

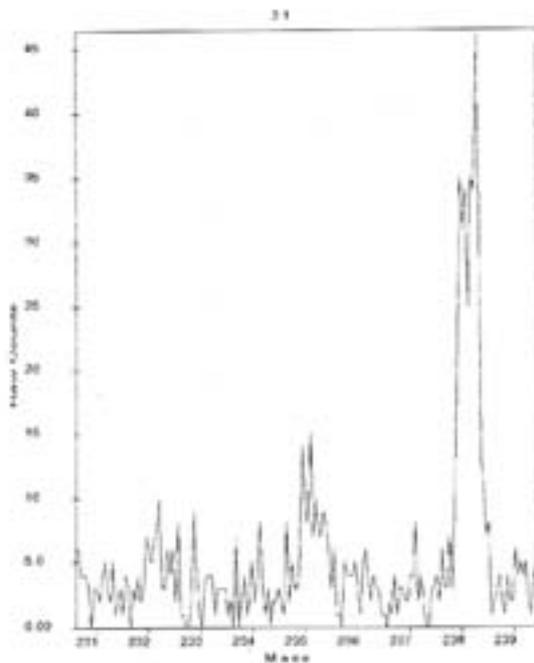


Fig. 2. Mass spectrum of a 3 pg particle of 20% enriched uranium.

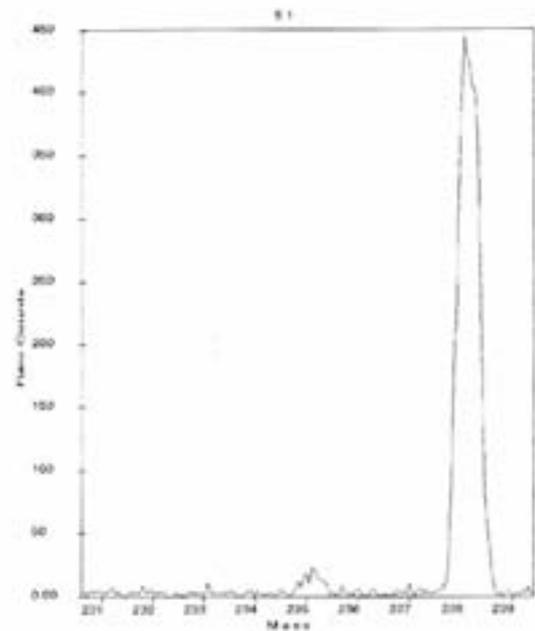


Fig. 3. 38 pg particle of 3% enriched uranium.

3. Results

The results of the analysed particles are presented in the next 5 tables with the standard deviation of three parallel measurements. No bias correction was made in the isotope ratio measurement. The ^{238}U standard was measured in each sample series to calculate the total amount of uranium in the particles. The particles were chosen both according to their track-etch print and the size of the particle. Some of the results have been left out from the list because the ^{235}U - concentration of the particle was below the detection limit. Usually the particles and the track-etch prints were both detectable. In the case where the track-etch print was big but the particle could not be seen the analysed U-ratio was high. (Fig. 4).

The amount of uranium in particles was calculated by comparing with the standard solution. To calculate the diameter, the particles were assumed to be spherical 5% U_3O_8 in glass matrix with the density of 2. The comparison with the micrographs shows that the size of the particles was in most cases bigger than that calculated from the uranium content. It is also unknown, if we were able to dissolve all uranium from the particles.

The track-etch prints of particles and the particles in the Collodion film are shown in the Figs 4 and 5.

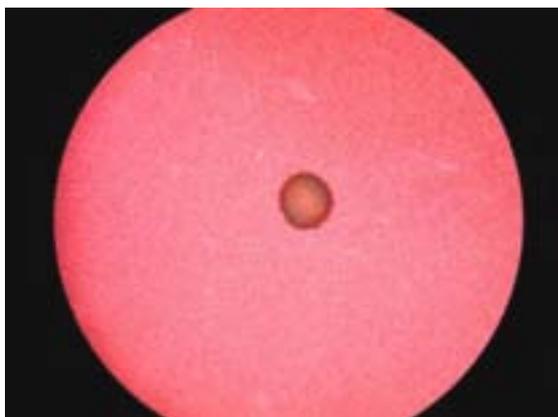


Fig. 4. Track-etch print of the particle 150604/78 in sample 2.

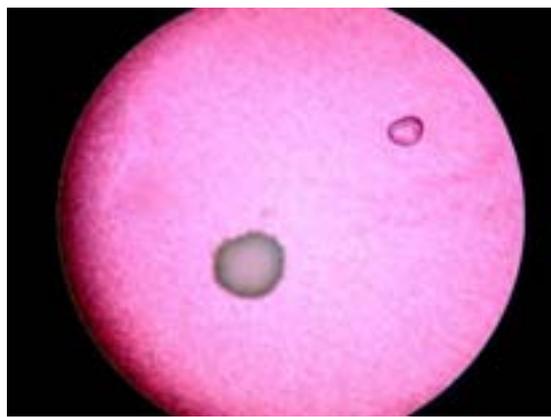


Fig. 5. Track-etch print and particle 150604/75 in sample 1.

The results of analysed particles are presented in Tables 1 to 5.

Table 1. Sample 1

Particle code	tot. U pg	d μm^* calculated	d μm measured	$^{235}\text{U}/^{238}\text{U}$	Stdev
111203/1	73.2	12	12	0.042	0.001
111203/3	5.5	4.7		0.040	0.002
111203/6	47.3	9.7		0.051	0.001
111203/7	27.7	8.1		0.0441	0.0004
111203/4	191	15		0.261	0.001
111203/5	15.8	6.7		0.410	0.012
120504/5	919	26		0.0075	0.0001
120504/7	246	17		0.0075	0.0003
120504/8	253	17		0.0128	0.0002
120504/9	124	13		0.0072	0.0003
140604/62	15.7	6.7	9.1	0.020	0.001
140604/69	291	18		0.011	0.0004
140604/70	4.28	4.3	14	0.089	0.008
140604/71	46.7	9.63		0.0422	0.0002
140604/72	169	15		0.0045	0.0003
150604/75	88.3	12	16	0.0294	0.0008

Table 2. Sample 2

Particle code	tot. U pg	d μm^* calculated	d μm measured	$^{235}\text{U}/^{238}\text{U}$	Stdev
140604/53	13.8	6.4	14	0.035	0.004
140604/54	10.8	5.9		0.039	0.004
140604/56	6.94	5.1	11	0.040	0.005
140604/57	3.70	4.1		0.431	0.026
140604/60	19.0	7.1		0.041	0.004
150604/78	5.75	4.8	11	0.525	0.039
150604/80	110	13	31	0.043	0.001
150604/81	14.1	6.5	11	0.043	0.002

Table 3. Sample 3

Particle code	tot. U pg	d μm^* calculated	d μm measured	$^{235}\text{U}/^{238}\text{U}$	Stdev
111203/8	11.4	6.0		0.045	0.006
111203/9U	30.8	8.4		0.043	0.001
111203/11	7.00	5.1		0.045	0.002
140504/26	58.7	10	19	0.044	0.002
140504/27	3.86	4.2	16	0.055	0.003
140504/28U	6.87	5.1		0.042	0.002
140504/29	26.3	8.0	15	0.044	0.007

Table 4. Sample 4

Particle code	tot. U pg	d μm^* calculated	d μm measured	$^{235}\text{U}/^{238}\text{U}$	Stdev
140504/33	348	18	20	0.0074	0.0002
140504/37	148	14	7.9	0.0084	0.0003
140504/38	381	19	11	0.0073	0.0001
140504/41U	486	21	4.5	0.0074	0.0002
140504/39	11.3	6.0		0.042	0.002
140504/32U	6.80	5.1	6.8	0.038	0.003
140504/34U	19.3	7.2	9.2	0.038	0.001
140504/35U	9.00	5.6	9.1	0.041	0.002

Table 5. Sample 5

Particle code	tot. U pg	d μm^* calculated	d μm measured	$^{235}\text{U}/^{238}\text{U}$	Stdev
140504/43	96.4	12	23	0.0077	0.0006
140504/44	2.75	3.8	5.7	0.037	0.002
140504/45	46.9	9.6	14	0.0064	0.0001
140504/48	2.90	3.8	5.7	0.045	0.004
140504/49	50.2	9.9	14	0.043	0.001
140504/42U	29.1	8.2	16	0.043	0.001

* calculated from the total counts of uranium in ICP-MS measurement

The low background in the mass peak 235, only <5 counts/s, makes it possible to separate rather small particles with different enrichment from each others. For natural uranium, the total amount of uranium in the analysis should be at least 200–250 pg for the isotope ratio measurement. According to these measurements the precision of $\pm 10\%$ could be possible in that case. The theoretical diameter of the particle is about 4 μm .

In some cases the big particle gave just few counts for ^{238}U . Probably the dissolution of the glass particle was only partial. The biggest particle among the picked up ones (Fig. 5.) gave an isotope ratio of 0.029. The isotope ratio 0.04 was found in some particles in all of the samples.

The precision of three parallel measurements of 235/238 ratio was <10% where the number of counts for each isotope was over the detection limit (3 sigma). The total time used for measuring the isotope ratio in one sample was less than 10 min.

The results in the tables were sent to IAEA, because the isotopic ratios of the glass particles were unknown to us at the time of the analyses. The weight ratios of the particles in the 5 samples are presented in the Table 6. The HEU particles have the isotopic ratio ($^{235}\text{U}/^{238}\text{U}$) 0.546 and the LEU particles 0.0433.

Table 6. The weight ratios of the glass particles containing different enrichment factors

Sample number	Weight ratios of particles in samples
1	HEU:LEU 1:10 and no NU
2	HEU:LEU 1:110 and no NU
3	HEU:LEU 1:1010 and no NU
4	HEU:LEU:NU 1:1:220
5	HEU:LEU:NU 1:10:1110

In samples 2 and 3 we found particles that match quite well with the expected ratios, but sample 1 has contamination of NU particles. This contamination probably originates from the laboratory. The ratio of HEU and LEU particles in the samples is good. The analysed particles in the samples 4 and 5 do not reflect the expected ratio but the particle size differences between the standards could lead to quite different particle number ratios. Therefore the results are quite satisfactory.

4. Conclusions

With the precision reached with quadrupole ICP-MS it is possible to separate particles of NU, LEU and HU. The need for more experience in particle separation is obvious. The NU contamination in the samples can be avoided by performing the sample preparation and dissolution of separated particles in a clean room. Also the microscopic work including cutting of the Collodion film should be done in a cleaner environment than in normal laboratory. The results are anyway encouraging and when more particles from each sample were analysed it is possible that the isotopic ratios would be nearer the expected ones.

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Application of airborne gamma spectrometry to the detection of nuclear materials

Example of the French Helinuc[®] system

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Presented by S. Gutierrez

1 ABSTRACT

This document presents detailed information related to the use of AGS (Airborne Gamma Spectrometry) to detect ground contaminations. Based on the technical expertise acquired on the French airborne gamma mapping system Helinuc¹, it focuses on algorithms used to analyse the recorded data. Use of the standard windows method alone (method recommended by IAEA) is not well suited to detection of most nuclear materials because of their low energy signature. Dedicated algorithms have been implemented and tested in the last years by CEA to improve detection performances at low energy. These algorithms are presented and discussed. Examples of application of these algorithms to real data sets are also be presented.

2 INTRODUCTION

The Commissariat à l'Énergie Atomique (French atomic energy commission) has developed an airborne gamma mapping system known as Helinuc. The measurement principle consists in using appropriate means on board an helicopter for simultaneous acquisition of gamma spectra and of the position of the helicopter. The data recorded in flight are processed on the ground using a dedicated computer system, and output is in the form of false-colour maps representing the activity levels.

AGS (Airborne gamma spectrometry) has been recognised as a very powerful tool in case of emergency situation. Recent exercises (RESUME 95, HELGA 2004) and real campaigns such as Georgia in June 2000 [IAEA Project GEO9006-9002] have demonstrated the performances of airborne gamma mapping to detect ground contaminations and to locate lost radioactive sources.

Airborne gamma spectrometry can also be used to detect undeclared nuclear installations, nuclear material storage, or to inspect suspicious areas. Because of its high sensitivity, this technique is well adapted to the detection of small quantities of radioactive materials. Moreover, an airborne survey is not intrusive, and 5 to 10 km²/h can be covered with optimal detection limits. As an example, the Helinuc system was used in 1997 and 1998 by IAEA/Action Team to survey Iraqi suspicious industrial sites.

The increase of new threats that would include nuclear materials has encouraged efforts to optimise their detection by airborne gamma spectrometry.

¹ Helinuc is a registered trade mark.

2 MEASUREMENT PRINCIPLE

The survey of a site consists of a series of gamma measurements recorded along a predefined flight plan. Two types of detectors are used, a NaI crystals pack and two additional Ge detectors. Every 2 seconds, a NaI spectrum is recorded with the mean position (X, Y, Z) of the helicopter during the measurement.

The area to be surveyed is divided into a grid by equidistant flight lines (Figure 1). Line spacing, altitude and speed of the helicopter are defined before the flight according to the required detection limit and the flight time available to cover the area. These parameters are monitored continuously by the operator and the helicopter pilot.

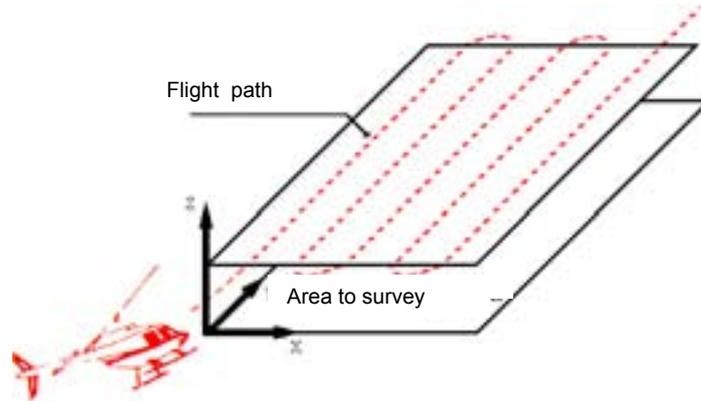


Figure 1 : Coverage of the area to be mapped.

3 EQUIPMENT

The main detector is a 16 litres sodium iodide crystals pack inside a pod fitted under the helicopter. In such a configuration, the gamma signal is unaffected by the floor of the aircraft. The system is described in Table 1 and presented in Figure 2.

The NaI spectra are stored on 512 channels between 40 keV and 2800 keV. NaI measurement is managed by an Exploranium GR-820 airborne spectrometer. The gain is set automatically by continuously monitoring the position of the absorption peak of a natural radioelement, usually potassium or thorium.

The position of the helicopter is given each second by a GPS, which can be used in real-time differential mode to obtain a sub-metric precision. The ground clearance is given by a radioaltimeter with a precision of 1 meter. Data processing is performed by a recording rack inside the cabin. An operator can control the acquisition process and the real-time data-processing.

The system can be installed onboard the helicopter in two hours.

Detection system A	NaI pack of 16 litres
Sample time	2 s
Spectrometer	Exploranium GR-820
Energy range	40 keV to 2800 keV
Positioning system	GPS Trimble, type AG 132
Altimetry system	Radioaltimeter Thomson ERT 011
Hardened PC	Kontron FW 8500

Table 1: System description.

The acquisition process is controlled by the Gr-660 software, developed by the Exploranium company. A preliminary processing of NaI data is done in real time. The results are displayed in different windows. The first window displays the charts of count rate, or activity of specified nuclides, or dose rate. A "rainbow" graph, showing the last fifty spectra is also displayed. In case of suspected presence of

radionuclide, the spectrum can be displayed to identify the anomaly. A navigation screen is located in the flight board in front of the pilot. Theoretical and real flight lines are displayed, as well as the flight parameters (speed, line spacing and ground clearance). At the end of the flight, the data are transferred on a USB memory card for post-processing.



Figure 2: The Helinuc system installed on-board an AS355 helicopter.

4 EXPERIMENTAL CONDITIONS

4.1 FLIGHT PARAMETERS

- Speed : 70 km/h (typical value),
- Altitude : 40 metres (typical value),
- Line spacing between 50 and 500 metres.

4.2 RADIOLOGICAL MEASUREMENT PARAMETERS

- Acquisition time : 2 seconds (as a general rule 2 to 10 seconds),
- Measurement energy range : 40 to 2,800 keV for NaI detectors,
- Automatic energy stabilization.

4.3 GEOGRAPHICAL POSITIONING

The GPS provides geographical coordinates (WGS 84) which are converted for mapping purposes either into the UTM system (cylindrical projection) or the Lambert system (conic projection generally used in France).

Previously-scanned, geographical maps appear as transparent overlays on each map. Two bars in the bottom left-hand corner indicate the map scale in X and Y.

5 DATA PROCESSING

The objective of the data analysis is twofold : i) to find all γ emitting radionuclides, ii) to calculate the activity of each radionuclide (everywhere, if possible). To achieve these tasks, we use several data analysis algorithms. Some of them have been developed at CEA.

Generally the mapping of raw measurements starts the data processing in order to evaluate the main changes in gamma signal. Secondly the mapping of natural radionuclides is performed. It's very helpful to know activities and spatial distribution of natural radionuclides to detect low amount of anthropic radionuclides. Then anthropic contributions are isolated with several analysis methods, depending on the energy range of the gamma emission.

5.1 SPECTRAL ANALYSIS OF NaI DATA

5.1.1 Stripping at low energy

We have developed a method to estimate the low energy background with a higher accuracy. First, a stripping coefficient between the low energy window and a reference window at higher energy is calculated considering the whole data set.

$$S_{LE} = \frac{C[40-80]}{C[200-2800]}$$

This coefficient is then used to calculate the low energy contribution generated by all radionuclides which have gamma lines in the high energy window.

$$C_{LE} = C[40-80] - S_{LE} * C[200-2800]$$

After this stripping, the signal from low energy emitters is enhanced.

5.1.2 IPA

This algorithm called « IPA » (Anomaly Detection Criteria) allows to detect the presence of an anomaly in a gamma spectrum by comparison with the background, thanks to a reference spectrum. This reference can be chosen as an average of all the neighbour measurements of the considered spectrum or as an average of all the measurements from the site.

$$\bar{R} = \frac{\bar{C}_{BE}[40-80]}{\bar{C}_{HE}[200-2800]}$$

The spectrum is divided in energy windows of 100 keV between 40 keV and 1500 keV. For each window, the difference between the count rate of the current spectrum and the count rate of the reference spectrum in the same window is calculated.

The residual count rate is then divided by the standard deviation of the variations in this energy window, calculated on the reference measurements.

$$IPA_k = \frac{R_k - \bar{R}}{\sqrt{\sigma(R)}}$$

This ratio called IPA represents the probability to have an anthropic signature in the spectrum. The higher the IPA, the higher the probability to have an anthropic signature in the spectrum. IPA is calculated for each window and the maximum value is stored for each spectrum.

This method provides a high detection sensitivity but does not allow to precisely identify the radionuclide at the origin of the anomaly. Only the energy window which leads to the highest IPA is identified.

5.2 DETECTION LIMITS

The detection limit associated with any radiation measuring instrument is an essential parameter, since it is used to select the appropriate means of measurement for the activities to be measured. Other criteria, such as the area covered per hour or measurement costs, also influence this choice. Knowledge of the detection limits in airborne gamma spectrometry is particularly important because many parameters are involved in their calculation: energy, sample time, ground clearance, line spacing, radiological background of the site, etc...A detection limit is associated with a set of experimental parameters. Knowledge of the influence of each parameter enables the appropriate flight conditions to be defined to obtain the desired detection limits.

5.2.1 Uniform activities

Calculation of the activity corresponding to the minimum detectable count rate requires an assumption about the distribution of the radionuclides in the ground. The activity detected over the field of view (~30,000 m²) is considered as being locally uniform with respect to area or to volume in the ground.

Table 2 presents the typical detection limits for radionuclides of major interest in AGS. These results are obtained with a sample time of 2 s. A range of activities is given for each radionuclide because the detection limit is strongly dependant on other radionuclides activities.

The detection of small amounts of radioactivity requires accurate knowledge of the detection limits. Consequently, these limits must be calculated for each area surveyed. Depending on the radiological background of the site, the flight parameters (ground clearance and sample time) are chosen considering the detection limit that must be obtained.

Table 2 : Detection limits of uniform activities with a 16 liters NaI detector
(detection probability at 1 σ ; acquisition time = 2 s; altitude = 40 m) - Uniform activities

Radionuclides	Detection limits at 40m	Equivalent mass
²⁴¹ Am	15 - 40 kBq/m ²	~ 3.10 ⁻⁷ g/m ²
²³⁹ Pu	25 - 50 MBq/m ²	~ 20 mg/m ²
Uranium ore	15 - 40 Bq/kg of soil	~ 3 mg/kg of soil
Purified uranium (²³⁸ U)	70 - 120 kBq/m ²	~ 10 g.m ²
²³⁵ U	10 - 20 kBq/m ²	~ 0.3 g.m ²

5.2.2 Point sources

The study of detector response to a point source has enabled calibration of a model for the variation in the count rate as a function of the various parameters that influence the measurements, in particular the position of the detector with respect to the source during the measurement.

The position of the helicopter in comparison with the point source is never known precisely. Consequently the estimated activity is always expressed as a range of activities.

Table 3 : Detection limits of point gamma sources with a 16 liters NaI detector
(detection probability at 1 σ ; acquisition time = 2 s; altitude = 40 m) -(Auto absorption is not considered

Radionuclides	Detection limits at 40m	Equivalent mass
²⁴¹ Am	150 - 250 MBq	~ 2 mg
²³⁹ Pu	350 - 550 GBq	> 250 g
Uranium ore	60 - 100 MBq	> 10 kg
Purified uranium (²³⁸ U)	900 - 1500 MBq	> 100 kg
²³⁵ U	100 - 250 MBq	> 3 kg

helpful to confirm the detection. In the surroundings of nuclear sites or areas of interest, baseline surveys are also very helpful to estimate any change of the radioactivity.

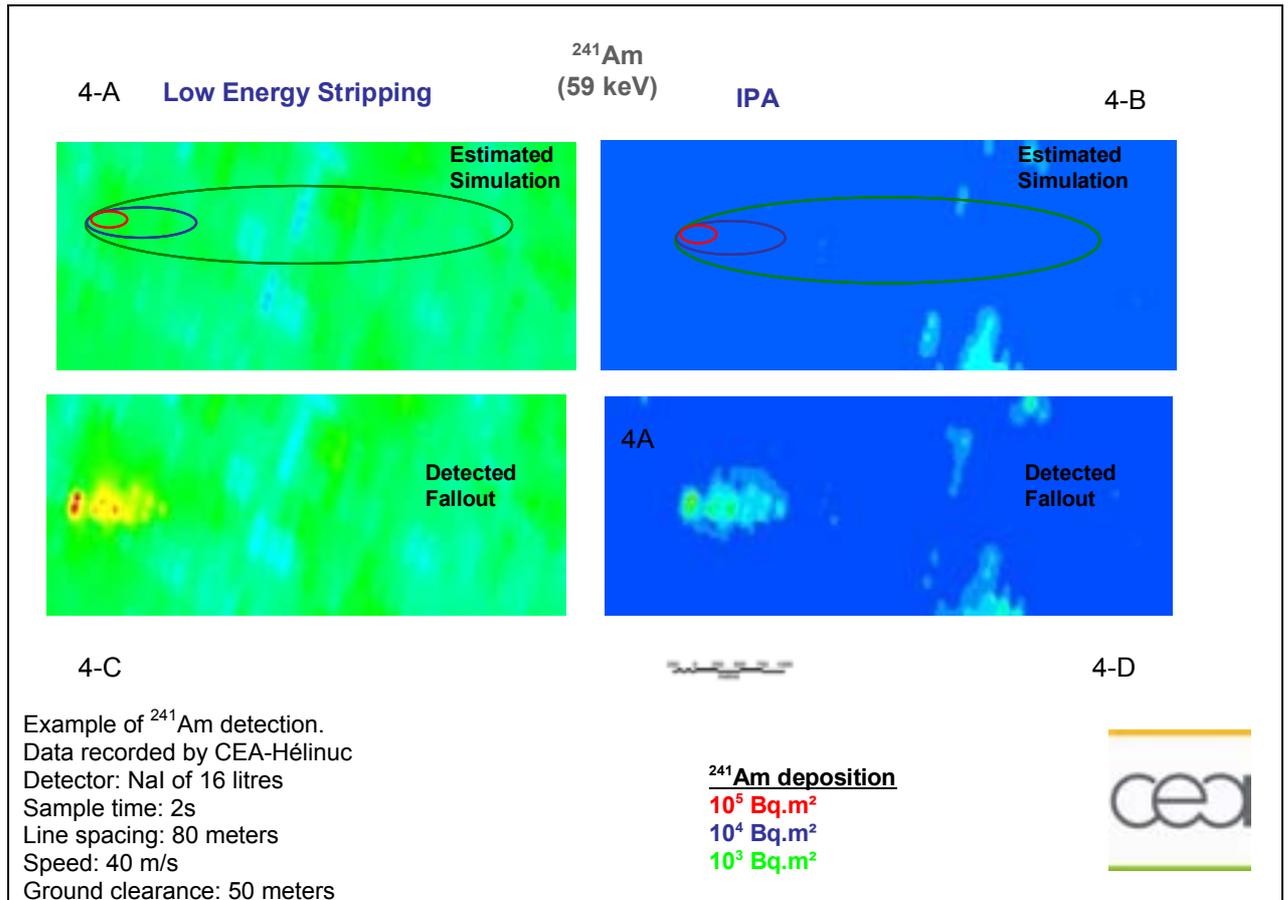


Figure 4 : Detection of ²⁴¹Am - Low Energy Stripping and IPA cross-compared.

7 CONCLUSION

The detection of nuclear materials by AGS is a difficult task, because of their low energy gamma rays. Consequently, the airborne systems should be designed to optimize the sensitivity at low energy. The use of a pod fitted under the aircraft should be recommended to reduce attenuation of low energy gamma rays.

The processing methods currently used in AGS are not well suited to the detection of low energies. Two methods providing a good sensitivity have been described here. The results should nevertheless be interpreted carefully, because local background variations may generate false detections.

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Investigating the applicability of anions as indicators for verification of consistency of declarations*

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Presented by K. Mayer

Abstract. Nuclear material is subjected to chemical processing throughout the entire fuel cycle. Traces of the chemical reagents and solvents are left behind in the nuclear material. So far essentially metallic impurities or light elements have been investigated for their potential in providing clues on the type of process they originate from.

In the present investigation, the applicability of anions for attributing nuclear material to a certain chemical process has been investigated. Anions (e.g. nitrate, sulphate, phosphate, chloride) originate from acids or salt solutions that are used for processing of solutions containing uranium or plutonium. The study presented in this paper focuses on yellow cake samples originating from different mines applying different chemical processes for leaching, dissolving and precipitating the uranium. Consequently, the anionic patterns should be different. The concentrations of different anionic species were measured by ion chromatography using conductivity detection.

First, basic studies on possible interferences were carried out, e.g. testing the effect of a large excess of nitrate on the signal of small amounts of other anions. Then the linearity and the stability of the calibration in the presence of high concentrations of cations, such as sodium and calcium were checked. Finally, samples of yellow cake were investigated. Particular attention was paid to reagent blanks and to the repeatability of leaching prior to measurement of the anion concentrations.

A comparative evaluation of the results will be given and the potential for application in safeguards and in nuclear forensics will be discussed.

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

Improvement of non-destructive radioisotope search devices of smuggling by application of switchable neutron source Am-LiF

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In devices of neutron nondestructive testing traditional isotope sources of the neutrons including a radioactive isotope - a source of alpha radiation and a target constantly generating neutrons, are used basically. In the equipment where sources of neutrons are used, it is desirable to have such source which can "be switched off" for that time when the device does not work. It is especially important to have such source in the portable equipment as for safe storage and transportation of an isotope source of neutrons with an output 105-106 n/c, it shall be placed in the protective container in weight in tens kg.

Detector "CINDI" (Compact Integrated Narcotics Detection Instrument) developed by the American enterprise "NOVA R&D, Inc." 2001) is the characteristic representative of the portable neutron devices intended for detection of illicit hidden places behind barrier. It contains a source of fast neutrons of Californium – 252 having activity about 50 μCi .



In 1995 in Georgia applicants of the given project had been developed, tested and introduced into customs department of the country portable neutron search detector explosive

substances, narcotic and other illicit substances hidden in the thick-walled bookmarks of different vehicles, containers, etc. It also contained Californium – 252 having activity no more than 50 μCi , Helium counters were used as the detector of the reflected slow neutrons.



Testing and operation of device has shown that it reliably controls metal thick-walled (1,5-2,0 cm and more) eaves, doors, wheels, the chassis cargo and cars, deaf eaves and edges of rigidity of containers used as hidden places for illegal materials – explosives, drugs etc. having minimal weight 10-20 g, located on depth about 25 cm. It could be effectively used for detection and control of whatever hidden places by relevant services



A working surface of a sensor control of the device made 6,0x10,0 cm, and the weight did not exceed 350 g. Application of the telescopic handle of the device provided radiating safety of the operator.

Our enterprise made simpler updating of this development, level indicator, and successfully sold it to the different petrochemical enterprises of the former Soviet Union, had preliminary orders on hundreds of neutron level gauges.



However the certain inconveniences and complexities at storage and transportation of neutron sources because of high weight of containers, as well as radio phobia, push away consumers.

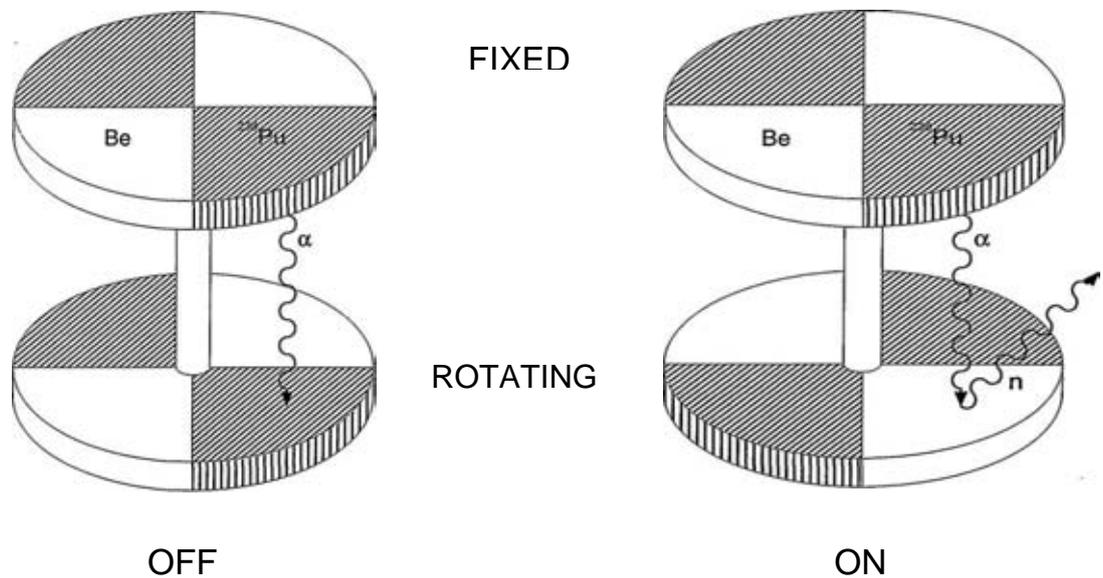
The wide application of above mentioned devices confirms this consideration. 6000 BUSTER K910 based on gamma-radiation is used in more than 50 countries, while few specimens of neutron devices, despite their significantly higher technical characteristics, are used for the same purposes.

The real prospect for safe, wide consumption of our as well as of many other neutron development represents transition on qualitatively new, radiation safe radioisotope sources of neutrons with an regulated output.

In the scientific literature the sources of this kind are named differently – radioisotope generators of the neutrons, adjustable sources of neutrons, Switchable Neutron Sources (SNS), "ON-OFF", etc. As a matter of fact they are included and switched off mechanically and are radiation safe at storage and transportation. Development of manufacture and duplicating of such sources of neutron radiation will give a new pulse for perfection of neutron methods of not destroying testing.

In SNS it is necessary to result in working position in contact of substance of an alpha emitter and a target so that in switched off position they could be divided again, not having polluted one another. Generating of neutrons from sources with a firm, flat target can be changed through introduction of plates of a target between two plates covered with a layer of the chosen alpha emitter that could be done by shifting them relative to each other

Schematic of “on – off” operation



The output of neutrons of such sources basically depends on the area of contact of a target and an alpha emitter, and dimensions (volume) of SNS will definitely depend on the set activity (output) of neutron radiation. Except for that specific activity of alpha emitters and their nuclear characteristics, the different probability of (α, n) reactions on targets accordingly influence size of the necessary contact area of choose pairs - targets and an alpha emitter.

The negative characteristics of these constructions could be their overall dimensions and technological complexity of maintenance of his tightness and reliability.

Title Advanced concept proof-of-principle demonstration: Switchable radioactive neutron source

Creator/Author [Rhodes, E.A.](#) ; [Bowers, D.L.](#) ; [Boyar, R.E.](#) ; [Dickerman, C.E.](#)

Publication Date 1995 Oct 01

OSTI Identifier OSTI ID: 197135 ; DE96004828

Report Number(s) ANL/ACTV--95/2

DOE Contract Number W-31109-ENG-38

Resource Type Technical Report

Resource Relation PBD: Oct 1995

Research Org Argonne National Lab., IL (United States)

Sponsoring Org USDOE, Washington, DC (United States)

Subject 07 ISOTOPE AND RADIATION SOURCE TECHNOLOGY ;44
INSTRUMENTATION, INCLUDING NUCLEAR AND PARTICLE
DETECTORS ; NEUTRON SOURCES; FEASIBILITY STUDIES;
RADIATION DETECTORS; ARMS CONTROL; VERIFICATION;
PORTABLE EQUIPMENT; FISSILE MATERIALS; DETECTION;
DRUGS; EXPLOSIVES; PLUTONIUM 238; BERYLLIUM

Description/Abstract An advanced concept proof-of-principle demonstration was successfully performed to show the feasibility of a practical switchable radioactive neutron source (SRNS) that can be switched on and off like an accelerator, but without requiring accelerator equipment such as high voltage supply, control unit, etc. This source concept would provide a highly portable neutron source for field radiation measurement applications. Such a source would require minimal, if any, shielding when not in use. The SRNS, previously patented by Argonne staff, provides a means of constructing the alpha-emitting and light-element components of a radioactive neutron source, in such a fashion that these two components can be brought together to turn the source "on" and then be separated to turn the source "off". An SRNS could be used for such field applications as active neutron interrogation of objects to detect fissile materials or to measure their concentration; and to excite gamma-ray emission for detection of specific elements that indicate toxic chemicals, drugs, explosives, etc. The demonstration was performed using Pu-238 as the alpha emitter and Be as the light element, in an air-atmosphere glovebox having no atmosphere purification capability. A stable, thin film of Pu-238 oxide was deposited on a stainless steel planchet. The "on" output of the demonstration Pu-238 film was measured to be 2.5×10^6 neutrons/sec-gram of Pu-238. The measured "off" neutron rate was satisfactory, only about 5% of the "on" output, after two weeks of exposure to the glovebox atmosphere. After several weeks additional exposure, the "off" rate had increased to about 15%. This work demonstrates the feasibility of constructing practical, highly portable SRNS units with very low gamma-ray dose in the "off" position.

Country of Publication United States

Language English

Format 23 p. ; PL:

Availability INIS; OSTI as DE96004828

In the framework of the project the working breadboard model of design Switchable Neutron Sources (SNS) coordinated and optimized with the detector back absent-minded

neutrons and on his basis the measuring block of the new portable detector of Illicit Materials (Explosives, Drugs etc.) will be developed.

It is natural, that all previous neutron developments of small-sized devices have been optimized on application so-called dot, small-sized traditional sources of neutron radiation. By development of measuring block of the new device it will be necessary to investigate and optimize its new geometry (architecture), taking into account real dimensions (volume) of SNS.

Observation of radionuclide devices as well as experience gained in development, production and marketing of this equipment confirms that radiation safety is the main characteristic among others, determining demand for this equipment.

Real perspective for safe and wide application of nuclear devices could be found through development of substantially new radiation safe switchable neutron sources -SNS.

Constructed working models of SNS will be delivered to IAEA in accordance with the CRP №12591: “Development and Demonstration of the New Methods for the Detection of Shielded Highly Enriched Uranium (HEU)”, Part of Co-ordinated Research Project: Improvement of technical measures to detect and respond to illicit trafficking of nuclear material and other radioactive materials and CRP №13501: “Development of a design of a radioisotope switchable neutron source and new portable detector of smuggling”, part of Co-ordinated Project: Neutron based techniques for the detection of illicit materials and explosives.

Determination of uranium by the Brazilian Safeguards Laboratory – LASAL using “Davies & Gray/NBL” potentiometric method

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Presented by L. Souza Dunley

Abstract. Independent determination of uranium content in nuclear materials to verify operator's declaration is an important tool in the safeguards approaches applied to nuclear installations. This determination may be performed by non-destructive methods, generally performed in the field using radiation detection systems, or destructive methods by chemical analysis, when more accurate results are needed. In Brazil, samples are taken during safeguards inspections and sent to Safeguards Laboratory-LASAL of the Brazilian Nuclear Energy Commission-CNEN, where they are currently analyzed. LASAL also receives samples taken in Argentina by Brazilian – Argentine Agency for Accounting and Control of Nuclear Materials – ABACC to be also analyzed. To achieve the requirements of the Brazilian State System of Accounting and Control of Nuclear Materials – SSAC, LASAL has been applying the “Davies & Gray/NBL” method for Potentiometric determination of total uranium concentration in several forms of nuclear materials since 1984. In order to improve the accuracy and the repeatability, the method uses as titrant, standard reference potassium dichromate NIST SRM 136e and the results are also corrected for bias with uranium metal assay standard NBL CRM 112A. The samples aliquots are analyzed using a microprocessor – controlled Mettler DL-67 Potentiometric titrator attached to a Pt:Rh (90:10) as indicator electrode and a mercurous sulfate as the reference electrode. As an example, this work describes and discusses the results of analysis of a number of ten aliquots of uranyl nitrate sample provided by *Commissariat A`L'energie Atomique* of France (CEA) as part of an intercomparison program. To bias correction of the results and evaluate them, uranium metal standard NBL CRM 112A was used. The main contributions for uncertainty of the method were also studied in order to evaluate the uncertainty of the result obtained for this analysis.

1. INTRODUCTION

In the scope of the Brazilian SSAC, one significant activity is the independent verification of nuclear materials by non-destructive and destructive analysis in order to verify operators' declarations as well as their measurement systems. For destructive analysis, LASAL/CNEN has been using Davies & Gray/NBL method for determination of total uranium concentration in several kinds of nuclear materials since 1984, providing the SSAC with accurate and repeatable results [1,2,5]. In addition, LASAL has also been participating in international intercomparison programs, sponsored by the several international organizations such as “Agência Brasileiro-Argentina de Contabilidade e Controle de Materiais Nucleares - ABACC”, “Direction de L'Energie Nucleaire – CEA” and “New Brunswick Laboratory – NBL/US-DOE”[3,6,7].

This paper describes the method including the reagents used to prepare the samples, its chemical reactions as well as the instruments and equipment to perform the analysis. To illustrate the description of the method and its performance a run of ten aliquots from a sample of uranyl nitrate provided by CEA, analyzed at the laboratory in 2005, is described and their results are discussed [7]. In order to achieve the 2000 ESARDA's TARGET VALUES [10], the relative standard deviation must be equal or less than 0,1% and the

correction for systematic errors is made by a bias factor (B) obtained from a metallic uranium reference standard material NBL CRM 112A.

Usually, a number of ten aliquots of a standard solution of this material is analyzed and a mean bias factor is calculated.

2. DESCRIPTION OF THE METHOD

Potentiometric Davies & Gray/NBL [1,2,3] method for determination of uranium is a selective method based on the reduction of U (VI) to U (IV) in a concentrated solution of phosphoric acid by excess of Fe (II) in sulfamic acid media. The excess of ferrous ions is oxidized with nitric acid in the presence of Mo (VI) and sulfamic acid. Then U (IV) is titrated with a standard solution of $K_2Cr_2O_7$ until a preset end point potential of 130 mV. To precise this potentiometric end-point, vanadyl sulfate is also added.

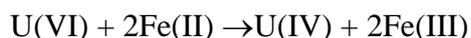
In this method it is essential to avoid any reoxidation of U (IV) before the titration. Thus, reactions kinetic times and the time elapsed between the preparation of the aliquot and the titration must be rigorously controlled. This time shall not be longer than 5 minutes to prevent reoxidation of U (IV) by atmospheric oxygen. Besides, the presence of As (III), Sb (III), halides and organic material should also be avoided to prevent positive biases [4]. Another important feature to assure the good performance of the method is the range of uranium concentration, which must be kept between 90 and 125 mgU/g of solution.

For the analysis of the aliquots, the laboratory uses a wire of Pt:Rh (90:10) as indicator electrode and a mercurous sulfate one as reference electrode [2], attached to a microprocessor-controlled analytical titrator Mettler DL-67. In order to avoid any calibration error from volumetric flasks and pipettes, this method is performed in a weight basis. Consequently, the burette is also calibrated in a weight basis by a factor F_B that converts the volume in milliliters of potassium dichromate dispensed, in grams of the titrant.

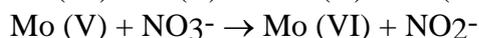
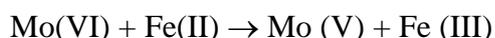
This method may be applied for uranium analysis of oxides, nitrates, ADU, AUC of any physical form [5].

2.1 The main chemical reactions are the following:

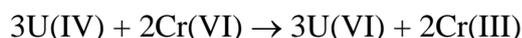
Reduction:



Reoxidation:



Titration:



3.EXPERIMENTAL

3.1 Potassium dichromate standard solution preparation

The first important step was the preparation of the standard titrant solution using potassium dichromate NIST SRM 136e. Firstly, $K_2Cr_2O_7$ crystals were dried in a drying oven for some hours to remove any moisture it may have, before weighing [2,4,11]. The titer of the solution was determined by the expression:

$$T_d = \frac{w \cdot 3A \cdot P}{W \cdot M \cdot 100} \frac{1 - \frac{\rho_a}{\rho_c}}{1 - \frac{\rho_a}{\rho_c}} \quad (1)$$

Where the input variables for this preparation were:

w= weigh of $K_2Cr_2O_7$ crystals = **4,9141g**

W= weight of $K_2Cr_2O_7$ solution = **1,99714 kg**;

A= atomic weigh of uranium = **238.0289g/atg**;

P= oxidimetric assay of $K_2Cr_2O_7$, CRM 136e (**99,984±0,010**)%;

M= molecular weight of standard $K_2Cr_2O_7$, **294,1844g/mol**;

a= specific weight of air, aprox. **1.2. 10⁻³ g/cm³** at 20 – 21°C

d= specific weight of the balance weights, aprox. **8,00 g/cm³** at 20 – 21°C;

dc= specific weight of $K_2Cr_2O_7$, aprox. **2,69 g/cm³** at 20 – 21°C.

The buoyancy correction factor, $\frac{1 - \frac{\rho_a}{\rho_c}}{1 - \frac{\rho_a}{\rho_c}}$ for the conditions of day is **C1= 1,000296229**.

Thus, the titer for this solution was:

Td = Titer of $K_2Cr_2O_7$ solution = 5.97346 mg of U/g of solution

3.2 Calibration of the burette

The burette was calibrated by dispensing 20 ml of the titrant solution into 6 previous tared plastic flasks. The weight of $K_2Cr_2O_7$ in each flask is determined and a relation between the weight and the volume is obtained. After outlier tests a mean factor is determined as the Burette factor. The relative standard deviation of this calibration must be $\leq 0.022\%$, in order to achieve the entire repeatability of the method that is 0.1% [3].

For this determination the burette calibration provided a factor of:

F_B = 0.99911g/ml; and a the relative standard deviation, **RSD = 0.005%**

3.3 Uranium concentration determination

The uranium concentration of each aliquot were calculated by the expression:

$$C = \frac{V \cdot F \cdot T}{m} \quad (2)$$

Where:

C=concentration of uranium in each aliquot, expressed in mgU/gsol;

V=volume of the titrant, in ml;

F=burette factor g/ml of titrant;

T=titer of the titrant, in mgU/g titrant solution;

m=weight of the aliquot, in grams.

4. ANALYSIS AND RESULTS

The sample analyzed was a solution of pure uranyl nitrate provided by *Commissariat A`L`energie Atomique* (CEA) of France [7]. The solution concentration was adjusted to be in the range recommended for this determination, e g between 90 and 125 mg/g. The weight of the sample of uranyl nitrate solution was **13.4298g** and the adjusted solution was **28.1770g**.

After the adjustment, aliquots of about 1 g were taken and treated with 1:1 sulfuric acid to white fumes in order to eliminate organic and halides interferences. After that, water and ortho-phosphoric acid were added before the reduction step with Fe (II). The excess of Fe (II) was then reoxidized by the addition of nitric and sulfamic acid in the presence of Mo (V). Finally, vanadyl sulfate was added and each aliquot were titrated with the standard $K_2Cr_2O_7$ solution previously prepared.

4.3 Results

Table 1 shows the results of the analysis of ten aliquots taken from the sample. They were obtained at the same analytical conditions such as room temperature, relative humidity, barometric pressure and using the same analytical balances.

Table 1. Results of uranium concentration obtained under the same conditions for aliquots from the same sample.

Ident.	Weight of aliquot(g)	Titrant volume (ml)	Burette factor (g/ml)	Uranium concentration (mg/g)
1-01	1.2161	22.2643	0.99911	229.2473
1-02	1.2194	22.3241	0.99911	229.2410
1-03	1.2197	22.3319	0.99911	229.2647
1-04	1.2160	22.2726	0.99911	229.3516
1-05	1.2212	22.3693	0.99911	229.3665
1-06	1.2193	22.3384	0.99911	229.4066
1-07	1.2206	22.3564	0.99911	229.3469
1-08	1.2180	22.3028	0.99911	229.2855
1-09	1.2175	22.3102	0.99911	229.4557
1-10	1.2163	22.2727	0.99911	229.2961

Outlier tests of Nalimov and Grubbs [9] were applied to the results and showed that the aliquot 1-09 was an outlier.

Statistical analysis have shown the following values:

- Mean: 229.3118mg/g;
- Std.deviation: 0.05823mg/g;
- Coefficient of variation: 0.02539%

According to the statistic criteria for this method, any aliquot analysis result should be compared to the others of the same sample. The relative error of the results among them should be less or equal to 0.14% relative. Results lying in these limits are accepted and reported. These values are according to the international target values for random and systematic components associated to safeguards measurements [10].

The results showed in Table 1, considering the nine aliquots of the sample analyzed, are according to the criteria described above.

4.2 Mean bias factor

This factor is obtained by comparing the results of analysis of a number of aliquots from a standard uranium material and its certified assay. For this determination nine aliquots of uranium metal assay standard NBL CRM 112A solution previously prepared, were analyzed at the same conditions and yielded the following mean bias correction factor: **B= 0.99955**.

4.3 Bias correction

Making the correction for bias with the factor B of 0.99955, calculated before, the final result obtained was:

(229,21±0,12)mgU/g solution.

While the reference value provided by (CEA) for this solution was:

(229.11±0.23) mgU/g solution.

4.3 Uncertainty

The uncertainty of ± 0.12 was calculated at a confidence interval of 95%, $k=2$ [7,9], taking into account the most important contributions of the procedure used by LASAL. Firstly, the parameters that can interfere with the uncertainty were studied. In order to evaluate their contribution in the final uncertainty, the uncertainties of all steps of the analysis method were collected and compiled. Then their sensitivity coefficients were calculated.

Finally, each of the following steps had its contribution in the uncertainty determined and the combined uncertainty was calculated:

- Burette factor;
- Weighing of the sample and the adjusted solution;
- Weighing the aliquots
- All steps of the determination of the titer of $K_2Cr_2O_7$ solution;
- The certificates of the standards and the calibration of the balances used;
- Sample titration;
- Statistic contribution;
- Bias correction factor.

As mentioned before, the expanded uncertainty was calculated at a confidence interval of 95% and a coverage factor of (k) of 2.

Comparing all the contributions of the steps mentioned in the final result the statistic of the titration of the nine aliquots analyzed had the most significant contribution in the uncertainty.

5. CONCLUSION

Destructive assay is a very important tool as part of any system for accounting for and control of nuclear materials. It enables the IAEA as well as any State Regulatory Commission to verify the operator's declaration in order to check its measurement system by the comparison of the analysis results and its uncertainties and also, the verification of bias defects. Therefore, accurate and repeatable analytical methods and the use of appropriated standards in the uranium analysis are essential to provide the safeguards system with reliable analytical results.

The example presented in this paper was just to illustrate the typical results currently obtained by LASAL in destructive analysis. In fact, since 1984 LASAL has been supporting the Brazilian State System of Control and Accounting for with destructive analysis of nuclear materials using Davies & Gray/NBL Potentiometric Titration Method [2,11] and contributing for its confidence.

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Simulation and test of spent fuel verification system for CANDU dry storage canister

A feasibility study

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Abstract. Gross neutron counting methodology was proposed to verify the CANDU spent fuel canister. The detailed MCNP model of the Korean Wolsong canister was constructed in order to evaluate the effectiveness of the neutron detection and to give the performance evaluation of the detection system. Prototype passive neutron detection systems were fabricated using He-3 and BF₃ detectors. The performance test was carried out using AmF neutron source and a actual spent fuel canister. The main goal of this study is to find an effective way of the CANDU spent fuel verification methodology. From the simulation results, it was identified that the neutron counting provides much more information of spent fuels in a canister than a gamma counting. The simulated neutron fingerprints were generated and it will be compared to the actual measurement result. In the future, the stability and reproducibility test of the measurement systems will be performed in order to identify the validity of the neutron counting for the spent fuel verification.

1. Introduction

Improvement of spent fuel verification technique and instrumentation is one of the significant challenge in the safeguards field. The CANDU spent fuels stored in the storage pond must be transported to dry storage for the long term storage. A dry storage canister is loaded with 9 baskets and each basket contains 60 spent fuel bundles. This corresponds to 34.9 kg of plutonium (1 SQ = 8kg for Pu) per each canister[1]. After the canister is fully loaded, “gamma-ray fingerprinting” is currently being implemented to compare a pre-established baseline measurement against subsequent measurements in particular case of a Containment/Surveillance failure. However, IAEA still lacks effective re-verification method for spent fuel dry storage.

Neutron detection has been proposed to use as verification and re-verification methods for spent fuel dry storage. A neutron detection system can provide for better visibility for the canister contents than a gamma ray detection system because the neutron has good penetration in the high density, high Z materials of reactor fuel comparing to the gamma ray characteristic. In this study, detailed MCNP model of the canister was constructed and the neutron detection system was designed.

2. Simulation

2.1 Source term calculation

CANDU spent fuel source term was calculated using ORIGEN-ARP[2]. The reference spent fuel was typical CANDU spent fuel of 7500 MWd/MTU burn up and 10 years cooling time. The total neutron production rate per metric ton is calculated to be 3.13×10^6 n/s and total gamma ray intensity is estimated to be 1.2×10^6 . Dominant source of neutrons are generated from the spontaneous fission of ²⁴⁴Cm and ²⁴⁰Pu and from (α ,n) reactions in the fuel.

2.2 MCNP calculation

2.2.1 Modeling of CANDU spent fuel canister

The MCNPX code[3] used to calculate the signal contribution of individual bundles in a canister and to generate the simulated neutron fingerprint. The detailed MCNP model was constructed consisting of concrete structure, 9 fuel basket and 540 fuel bundles. The individual bundles were modelled as a homogenized mixture of uranium, oxygen, and zirconium. The baskets are composed of stainless steel with a side wall thickness of 0.953 cm, the bottom plate thickness of 1.906 cm and the top cover plate of 0.953 cm thickness. The basket height from bottom to top plate is 53.8 cm and the out diameter of the basket is 106.7 cm. There is the overhang on the top cover plate and the height is 1.905 cm. The height of the canister is 6.4 m and the diameter is 3 m. A main storage cylinder, which is 0.35 cm thick stainless steel liner rings, is placed in the center of the canister. The inner diameter of the cylinder is 111.7 cm. Two re-verification tube are placed on opposite sides of the cask at distance of 73.66 cm from the central axis of the canister. Figure 1 shows side view of CANDU spent fuel canister. There is a bend in the re-verification tube and the distance from the central axis to the tube becomes 120.89 cm on the top of canister.

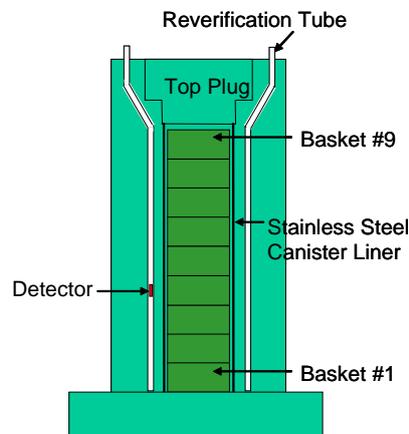
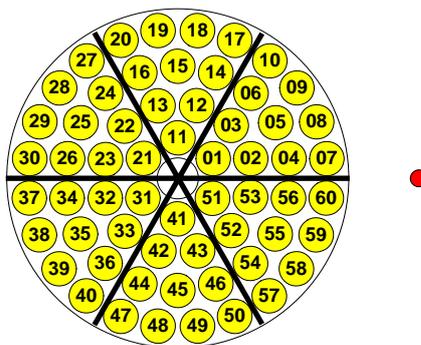


Figure 1. Side view of CANDU spent fuel canister.

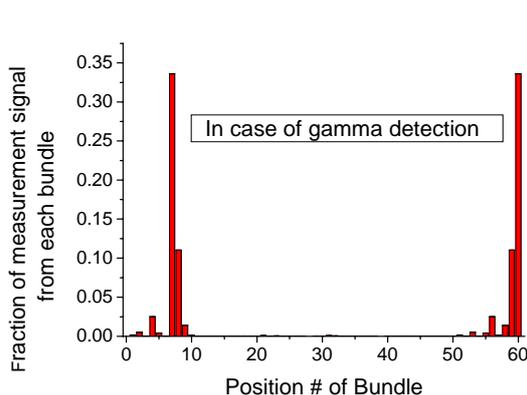
2.2.2 Evaluation of signal contribution from individual bundles

Contributed fraction of the measurement signal from each bundle in a basket for a given detector was simulated using MCNP code. The detector is positioned at the center of a specific basket in the re-verification tube. The source terms are activated for the nearest basket only, therefore the result does not reflect the effect from other baskets. In regard to neutrons, an expected count rate (count/s) for He-3 tube (7.5 atm, 1 inch diameter and 2 inch active length) from each bundle was calculated, whereas in regard to gamma rays, an expected dose rate (R/h) on a detector was calculated. In the calculation, the gamma detector was similar in size and shape to the He-3 tube and located at the same position in re-verification tube as the neutron detector.

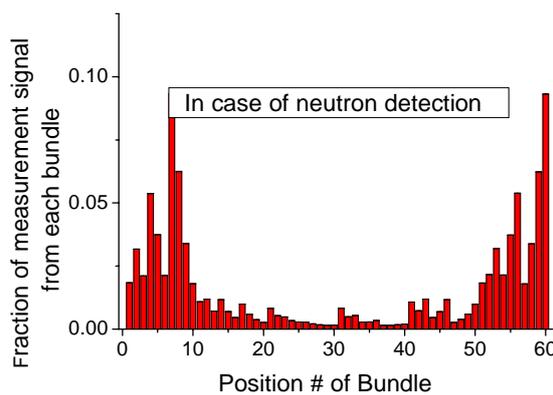
Figure 2 shows fraction of gamma and neutron signal from each bundle in a basket. The results show neutron detection can provide much more information of the canister contents than gamma-ray detection. The bundle number which contributes to the signal over 1% is 26 bundles for neutron detection while 8 bundles for gamma. By measuring neutron signal from two re-verification tubes, neutrons generated from most of bundles in a basket can be detected. Figure 2 (a) shows bundles array in a spent fuel basket. Fig. 2(b) and (c) show fraction of gamma and neutron measurement signal as a function of bundle position within the basket.



(a)



(b)



(c)

Figure 2.(a) Bundles array in spent fuel basket, (b) Fraction of gamma signal induced by an individual bundle, (c) Fraction of neutron signal induced by an individual bundle.

2.2.3 Simulated neutron fingerprints

Vertical profile of neutron detection was calculated. Figure 3 shows the calculated neutron count rate from He-3 tube (7.5 atm, 1 inch diameter and 2 inch active length) at different position in a re-verification tube. Figure 3(a) is a calculated vertical profile where the 9 baskets are placed in the canister. Figure 3(b) shows a calculation result in case one of nine baskets is supposed as a dummy. As expected, the neutron profile does not show the peaks as 9 basket positions due to the good penetration of neutron in the high Z material. However, the existence of the 9 baskets might be identified by comparison the profiles between calculation and measurement results. The signal contribution from the nearest basket was estimated to 54 %, and 46% signals comes from other baskets.

A sharp peak appeared at the end of 9th basket position. This peak is likely due to 6 cm air space in the model between the top of 9th basket and the plug. There might be an increase of the neutron flux due to the lots of scattered neutrons and absence of shield material.

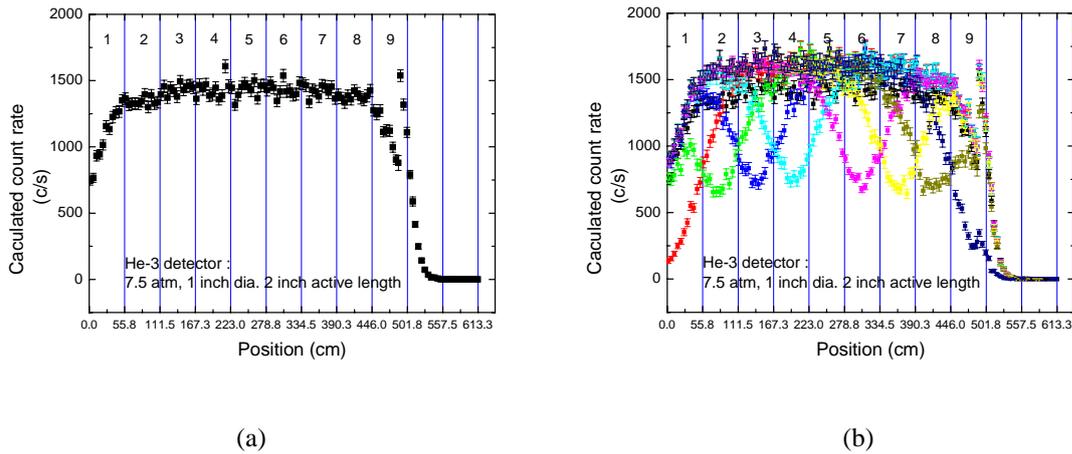


Figure 3. Calculation results of vertical profile in a re-verification tube (a) 9 spent fuel baskets in a canister (b) one dummy and 8 spent fuel baskets in a canister.

3. Neutron detection system

The neutron detection system was designed to pass through a bend on the re-verification tube. A detector connected to the preamplifier as shown in Figure 4. To traverse a bend, the solid part would be smaller than 7.1 inch length and 1.3 inch diameter. He-3 (7.5 atm with 1 inch diameter and 2 inch active length) and BF₃ (600 torr with 1 inch diameter and 2 inch active length) detectors were used for the test. We initially tried to test different type of detectors (He-3 and BF₃) having same gas pressure and same size, however, we could not find a commercially available BF₃ detector which is equivalent to the He-3 tube.

Acceptable operating temperature of He-3 and BF₃ detectors are known as 200-250 °C and 100 °C respectively. Comparing to the BF₃ detector, He-3 can be operated at much higher pressure with acceptable gas multiplication behaviour and are therefore preferred for those application in which maximum detection efficiency is important. However, the lower Q-value of the He-3 reaction makes gamma-ray discrimination more difficult than for an equivalent BF₃ tube. In strong gamma fields, the detectors may decrease neutron counting ability due to the effect of gamma pulse pileup. Therefore the operating high voltage should be determined with an actual spent fuel canister to verify the gamma ray pileup and to obtain maximum neutron count rate.

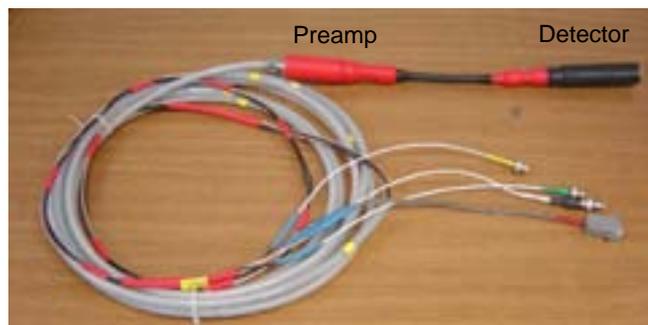


Figure 4. Photo of a neutron detection system.

Figure 4 shows a photograph of a neutron detection system. The neutron detector is connected with a PDT10- HN, charge sensitive preamplifier produced by Precision Data Technologies. The detector and preamplifier were housed in aluminium cases to protect against external impulse inside the re-verification tube. The outside of aluminium cases were covered with water proof heat shrinkable tube in order to operate at very humid environment. 10 m length cables for analogue and TTL signal

outputs, the HV power supply, and preamplifier power supply were connected to PDT module to provide redundancy and diagnostic capability.

4. Detector characterization

4.1 Test using a neutron source

The count rate was measured with respect to the applied high voltage using a $^{241}\text{AmLi}$ neutron source (2.9 mCi). The outer dimension of the source including shielding materials is 17.8 cm dia. and 24.3 cm length. The PDT module was provided with preset threshold level by the company for neutron detection. Figure 5 shows the measurement result from He-3 and BF3 detectors.

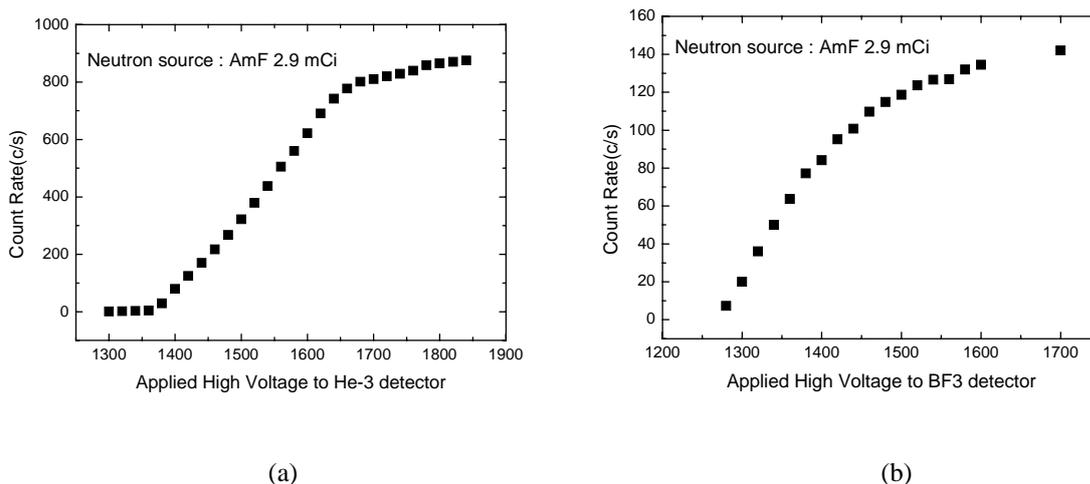


Figure 5. Detector count rate with respect to the applied high voltage (a) He-3 detector having 1 inch dia. and 2 inch active length with 7.5 atm gas pressure (b) BF3 detector having 1 inch dia. and 2 inch active length with 600 torr gas pressure.

The saturation of count rate was found from around 1680V for He-3, and 1400V for BF3. Recommended high voltages from manufacturers are 1400V-1800V for He-3 and 1200V-1450V for BF3.

4.2 Test in the CANDU spent fuel canister

In order to verify gamma ray pileup and to determine the applied high voltage, the neutron counting systems were tested at the actual spent fuel canister. The detector was positioned at the center of 4th baskets in the re-verification tube and measured count rate as applied voltage. As shown in Figure 6, typical gamma pileup characteristic was found for He-3 detector, while there was no visible gamma effect on BF3 tube. This might be due to the difference of gas pressure and Q-value of the detectors.

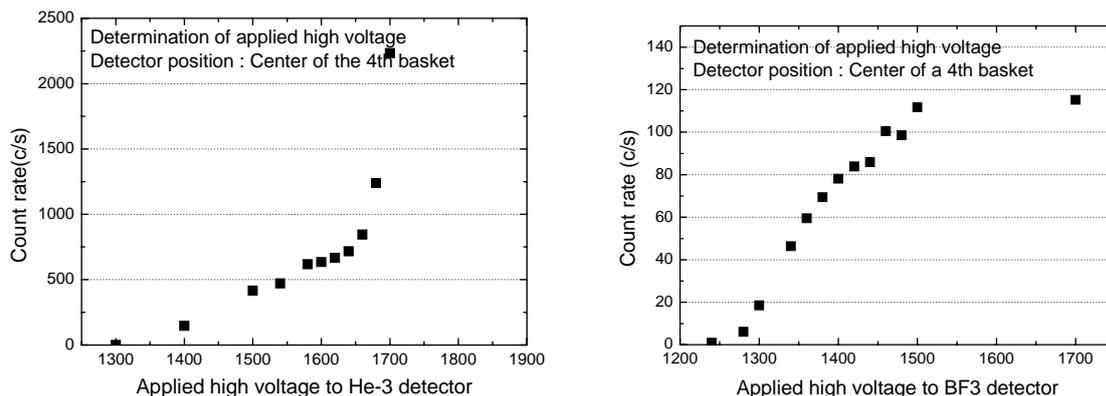


Figure 6. Detector count rate with respect to the applied high voltage (a) He-3 detector having 1 inch dia. and 2 inch active length with 7.5 atm gas pressure (b) BF3 detector having 1 inch dia. and 2 inch active length with 600 torr gas pressure.

5. Summary

Neutron detection was proposed to use verify CANDU spent fuel canister in case of a Containment/Surveillance failure. The signal contribution from individual bundles to the detector was calculated for the neutron and gamma detection. From the calculation, it is expected that the passive neutron detection provide entire spent fuel information in the canister in reasonable measurement time.

The prototype neutron detection systems were fabricated using He-3 and BF3 detectors, and the preliminary field test was performed in this study. Total neutron count rate was measured and the operating voltage was determined in order to achieve maximum count rate in the real field. For He-3 detector, the detector operating voltage was determined around 1640V and the count rate was 736 c/s. It seems that the gamma signal is dominant over 1750 V due to the gamma pileup effect. For the BF3 detector, the gamma pileup effect was not visible until applying the H.V. up to 1700V. The neutron count rate of the BF3 detector was measured 111 c/s at 1500V. In the future, the reproducibility and stability tests will be performed, and the detection limit and actual validity of the neutron counting methodology will be discussed.

ACKNOWLEDGEMENTS

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Towards the re-verification of in-process tank calibrations

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Abstract. Although knowledge of the calibration of in-process tanks in nuclear fuel reprocessing plants is essential for obtaining the near-real time account, operational costs limit the extent to which these tanks can be re-calibrated. A continuous-fill technique is proposed to provide limited assurance that a tank calibration continues to be valid. This is demonstrated on data obtained from the TAME facility at JRC, Ispra.

1. Introduction

Solution measurement data collection equipment and evaluation software are now part of the nuclear materials safeguards provision in nuclear materials reprocessing plants. Such systems collect and evaluate data from many tanks in a facility, to support inventory verification and to provide reassurance that a facility is operating as declared. In conventional safeguards, considerable effort is focused on the measurement of the contents of vessels located at key measurement points for the generation of the nuclear materials account, less effort is expended on others. With solution monitoring, the focus has widened. For instance the IAEA now monitors the contents of approximately 80 vessels at the Rokkasho Reprocessing Plant [1].

Often the quantity of solution contained in each vessel is estimated on the basis of dip-tube pressure measurements, temperature measurements and tank calibration data. The generation of tank calibration data is a costly and time consuming exercise, because it requires that a specific set of experiments should be performed on each vessel in turn, during commissioning. Higuchi et al [2] have described the preparations required to obtain high quality calibration data. They recommend that "the calibration liquid should be allowed to equilibrate with the cell temperature", conclude that this "requirement can only be met during the commissioning period", then recommend "that the resulting calibration function and associated calibration errors be used for the lifetime of the tank". DeRidder et al [3] confirm that "other effects ... are negligible in comparison with the uncertainties associated with the temperature corrections". This data must be re-verified at intervals to ensure that it has not changed over time. For instance the calibrations of certain tanks are re-verified on an annual basis. Although considerable effort might be devoted to ensuring that an appropriate environment ensues when re-verifying input and output accountability vessel calibrations, there are obvious disincentives for carrying out the same procedures on other vessels. An alternative approach is desirable.

Ideally a tank calibration would be re-verified by analysing normal operational data. More likely, a normal operational procedure might be adapted to facilitate re-verification. For instance the acid or water imported into a tank during washouts might be metered accurately at a constant flowrate, monitored and diptube/temperature trends evaluated. Both approaches involve continuous (as opposed to discrete,) filling. Monitoring during an emptying operation is not considered to be practicable,

because of the complications that might have to be overcome to meter solution out of a radioactive tank. Irrespective of the route taken, what is important is that uncertainties that result from longer-term temperature fluctuations are accommodated appropriately.

This paper argues that it would not be practicable to re-verify all tank calibrations on a regular basis, segment by segment, but only to confirm the validity of the calibration in general. To do this checks on the location of specific features in the calibration are proposed instead. These features can pertain to changes in the actual tank structure and to items that are inserted into its interior. Certain features might be so small that they are not visible in the calibration equations themselves, but nevertheless might still provide a 'crisp' reference to look for. For instance the DTank located in the TAME Laboratory at JRC, Ispra has a relatively thin, horizontal, internal support plate, which is located near the top of the vessel. Such features might be identified from engineering drawings, or more likely, from the first application of the experimental procedure described here. It is shown that, with flow metering that is generally available, it should be possible to detect gross changes in the calibration, for instance as a result of solid material build-up in the heel. Greater detail should be attainable with the installation of more accurate flow metering.

2. The Procedure

The precise procedure that would be adopted would depend on what is known beforehand. Even if a feature is first identified from drawings, experimentation would still be required to obtain their locations, because, as highlighted by De Ridder et al [3], the height in a tank calibration is not necessarily the same as a physically correct height. In other words it is more sensible to determine locations experimentally, preferably during commissioning, and then to re-verify their locations subsequently. If nothing is known, then it might be appropriate to perform a number of quick, relatively uncontrolled experiments, merely to ascertain the existence and approximate location of a feature. This is discussed further in Section Feature Identification. It is assumed here that the location (at a height L_o , and corresponding calibration volume, V_o , are known approximately. Clearly it would be preferable if at least one feature is near the top of the vessel, as the re-assessment of its location would re-verify the entire vessel.

During the previous evening, fill the target vessel with 'calibration solution' and leave the rest of the 'calibration solution' in its holding vessel overnight so that the temperatures in both vessels converge to the ambient temperature. In the morning confirm that the solution in the holding vessel is at a temperature not greater than a few degrees above the temperature, θ_r , of the tank solution and preferably a few degrees below. Empty the target vessel and quickly meter in approximately $V \approx V_o \pm \delta V$ litres of the calibration solution; the size of δV will be discussed later. Homogenise for a short period, stopping if the solution temperature rises more than a few degrees above θ_r . Now add $2\delta V$, this time metering at a constant mass flowrate, W , and record the level history, $L(t)$. At least one level history should be generated early on in order to provide a reference, r , to compare with. Further datasets would be collected on subsequent visits to the vessel.

3. Data Analysis

Again, the precise data analysis procedure that would be adopted would depend on certain factors, particularly on the instrumentation. For instance the approach would be different (a) if the level history is to be compared with knowledge about the geometry of the tank and its contents (e.g. through drawings), to (b) if the level history is to be compared with measurements taken previously using the same instrumentation and analysis, to (c) if the level history is to be compared with measurements taken previously using different instrumentation, but the same analysis, to (d) if the level history is to be compared with measurements taken previously using different instrumentation and different analysis. The most straightforward approach (b) is assumed here, because this avoids addressing the issues in the other approaches, which stem from the phase shifts that occur in both the instrumentation

and analysis. Even then, the precise approach will depend on the instrumentation available: this is explained in more detail in Appendix B of Reference [4].

The rate at which the tank level rises will change as it passes a crisp feature like a support plate, and this rate change should be observable at the same location in all the level time histories collected, provided that the solution is metered into the vessel with the same flow rate. Unfortunately slope change detection is sensitive to unwanted high frequency variations. To elaborate on this, suppose that the tank is filled at a constant flowrate so that its volume measurement can be approximated by

$$V(t) = \alpha t + \Delta V \sin \omega t + n(t) \quad [1]$$

where the sinusoidal term models bubble fluctuations and $n(t)$ is noise. The slope, or gradient α , can now be obtained by differentiating the above:

$$\dot{V}(t) = \alpha - \Delta V \omega \cos \omega t + \dot{n}(t) \quad [2]$$

A change in slope will be masked by the other terms: the size of the bubble term is now frequency dependent, the higher the frequency the greater its influence; similarly high frequency noise components will have a greater effect, than lower components. It is therefore important that level measurement records are processed carefully, in particular avoiding aliasing as much as possible and attenuating high frequencies where possible. However the degree of attenuation must not be so excessive that it 'masks' a change in slope. Thus the degree of filtering needed to ameliorate bubble and noise effects will affect the fidelity of the technique. Figure 1 gives an example of a filter that might be suitable: this provides differentiation at low frequencies and attenuation at high frequencies, the frequency at which the plot turns is slightly below bubble frequency, ω . A typical output from this filter is shown in Figure 2, which pertains to filtered data that was collected from a small tank that was filled at a rate of about 60 litres per hour.

The location of a particular feature can now be re-verified by estimating the shift, S , in the location of the feature in a new set of filtered data, x , relative to a reference filtered data set, y , obtained previously. This can be estimated by cross correlation. It is assumed that the collection of the datasets, one collected during one visit, another collected during a subsequent visit, were both initiated from the same starting volume V_o .

This can be estimated by cross correlation. It is assumed that the collection of the datasets, one collected during one visit, another collected during a subsequent visit, were both initiated from the same starting volume V_o . The following form of cross correlation function is recommended for tank sections that have a uniform cross-sectional area:

$$R_{xy}(k) = \sum_{i=f-M}^{f+M} \left[(x_i - \bar{x}) - (y_{i+k} - \bar{y}) \right]^2 \quad [3]$$

where $k : \mathbb{Z} \rightarrow \mathbb{Z}$ and \bar{x} and \bar{y} provide (horizontal) baselines. The shift is then located at the minimum of this function. The choice of focus, f , window length M and range N , and the preparation of the time series, x and y , are described more fully in Appendix B of Reference [4]. A modified version of Equation 3 is needed to perform the cross-correlation for tanks with a non-uniform cross-sectional area:

$$R_{xy}(k) = \sum_{i=f-M}^{f+M} [(x_i - b_i) - (y_{i+k} - c_{i+k})]^2 \quad [4]$$

The choice of baselines b and c is not that straightforward. The most intuitive approach would be to produce theoretical equations for these baselines by considering the curvature in the calibration equation, and then to evaluate these at the heights that are measured. This approach is outlined in Section 5. Unfortunately (bubble) noise corruption of these height measurements makes this route difficult. Instead, a long-time constant, bi-directional EWMA filter is found to perform adequately.

3.1. A test case

The approach was applied to the location of a thin, horizontal support plate positioned in the top most section of the 420 litre DTank (Figure 3). Water was metered into the DTank at a rate of 45 kg/hr and its diptube pressures were recorded using standard PPMD instrumentation [5], which outputs highly accurate, filtered data. Figure 4 shows time series pertaining to 3 separate experiments: in each case the pressure data has been filtered further and differentiated. The bottom plot is the average of these three time series. Focusing on the top trend, it can be seen that the noise levels are quite high, so that the thin, horizontal support plate is barely visible at around 400 seconds. It was for this reason that the experiment was repeated a further two times, so that an averaged series could be obtained, which would have slightly reduced noise levels. Note the feature initially goes the other way, which is similar to Figure 2 above. A correlation accuracy of about 10 seconds was possible, which was equivalent to about 0.15 kg of solution.

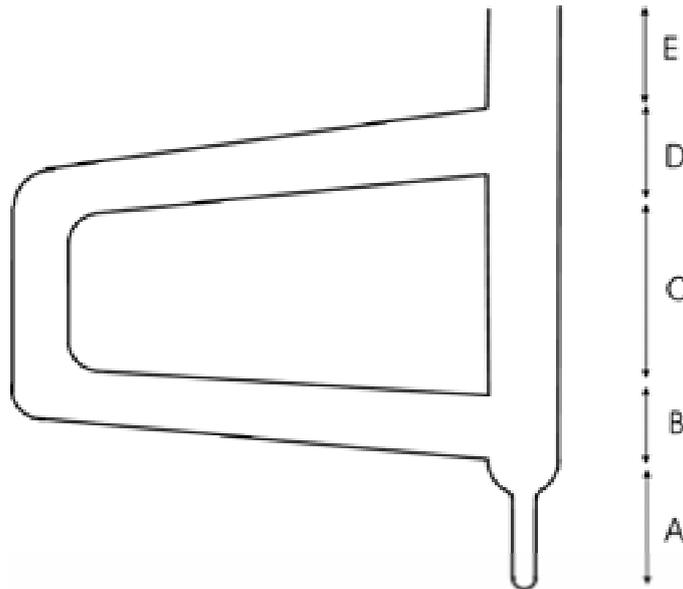


FIG. 1. The DTank & its regions.

4. Error Analysis

There are a number of sources of inaccuracy: with estimating the alignment of the two datasets, with metering, and with thermal non-equilibrium.

4.1.1. Estimating the alignment of the two datasets

The alignment of the two datasets can be estimated fairly precisely, provided that the same instrumentation and analysis algorithms have been used in the production of both datasets. For instance a mass error equivalent to about 10 seconds was obtained in the DTank experiment above assuming that the tank was of uniform cross-sectional area in the region examined.

4.1.2. Metering

The metering device installed on the DTank is typical of what is available commercially. Its operation is based on the Coriolis principle, which enables accurate mass flow metering that is independent of density. A conversion is therefore required to obtain the equivalent volumetric flow rate, which might vary with time if the temperature of the solution fluctuates significantly. The metering device installed on the DTank is rated at a maximum flowrate of 300 kg/hr with a repeatability of "0.1% of rate" and an accuracy of "0.5% of rate". The metering device forms part of a local proportional plus integral control system that regulates the flowrate to a setpoint. For a 400 litre tank, the total quantity added into the tank could therefore be in error by as much as 2.4 kg. There are number of ways of reducing this error, the most obvious being to repeat the experiment many times.

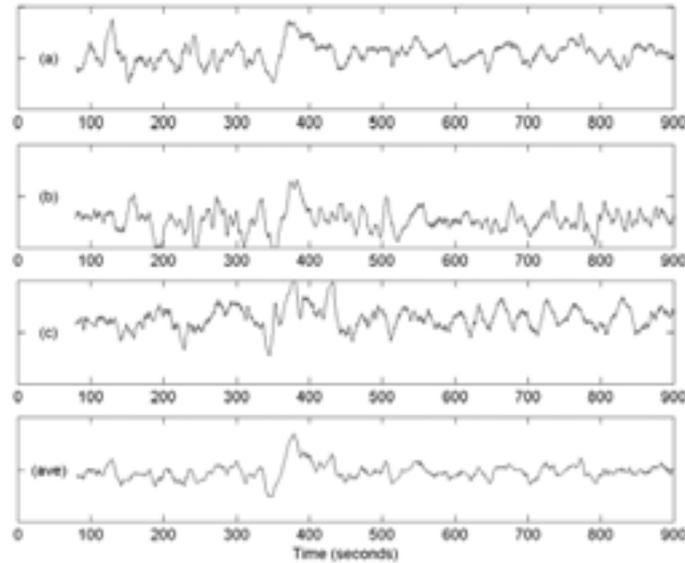


FIG. 2. Three DTank processed time series + their average.

4.1.3. Thermal non-equilibrium

An idea of the scale of inaccuracy caused by thermal non-equilibrium can be obtained by seeing what happens if temperature variations are neglected. Normally a tank calibration is corrected for:

shrinkage/swelling:
$$V = \left(\frac{\rho_{ref}}{\rho} \right) V_{ref}$$

and metal expansion:
$$V = V_{ref} \left(1 + \alpha_m (\theta - \theta_{ref}) \right)^3$$

$$\square V_{ref} \left(1 + 3\alpha_m (\theta - \theta_{ref}) \right)$$

on the basis that the tank was originally calibrated whilst in thermal equilibrium, at a temperature θ_{ref} and with a solution density ρ_{ref} , but is now operated in thermal equilibrium, at a different temperature, θ and density, ρ . Table 1 quantifies the effect of failing to make these corrections, for a 400 litre tank, calibrated at 25°C, but operated at 5°C, 15°C or 20°C.

Table 1. Shrinkage/swelling/expansion errors in litres.

Temperature difference	Shrinkage/swelling/expansion	Expansion only
5°C	0.572	0.103
10°C	1.015	0.206
20°C	1.348	0.413

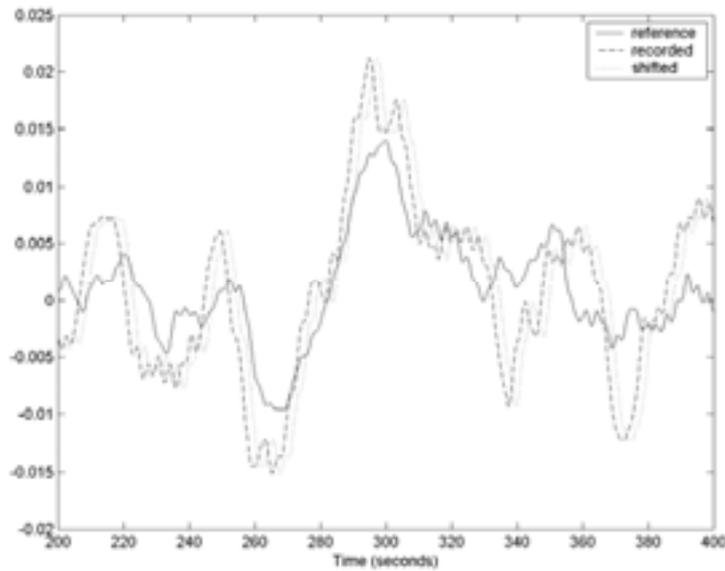


FIG. 3. The result of shifting the 'new' series by the optimum lag.

5. Feature Identification

As stated previously, the identification of suitable crisp features might either be based on knowledge of the construction of the tank, or on an analysis of data obtained during a constant, metering operation. Clearly the latter might be computationally intensive, unless the search can be restricted to particular regions of the tank. A feature will only be visible if the changes in the associated differentiated, filtered time series can be seen in the background of noise. These changes can be estimated from the theoretical, noise free, equation for the differentiated level history, $L'(t)$:

$$L'(t) \equiv \frac{dL}{dt} = \frac{dL}{dV} \frac{dV}{dt} = \alpha \frac{dL}{dV} \quad [5]$$

if the solution is metered into a vessel at a constant volumetric flowrate, \dot{V} . If a feature causes a 'sudden' or crisp change, ΔA , in the cross sectional area of a tank that is, locally, of constant area, A , then $L'(t)$ at the feature will be:

$$L'(t) = \frac{\alpha}{(A + \Delta A)} \approx \frac{\alpha}{A} \left(1 - \frac{\Delta A}{A} \right) \quad [6]$$

In other words the absolute change in the differentiated time series will be proportional to the relative change in area. It has already been demonstrated that the top support plate in the D-tank was just about discernible in data collected at a frequency of about 0.2-0.3 Hz and when filled at 45 kg/hr. It is therefore unlikely that a small feature would be discernible in regions of the tank where the cross sectional area is much larger, unless the instrumentation is improved. In addition, the fill rate may also be a practical issue. For instance the DTank would take in excess of 8 hours to fill, if filled at a rate of 45 kg/hr. Figure 6 shows averaged, differentiated, filtered time series obtained when the DTank was filled, twice, at a rate of 100 kg/hr, where the regions are as those defined in Figure 3. Height information has been added to the right-hand graphs to give an idea of the location in the tank. The thin, horizontal plate is located at a height of about 1830mm. Note that the slope estimate for the top most region (E) is about a factor of 10 times larger than that for Regions B & D, and about a factor of 2 for Region C. Note also that the slopes change significantly over Regions B & D and that ('absolute' sized) noise is seen to dominate these plots more than in the other two. Thus a feature identification strategy should contain the following: mark regions where identification might be practicable (based on cross-sectional areas); fill these regions at relatively low input flow rates; if time allows, repeat and average.

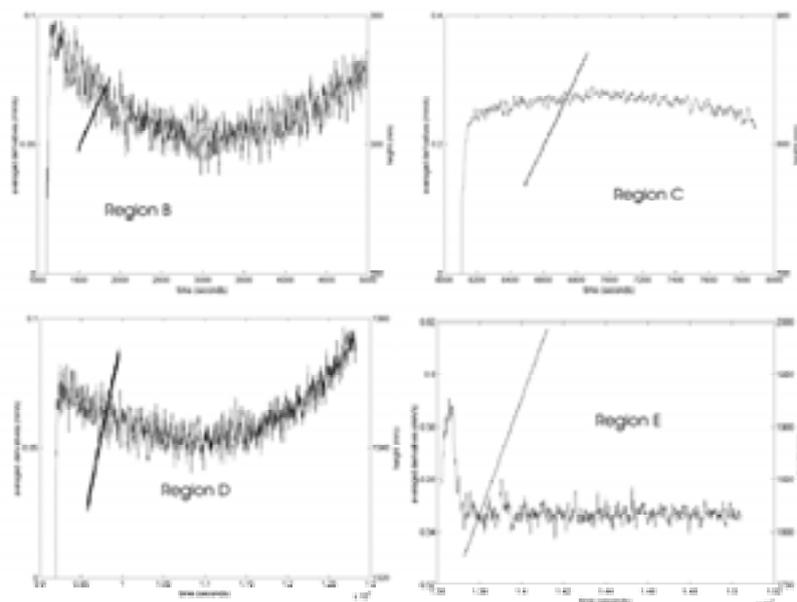


FIG. 4. Regional slopes.

6. Conclusions

Although it should be possible to verify the location of crisp features accurately, inaccuracy will still centre on issues relating to metering and thermal non-equilibrium. Ideally the testing should be performed under the same environmental conditions, with the same instrumentation and following the same procedures. However this should still be far less stringent than with conventional tank calibration, because most of the solution is added as one batch, and subsequent to this, a wait period can be implemented so that the tank is encouraged to reach a state of thermal equilibrium.

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Strategic safeguards considerations of the global nuclear energy renaissance

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Abstract. As nuclear energy use advances through the next decade, the international community has an important opportunity to work together to ensure that nonproliferation is an essential element of the development process. For the United States, this idea is incorporated in the U.S. initiative titled the Global Nuclear Energy Partnership (GNEP) announced by the U.S. Department of Energy in February 2006. GNEP is one of the key conceptual international fuel cycle approaches that have been proposed to enhance nonproliferation as nuclear energy use expands worldwide. This paper will look at how basic elements of two hypothetical international fuel cycle approaches could impact international safeguards approaches and technologies. For example, designing advanced safeguards directly into new nuclear energy systems and fuel cycle facilities could help ensure that any diversion of nuclear materials or modification of the energy systems would be revealed. New international approaches could also encourage vendors of nuclear fuel cycle components, systems, and facilities to work with the IAEA and others to incorporate safeguards systems into the purchased product. This paper will look specifically at how 1) reliable fuel cycle service arrangements and 2) international fuel cycle facility approaches create advantages as well as challenges for the international safeguards system.

1. Introduction

The global renaissance of nuclear energy is well underway. For the first time in the United States since 1978, applications (over 25) for combined construction and operating licenses for new nuclear power plants. The situation is even more dramatic in other parts of the world. Even in environmentalist circles, the need to address pressing energy demands while reducing greenhouse gas emissions has led to general acceptance of a larger role for nuclear energy in the future. Some barriers to nuclear energy growth appear to be weakening. However, sustainable global growth in nuclear energy requires close attention to safety, security, safeguards, and nonproliferation. Failure in any one of these areas has the potential to significantly undermine nuclear power as a viable option. This paper analyzes the safeguards considerations implicit in two conceptual international fuel cycle models, each intended to promote nuclear energy expansion while reducing proliferation risk. The two models are:

- 1) Reliable fuel cycle service arrangements (including enrichment services, fuel supply and eventually, spent fuel disposition) for recipient states provided by supplier states.
- 2) International fuel cycle facilities: multinational investment in and possibly operation of a nuclear fuel cycle facility.

¹ The views expressed herein are those of the authors and do not necessarily reflect the views of the Pacific Northwest National Laboratory or the U.S. Department of Energy.

2. Premise

Each model offers a conceptual approach to promoting growth of nuclear energy worldwide while reducing proliferation risk. Each approach introduces issues that may impact nuclear safeguards to varying degrees, and early examination and analysis of the impacts can help the international community prepare to take advantage of or manage these. An analysis of safeguards considerations may also lead to identification of common themes and opportunities to harmonize elements of various fuel cycle approaches to optimize safeguards effectiveness and efficiency.

3. Scope and objective

This paper will review the strategic safeguards considerations created by two conceptual models of a nonproliferation-focused international nuclear fuel cycle:

- 1) Reliable Fuel Cycle Services
- 2) International Nuclear Facilities

For each model, the paper will highlight potential safeguards challenges, advantages, and general considerations. In particular, four general areas will be discussed:

- 1) Effectiveness of safeguards – application of full scope safeguards on nuclear material from production through use and final disposition; proliferation resistance; reliance on advanced safeguards; level of transparency of operations
- 2) Cost of safeguards (human and financial resources) – inspector effort, technical complexity, potential for remote monitoring, integrated safeguards
- 3) Managed access – facilitating effective inspections and providing necessary data while limiting access to sensitive information
- 4) Breadth of safeguards – extent to which materials involved within each model would be placed under full-scope IAEA safeguards

The objective of the paper is to identify important safeguards implications of two conceptual nonproliferation-focused international fuel cycle models. Common themes, advantages and challenges to international safeguards will be noted for further consideration and study within the international safeguards community.

4. Features of the conceptual models in brief

Reliable Fuel Cycle Services

Concept

In this model, States able and willing to do so provide enrichment and possibly fuel fabrication services to recipients. Participation in the arrangement is voluntary, although recipients agree to refrain from acquiring enrichment or reprocessing technology. In exchange, suppliers offer reliable access to fuel cycle services at competitive prices to help mitigate the risk of disruptions to reactor operations and obviate the need for indigenous acquisition of enrichment capability. The model would eventually include reliable provision of spent fuel services including interim storage and reprocessing in States able and willing to provide such services. This model would employ the assistance of the IAEA in facilitating arrangements between suppliers and recipients as necessary or desired. A virtual fuel bank or fuel reserve could also be established, potentially under international supervision, to provide additional assurance of supply in cases where political or other unforeseen obstacles arise.

Assumptions

- 1) All nuclear material provided to recipients will be obligated to the State(s) that provided it (both the State that provided the raw uranium as well as the State(s) that further refined and enriched it).
- 2) All nuclear material provided to recipients will be under comprehensive IAEA safeguards including an Additional Protocol.
- 3) Nuclear facilities involved in providing the reliable fuel services will be eligible for IAEA safeguards with an Additional Protocol if located in nuclear weapon State parties to the NPT (NWSs) or other States possessing nuclear weapons, and will be under comprehensive full scope safeguards and the AP if located in Non-Nuclear Weapons States (NNWSs).
- 4) Spent fuel will remain obligated to the supplier State(s) and under comprehensive IAEA safeguards.
- 5) Spent fuel will eventually be sent to a country supplying reliable spent fuel disposition services where the material will continue to be eligible for or placed under IAEA safeguards until it reaches a form suitable for termination of safeguards.
- 6) Transportation of nuclear material will be conducted in compliance with relevant conventions and internationally accepted best practices pertaining to safety, security and safeguards.

5. International nuclear facilities

Concept

Nuclear facilities, whether existing or new, become international in nature. They remain subject to the laws of the country in which they are located. Interested participants (presumably meeting some minimum set of financial or other criteria) invest in part-

ownership of the joint venture (or other form of corporate entity). No access to technology will be provided to or sought by these equity owners/managers. A feedstock reserve (in the case of an enrichment plant) or enriched uranium reserve (in the case of a fuel fabrication plant) would be established in the facility host country to further assure participants that any unforeseen bottlenecks in the supply chain would not disrupt provision of services from the company. The fuel or enriched uranium supplied by an international facility, once removed from the reactor, would be eligible for storage and/or reprocessing by an international facility established for such purpose. The IAEA would provide a forum to facilitate interactions between participants and the facility, or between the facility and interested customers, as requested.

Assumptions

- 1) The material provided by this facility to customers would be obligated to the host country of the facility and the owners of the technology used in the plant (if different from the host country), and both parties would maintain consent rights on the material through its disposition.
- 2) The international facility, regardless of its location, would be under comprehensive IAEA safeguards (based on INFCIRC/153) and the Additional Protocol (of the host country).
- 3) Neither enrichment nor reprocessing technology would be revealed to any participant countries or the IAEA; managed access would be used to facilitate provision of inspection data without revealing sensitive information.
- 4) Material provided by the international facility would be under comprehensive IAEA safeguards including an AP in NNWSs. The material would be used in facilities made eligible for IAEA safeguards in NWSs or other State possessing nuclear weapons. The material would remain under IAEA safeguards until in such form as to be eligible for termination of safeguards.
- 5) Any material in a fuel reserve or feedstock reserve established as part of the international facility would be under comprehensive IAEA safeguards in a NNWS or eligible for IAEA safeguards in a NWS or other State possessing nuclear weapons.
- 6) The international facility and its customers would require that transfers of material associated with the facility would be made in accordance with relevant international conventions and internationally accepted best safety and security practices.

6. Safeguards considerations

Raising the Bar

Several common themes of interest to the international safeguards community emerge when reviewing the terms and assumptions of these conceptual models. First, both models offer an opportunity to raise the standard of nuclear material safeguards by creating additional pressure for comprehensive IAEA safeguards and the additional protocol to be applied more broadly and consistently than currently is the case. Both models afford an opportunity to make such a standard a condition of supply of nuclear

fuel cycle services. All participants in these models, both providers of fuel cycle services as well as customers that receive and use the material, would place the material (or make the material eligible for placement) under comprehensive safeguards plus AP. In the case where material is made eligible but not selected for IAEA safeguards in a NWS, it may be useful for that State to commit to reporting information to the IAEA to facilitate transit matching. Both models envision the nuclear material in reprocessing or in storage awaiting further processing also remaining under safeguards and AP. Material that re-entered the fuel cycle, for instance as fast reactor fuel from recycle, would also remain subject to or eligible for IAEA safeguards. The models promote comprehensive safeguards from material production through final disposition. Furthermore, material in transport would be required to be protected according to international best practices by suppliers and recipients alike. In summary, both models present an opportunity to more broadly apply international safeguards and to raise the standard of safeguards and security across the board.

Right-sizing the Global Nuclear Footprint

Secondly, the use of either an assured fuel supply regime or international fuel cycle facilities could have the net effect of consolidating capabilities at fewer locations and maintaining their operations consistently at or near capacity. This would have a positive impact on the resource burden to the IAEA in carrying out safeguards, both in terms of minimizing the increase in numbers of complex facilities subject to safeguards (increases in numbers of power reactors, particularly light water reactors, will not pose a significant resource burden as compared to processing facilities) and promoting consistent operating schedules (fewer suspensions of operation, which often require additional measurements and inspection effort at restart). Minimizing the numbers of processing facilities might also reduce overall material transportation volume if these can be co-located in the same countries, or better yet, fuel fabrication could be co-located with enrichment facilities. The models may promote coordinated planning by service providers (whether suppliers or international facilities) which would make decision-making more transparent and afford an opportunity to assure the market that facility capacity is balanced with market demand. Right-sizing the nuclear footprint will certainly help reduce the overall cost of the international nuclear fuel cycle by avoiding unnecessary and expensive infrastructure beyond what the market demands. However, as implied above, raising the standard of safeguards may require additional agency effort in the nearer-term.

Designed-in Safeguards

Thirdly, both models envision increased collaboration with international partners. With such collaboration, each participating country may contribute to incorporating safeguards design, instrumentation, modeling, and proliferation resistant features into plants and operations. The net result should exceed that of any one country or company alone. Additionally, pooling the tools, skills, financial resources and technologies of many partners will offer diversity in finding solutions and could reduce the cost of R&D. Broad international involvement in planning for safeguards during facility design, using modeling and proliferation resistance analysis to optimize safeguards approaches, and integrating safeguards with process monitoring certainly have the potential to increase the

efficiency/effectiveness, while reducing the cost of nuclear material safeguards and security.

These models present an intriguing opportunity to establish a new norm for vendors of nuclear facilities and technologies. Just as with safety systems, the future international fuel cycle could encourage vendors to incorporate features which facilitate safeguards directly into products that are provided as part of the export of the facility or technology to the customer. The costs of these “enhanced safeguardability” features would be included in the sales price. The IAEA could participate in defining design requirements for such features, to ensure that they meet IAEA needs and enhance the verification process. This approach could reduce uncertainty in applying safeguards to new plants, establish some minimum “safeguardability” of various facility types, and facilitate training for new inspectors. This approach could also improve the economics of developing measurement devices and other tools and instrumentation, as they could service a larger number of facilities (reduce the need for unique instruments). The safeguards community would need to work together to address any risks of defeating features of a designed-in safeguards system, and may wish to initiate this concept in a phased manner, beginning with vendors ensuring enhanced “safeguardability,” followed by integrating safeguards measurements with process monitoring features, ending with a situation where proliferation resistant features and designed-in safeguards become the norm.

Increased Transparency and Managed Access

Fourth, both models foresee an international fuel cycle operating more transparently in the future. Transparency will be achieved through involvement of international partners in a single facility or network of facilities, increased collaboration in industry as well as government in R&D, construction and operation of facilities, and broader application of comprehensive IAEA safeguards plus the AP. In addition, both models see the IAEA having a role in facilitating arrangements between suppliers and customers, assuring undisrupted supply of fuel and/or services, and monitoring reserves intended to increase confidence in assured supply. The international facility model in particular, as it involves presence of foreign partners at the site, will require careful planning for managed access. The business partners may require access to the plant for quality control, oversight of management of operations, safety, licensing, and so on. Customers, if different from business partners, will require information for quality control and verification of product specifications. However, no access to sensitive technology would be provided. Therefore, managed access needs to be well defined by the technology holder so it can be used to support such needs, including effective IAEA inspections, without revealing sensitive information. It is widely recognized that transparency facilitates effective verification, and thus, these two models appear to offer progress in this area.

7. Conclusion

History may look back on the early part of the 21st century as a time when the world realized that nuclear power was needed to meet increasing energy demand without adding

to the greenhouse gas burden. This paper discusses the possibility that a large increase in nuclear power need not result in commensurate increase in nuclear proliferation. The two conceptual models analyzed in this paper offer opportunities to improve international nuclear safeguards in several ways. First, the overall standard of safeguards can rise by creating pressure on both suppliers and recipients of nuclear material to maintain comprehensive safeguards and the AP on nuclear material from production through use and final disposition. Second, international fuel cycle models facilitate “right-sizing” the global nuclear footprint by consolidating capability in fewer facilities and optimizing each facility’s capacity. Third, increased collaboration of international partners in both models creates pools of expertise, equipment, funding, technologies and research from which to design optimal safeguards systems integrated with process monitoring and benefiting from enhanced proliferation resistant plant features. International collaboration may also present an opportunity to encourage vendors of nuclear technology and facilities to design-in safeguards and provide these integrated systems as part of the overall package. Fourth, involvement of the IAEA and international partners at nuclear facilities will increase the transparency of their operations, and will require effective use of managed access to provide inspection and other information without revealing sensitive technology. Both models encourage limits on the spread of enrichment and reprocessing technology which benefits nonproliferation and international safeguards, and should reduce the overall cost of the international fuel cycle by avoiding construction of unnecessary and expensive infrastructure.

Enhancement of proliferation resistance of a fuel cycle by applying the INPRO methodology

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Abstract. In the course of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) the IAEA has recently developed the INPRO Methodology for assessing Innovative Nuclear Systems (INS), and has published its INPRO Phase 1B Report describing the assessment methodology proposed. This methodology covers five areas, namely Economics, Environment, Waste Management, Safety of Nuclear Installations and Proliferation Resistance. In the area of Proliferation Resistance (PR), the INPRO Phase 1B Report assessment methodology ('Ph 1B') proposes to introduce two basic principles, 5 user requirements, 7 indicators, and a number of subsequent variables. Based on several case studies, an INPRO manual has also been prepared for practical implementation of an assessment using the INPRO Methodology.

From the viewpoint of PR, the vulnerabilities and strengths of an innovative nuclear energy system can be identified through an assessment of its PR characteristics, by applying the INPRO Methodology. Then based on the results of this assessment, recommendations for design improvements and future R&D to enhance its PR features can be prepared.

Through testing the assessment methodology as proposed in the INPRO Phase 1B Report in an additional case study, the INPRO Methodology in the area of a proliferation resistance has now been developed further. In order to apply the methodology appropriately to an innovative nuclear energy system, the system of indicators and variables has been expanded to 14 indicators with corresponding evaluation parameters, by rearranging the 7 previous indicators and the subsequent variables. The structure of the developed INPRO Methodology and the meanings of the evaluation schemes such as the basic principles, user requirements, indicators and evaluation parameters for the assessment of a proliferation resistance are explained. The practical assessment procedures and results of a sample application of the INPRO Methodology to the DUPIC fuel cycle are described. The prospects of using the INPRO Methodology to enhance the PR of a future innovative nuclear energy system are also presented.

1. Introduction

INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycles) was initiated by the IAEA in mid 2000 following the recommendations of the 44th IAEA General Conference. INPRO proposed a concept of sustainability with proliferation resistance (PR) as a key component of the future innovative nuclear system (INS), along with economics, safety of nuclear installation, environment, and waste management, for fulfilling the energy needs of the 21st century. In order to set the desired goals for the innovations of a nuclear energy system, a set of basic principles, user requirements and criteria including indicators and acceptance limits were developed and published as IAEA-TECDOC-1362 [1] in June 2003. As a result of 6 national case studies including a Korean case study[3] and 8 individual case studies from other member countries of INPRO, and various IAEA consultancy meetings, a revised set of basic principles, user requirements and criteria were proposed and published as IAEA-TECDOC-1434[2] in Dec 2004.

In the PR area, 5 basic principles (BP) and 5 user requirements (UR) were originally proposed. There was considerable overlap between the BP and the UR, sometimes obscuring the main intent of a particular basic principle. Some user requirements even appeared to be completely redundant, apparently being included only for emphasis. For TECDOC-1434 some of the original BP and UR were deleted or modified and the whole set was reorganized.

After this, at the request of the IAEA, the INPRO methodology for PR described in IAEA-TECDOC-1434, was re-evaluated for its completeness and usefulness, as an activity for the INPRO Phase 1B Part 2, by applying it to the whole fuel cycle of DUPIC (Direct use of PWR Spent Fuel in CANDU Reactors). This PR assessment of the DUPIC fuel cycle was performed by the Republic of Korea and completed in June 2006[4].

This study shows how the indicators of the user requirements for the basic principles in the INPRO Phase 1B report were modified or newly proposed in order to appropriately assess the PR of an innovative nuclear energy system. Using the modified or newly proposed indicators, the practical assessment procedures and results of the application of the INPRO Methodology to the DUPIC fuel fabrication plant (among the 15 system elements of the whole fuel cycle of DUPIC) are described, to examine and demonstrate the appropriateness and usefulness of the presently proposed modifications of UR 1.1 and UR 1.2. Finally, the prospects of utilizing the INPRO methodology to enhance the PR characteristics of a future innovative nuclear energy system are discussed.

2. INPRO PR Methodology

2.1. Review of the INPRO PR methodology

For the assessment of the PR of future innovative nuclear systems, 2 BP and 5 UR were suggested in the INPRO Phase 1B report, in order to provide guidance to governments, designers, regulators, investors and other users of nuclear power and fuel cycle facilities. The three URs under BP1 had one indicator each, and the two URs under BP2 had two indicators each. Several variables were allocated to each indicator, such as the extrinsic measures and the intrinsic features described in IAEA-TECDOC-1434. These intrinsic features and extrinsic measures as variables are classified into four and five categories, respectively.

For the three URs under BP1, however, each indicator was similar to the meaning of the corresponding user requirement, expressed in such a way that it accomplished the role of an indicator. The intrinsic features and extrinsic measures that function as barriers and indicate the PR of the system were expressed as variables under the corresponding indicator. As a result of these changes, it was necessary to revise the INPRO methodology proposed in IAEA-TECDOC-1434 before using it for the DUPIC assessment.

2.2. Modification of the INPRO PR methodology

It is desirable that the features and measures that constitute barriers in terms of PR become indicators, and that each indicator has its own meaningful PR characteristics. Hence, a new modified structure of the BPs and URs including indicators has been proposed as shown in Figure 1.

The modified URPR1.1 (User Requirement of Proliferation Resistance 1.1) in Figure 1 comes mainly from the previous URPR1.2 in IAEA-TECDOC-1434. Moreover, the 'Variables' in IAEA-TECDOC-1434 are rearranged, and 4 new indicators of URPR1.1 have been proposed. In particular, the term of "nuclear technology" has been added to UR1, because nuclear technologies such as the possession of an enrichment facility, a technology capability for extraction of fissile material and the capability to irradiate a target using a reactor or an accelerator are directly linked to the requirement that the

‘Attractiveness of undeclared nuclear material, that could credibly be produced or processed in the innovative nuclear system for a nuclear weapons program, should be low’. The 4 new indicators are divided into 12 detailed evaluation parameters, which are important to evaluate the intrinsic barriers regarding a material’s characteristics and a nuclear technology.

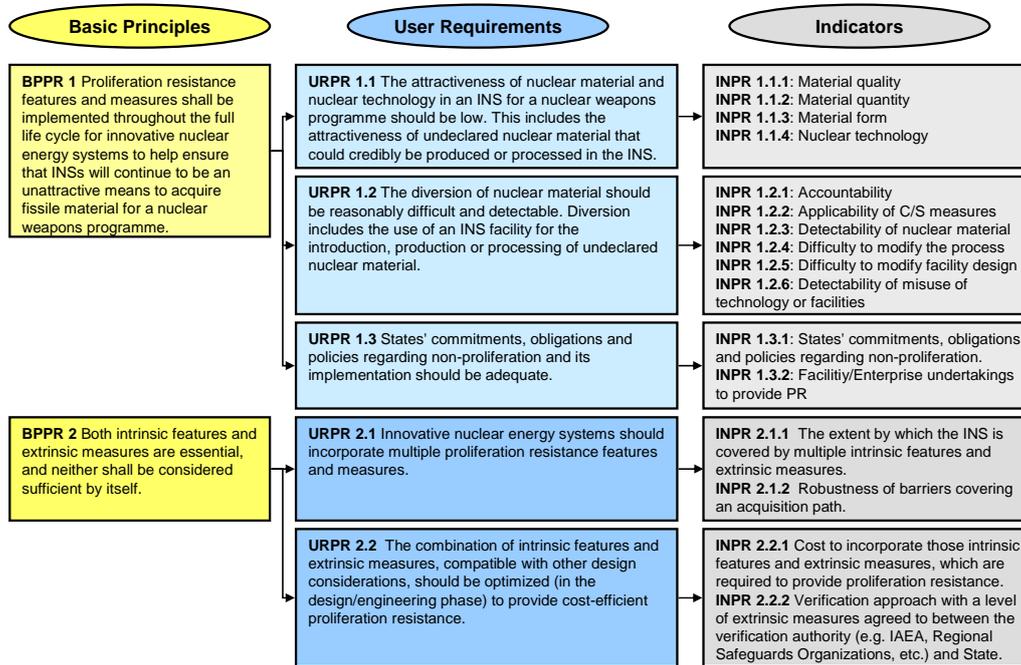


Figure 1. New modified structure of the basic principles, user requirements and indicators.

URPR1.2 comes from the previous URPR1.3 in IAEA-TECDOC-1434. However, 6 new indicators and 13 new evaluation parameters have been proposed in order to both strengthen the safeguardability of a future nuclear energy system, and to clarify the meaning of the evaluation criteria and variables given in IAEA-TECDOC-1434.

The URPR1.3 comes from the previous URPR1.1 in IAEA-TECDOC-1434, and it has two new indicators and 13 new evaluation parameters to evaluate the extrinsic measures. The INPR1.3.2 (Indicator of Proliferation Resistance, assigned to BP1, UP3) in Figure 1 has been introduced to emphasize possible institutional structural arrangements such as a multi-lateral ownership, which was considered as a part of the previous INPR1.1.1 in IAEA-TECDOC-1434.

As a result of the review of IAEA-TECDOC-1434 in the area of PR, modified or new indicators and evaluation parameters have been proposed and utilized to assess the PR characteristics of the whole DUPIC fuel cycle.

3. Application of the INPRO assessment methodology to the dupic fuel cycle

3.1. Characteristics of the DUPIC fuel cycle

The basic concept of the DUPIC fuel cycle is to fabricate CANDU nuclear fuel from PWR spent fuel by use of dry thermal and mechanical processes without separating out the stable fission products. Since a CANDU reactor utilizes natural uranium fuel, the content of the remaining fissile materials in a PWR spent fuel is high enough to be reused in a CANDU reactor even though it still contains fission products.

The main element of the DUPIC fuel cycle is the manufacturing of DUPIC fuel from PWR spent fuel. The PWR spent fuel is first disassembled and the spent fuel elements extracted from the assembly.

These spent fuel elements are cut to small rod-cuts for easy handling. The rod-cuts are de-cladded by mechanical and/or thermal methods to retrieve the PWR spent fuel material. These materials are subjected to a series of oxidations and reductions to make them re-sinterable by the process named OREOX (Oxidation and REDuction of OXide fuel).

Once the re-sinterable powder feedstock is prepared, the manufacturing process is quite similar to the conventional CANDU fuel manufacturing process using a powder/pellet route, except for processing it in a heavily shielded hot cell by remote handling. It involves pre-compaction, granulation, compaction, sintering, grinding, end cap welding by a laser, and final assembly of the DUPIC bundle.

Since there is no process step for the separation of the fission products and transuranic materials while the volatile and semi-volatile elements are removed during the thermal/mechanical treatments, the process materials are highly radioactive throughout all the manufacturing processes. Therefore, the manufacturing process must be performed inside a heavily shielded hot cell by remote handling. The characteristics are a problem for material handling during the fuel manufacture, but provide a strong incentive with regard to the PR of the DUPIC fuel [5,6,7].

In order to investigate the applicability of the INPRO methodology and the new proposed indicators to a future nuclear energy system, this study presents the PR characteristics of the DUPIC fabrication plant among the several system elements in terms of URPR1.1 and URPR1.2. It is limited to these URs because URPR1.3 is a requirement on the extrinsic measures, and applies to the State rather than the DUPIC fuel cycle itself; and the evaluation methodology of URPR2.1 and URPR2.2 of basic principle 2 are as yet not fully established.

3.2. PR evaluation of the DUPIC fuel fabrication plant

3.2.1. Scales for the evaluation parameters

Some barriers can be quantified, but other barriers, such as extrinsic measures or safeguardability, may be expressed only by a logical value such as 'Yes' or 'No'. The present study suggests a five-stage scale, VW(very weak), W(weak), M(moderate), S(strong) and VS(very strong), for the quantifiable evaluation parameters, a logical scale, consisting of U(unacceptable) and A(acceptable) for the extrinsic measures, and W(weak) and S(strong) for some of the intrinsic features.

3.2.2. Evaluation of URPR1.1

Because of the use of the dry process, no fissile material can be separated in a pure form. The material requires a further chemical reprocessing in order to obtain material suitable for a weapons purpose. The presence of some fission products leads to a high dose rate from the material. The DUPIC process has to be carried out in a heavily shielded hot cell because it uses highly radioactive materials. The processing is self-contained, and there is no transporting of intermediate materials outside a facility. Therefore, access to the nuclear materials is extremely difficult.

The material type during the DUPIC fabrication process is characterized as "irradiated direct use material". The isotopic composition, $^{239}\text{Pu}/\text{Pu}$, is ~60 wt%. The radiation dose rate of a DUPIC fuel bundle is ~0.15 Sv/h. The heat generation rate is related mostly to $^{238}\text{Pu}/\text{Pu}$ which is 1.7 wt%, and the spontaneous neutron generation comes from $(^{240}\text{Pu}+^{242}\text{Pu})/\text{Pu}$ which is ~30 wt% [4].

For the material quantity, there are three evaluation parameters: 'Mass of an item', 'Number of items to get one SQ (Significant Quantity)' and 'Number of SQ (material stock or flow)'. The mass of an item is ~ 24 kg, the number of items to get one SQ is ~ 48 assemblies (because ~0.9 MTHM is needed to make one SQ of Pu from the DUPIC fuel). The material form of the DUPIC process is 'spent fuel'.

Regarding the nuclear technology, the whole process employs only thermal and mechanical processes; there is no chemical process. Therefore, it is impossible to extract fissile materials, or to modify a

DUPIC fuel cycle facility and its processes for enrichment. Also, there is no irradiation capability of a target during the DUPIC process. The evaluation results for URPR1.1 are summarized in Table 1.

3.2.3. Evaluation of URPR1.2

Six indicators of URPR1.2 are suggested in this study. The first indicator is 'Accountability' which considers the ratio of Σ MUF (Material Unaccounted for) to a SQ. During the DUPIC process, the Σ MUF/SQ in terms of Pu or ^{233}U is evaluated as ~ 0.5 , based on the assumption of a measurement error of 0.01 and a period of 3 months, respectively. For the measurement method and equipment, a near real time accounting system (NRTA) for a fissile accountability system will be used in the plant. The NRTA system is integrated with an individual nuclear material measurement system. The item accounting system for both the PWR incoming fuel and outgoing DUPIC fuel is based on the modified curium counter. A weighing and NDA system for accounting of bulk material will be applied.

For the second indicator, 'Amenability to C/S (Containment and Surveillance) measures' is proposed by the present study and is composed of three kinds of evaluation parameters: the amenability to containment measures, the amenability to surveillance measures and the amenability to other monitoring systems. These evaluation parameters are assessed as being 'acceptable' in the DUPIC fuel fabrication facility because the C/S measurement systems can easily be installed at the hot cell facility, feed material measurement can be carried out by a PWR spent fuel rod scanning system, and process monitoring can be carried out by an unattended continuous hot-cell monitoring system.

For the third indicator, 'Detectability of nuclear material', two evaluation parameters are suggested by the present study, which are the possibility of identifying a nuclear material by NDA, and the detectability of a radiation signature. Considering the characteristics of a DUPIC fuel fabrication process, the radiation signature during the fuel fabrication process can easily be obtained because of the fuel's strong radioactivity, and the nuclear material during a fuel fabrication process can easily be identified by measurements in passive mode.

For the 4th indicator, 'Difficulty to modify the process', some of the fabrication processes will not be automated, and all the data acquired through the DUPIC fabrication process can be transmitted on-line to the operator. Regarding the transparency of the process, all the activities and all the data in the fabrication facility are authenticable and open to the IAEA.

For the 5th indicator, 'Difficulty to modify the facility design', a hot cell facility is required for treating the PWR spent fuel. The facility design can easily be verified by inspectors.

From the above considerations, the evaluation results for URPR1.2 are summarized in Table 2 below.

Table 1. Evaluation results of URPR1.1.

Indicators	Evaluation Parameter		Evaluation scale					
			W			S		
			VW	W	M	S	VS	
Material quality	Isotopic composition	²³⁸ Pu/Pu (wt%)	> 93	80-93	70-80	60-70	< 60	
		²³⁵ U/U (wt%)	> 90	50-90	20-50	5-20	< 5	
		²³² U _{contam.} for ²³³ U (ppm)	< 1	1-100	100-4000	4000-7000	> 7000	
	Material type		DUM	DIM	L	N	D	
	Radiation field	Dose (mSv/hr)	< 10	10-150	150-1000	1000-10000	> 10000	
	Heat generation	²³⁸ Pu/Pu (wt%)	< 0.1	0.1-1	1-10	10-80	> 80	
Material quantity	Spontaneous neutron generation rate		(²⁴⁰ Pu+ ²⁴² Pu)/Pu (wt%)	< 1	1-10	10-20	20-50	> 50
	Mass of an item (kg)		10	10-100	100-500	500-1000	> 1000	
	No. of items for SQ		1	1-10	10-50	50-100	> 100	
Material form	No. of SQ (material Stock or flow)		> 100	50 ~ 100	10 ~ 50	10 ~ 1	< 1	
	Chemical/physical form	U	Metal	Oxide/Solution	U compounds	Spent fuel	Waste	
		Pu	Metal	Oxide/Solution	Pu compounds	Spent fuel	Waste	
Thorium		Metal	Oxide/Solution	Th compounds	Spent fuel	Waste		
Nuclear technology	Enrichment		Yes			No		
	Extraction of fissile material		Yes			No		
	Irradiation capability of target		Yes			No		

D: Depleted U, N: Natural U, L: Low enriched U, DUM: Direct Use of Un-irradiated Material, DIM: Direct Use of Irradiated Material

Table 2. Evaluation of user requirement 1.2.

Indicators	Evaluation Parameter		Evaluation scale				
			W			S	
			VW	W	M	S	VS
Accountability	Σ MUF/SQ	Kg Pu or ²³³ U	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with HEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with LEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
	Inspector's measurement capabilities						
Amenability of C/S measures	Amenability of containment measures		No			Yes	
	Amenability of surveillance measures		No			Yes	
	Amenability of other monitoring systems		No			Yes	
Detectability of nuclear material	Possibility to identify nuclear material by NDA		No			Yes	
	Detectability of radiation signature		No reliable signature			Reliable signature	
Difficulty to modify the process	Extent of automation		N/A	Manual operation	N/A	Partial automation	Full automation
	Availability of data for inspectors		Very low	Low	Medium	High	Very high
	Transparency of process		No			Yes	
Difficulty to modify facility design	Accessibility of material to inspectors for verification		No			Yes	
	Verifiability of facility design by inspectors		No			Yes	
Detectability of misuse of technology or facilities	Possibility to detect misuse of the INS facilities for processing of undeclared nuclear material		No			Yes	

4. Enhancement of the PR characteristics of a future INS

The PR of a system is characterized by a resistance level which is built up by a combination of intrinsic features and extrinsic measures. An improvement in PR can therefore be achieved on the one hand by the implementation of technical features for reducing the attractiveness of the nuclear materials for nuclear weapon programs, for preventing or inhibiting a diversion or undeclared production of nuclear materials, for facilitating the verification of nuclear material and for providing continuity of knowledge; or on the other hand it can be improved by extrinsic measures comprising the institutional barriers such as treaties, bilaterals, and multinational agreements and the application of IAEA safeguards as appropriate.

Since implementation of the technical aspects for improving PR can reduce reliance on extrinsic measures and thus reduce the costs for the implementation of safeguards, any technical option for improving PR must be considered in balance with the requirements for economics, safety, energy security and public acceptance concerns. For assessing Basic Principle 2, INPRO suggests cost effectiveness regarding the combined facility costs created by additional intrinsic features and safeguards costs. Facility costs, however, affect the pace and direction of a nuclear power development in the future, and this is best left in the hands of the individual States for their own assessment within the context of their own situation. In addition, the INPRO methodology for an evaluation of cost effectiveness has not yet been fully established.

There exist several technology options to improve not only the intrinsic features but also the extrinsic measures. The following features and measures for enhancing the PR of an innovative nuclear energy system can be considered. There are high-level extrinsic measures like multi-national fuel cycle facilities, co-location of fuel cycle facilities, closed fuel cycles and stockpiling. Centralization can provide a stronger international control of proliferation-sensitive technologies. Co-location can limit transportation and storage of a potentially proliferation-sensitive material. A closed fuel cycle can minimize the quantity of nuclear material in the fuel cycle and the production of a proliferation-sensitive material, and can be therefore beneficial for proliferation resistance. The stock piling of excessive inventories of nuclear material may also provide benefits for PR. Source materials such as natural uranium, depleted uranium, and thorium are input materials for many fuel cycles. Although they can not be directly used in a nuclear weapon, the availability of these materials requires due consideration in a PR assessment because they can be used as a source to generate weapons usable materials.

Other areas for consideration are the use of high burnup fuels; use of non-fertile fuels; pyroprocessing technologies in comparison with the conventional aqueous processes; the DUPIC process which does not separate fission products; mixing fission products into attractive separated materials; use of the very dilute fuels large mass and bulk for a weapons useable material, etc. to enhance the material barrier itself.

5. Discussion

The INPRO methodology described in IAEA TECDOC-1434 shows basically comprehensive and useful guidelines for the evaluation of an innovative nuclear energy system. It is more developed than IAEA TECDOC-1362 since the correspondence or the links between the basic principles and user requirements are described more clearly.

The two basic principles and five user requirements for PR are more reasonable and practicable than those described in the previous report. They follow the top-down derivation as well as including the bottom-up evaluation and a general hierarchy of the INPRO methodology. The key to the bottom-up approach for an evaluation is to determine if a nuclear energy system can meet the acceptance limits suggested in the INPRO reports [1,2], and then to judge the higher level requirements.

Seven indicators and their subsequent variables under the user requirements are suggested in IAEA TECDOC-1434. It is noted that the indicators and their subsequent variables are better classified and rearranged for applying the INPRO methodology to a realistic evaluation of an innovative nuclear system. For User Requirement (UR 1.1) referring to 'material attractiveness' four indicators have been defined and each indicator has several evaluation parameters. Regarding the detectability of a diversion (User Requirement 1.2 in IAEA TECDOC-1434), six indicators are proposed describing those features of an INS that may facilitate or impede the implementation of IAEA safeguards. In the present study it can be shown that the re-structured indicators and corresponding evaluation parameters for URPR1.1 and URPR1.2 are reasonably applicable to the DUPIC fuel cycle.

Some areas still need development. As yet there is no method to aggregate the results of the evaluation of each indicator, or to decide how they would be used to determine whether the URs and BPs are satisfied. There is no decision on defining the relative significances among the URs and the corresponding indicators in order to follow the bottom-up process and to integrate the lower level results for each evaluation step. The simplest integration method is based on an expert judgment; other integration methods use the multi attribute decision theory, etc. Thus even when the PR characteristics of the DUPIC fuel fabrication plant were assessed using the INPRO methodology, the present results did not contain the integration process since this needs the relative significances among the barriers, for which there is no general consensus.

The strength of the PR provided by some intrinsic features can depend on State-specific information such as the presence of indigenous uranium resources or the presence of other nuclear facilities. State-specific extrinsic measures such as fuel supply agreements for procurement of fresh fuel and a return of spent fuel (e.g., commitment to multilateral fuel cycle facilities) can affect the PR of a nuclear energy system. However, the intrinsic features that facilitate the implementation of verification measures generally provide PR and are independent of the State in which the system is deployed.

6. Conclusion

In the present study the INPRO methodology described in IAEA-TECDOC-1434 was reviewed and further improved with regard to the completeness and usefulness of the BPs and URs. A total of 14 UR indicators were proposed instead of the previous seven, and these were tested by applying them to the DUPIC fuel fabrication plant. From this, it was concluded that the INPRO methodology as improved by the present study is applicable to the DUPIC fuel cycle with the proposed establishment of the indicators and the corresponding evaluation parameters.

It was also noted that some areas of the methodology, such as the issue of aggregation and a quantification of the barriers should be developed further, both for policy decision makers and for system designers, in order to help them identify which innovative nuclear energy system is more robust against proliferation, and what technical options are needed to enhance the PR.

Since the PR of a system is affected by both intrinsic and extrinsic barriers, it can be improved by implementation of technical features, of which several examples were noted in the present study. Increased PR can be accomplished by adopting high level intrinsic features which might also lessen the requirements for extrinsic measures for a system. However it will be necessary to consider the incremental costs by combining both intrinsic features and extrinsic measures, noting that some extrinsic measures are State dependent and others are not.

Finally, it is indicated that the general methodology to assess the user requirements and corresponding indicators of BP2, which was not described in the present study, should be developed to complete the establishment of the INPRO methodology.

ACKNOWLEDGEMENTS

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Expected new role of IAEA in the area of transparency and proliferation resistance in advanced nuclear fuel cycle

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Abstract. International confidence in the peaceful uses of nuclear energy and nuclear nonproliferation has been maintained by “nuclear material control and accountancy” and “international safeguards” implemented by the International Atomic Energy Agency (IAEA) under Nuclear Nonproliferation Treaty (NPT) framework.

Several recent issues related to nuclear nonproliferation, such as North Korea's announced withdrawal from the NPT, Iran's enrichment program and safeguards failures, and proliferation of sensitive technologies in the nuclear fuel cycle, have given rise to international concern and are still under discussion.

The uncertainty of the current non-proliferation situation arouses proliferation concerns and decreases the overall confidence in peaceful nuclear energy programs. Taking this situation seriously, JAEA has conducted studies on “transparency” and “proliferation resistance” to increase international confidence in its own peaceful energy nuclear programs, especially for an advanced nuclear fuel cycle.

Several efforts to improve “transparency,” such as “Guidelines for the Management of Plutonium (INFCIRC/549 of 16 March 1998)” are on-going. Also, several initiatives on “proliferation resistance” under international programs, such as the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) and the Generation IV nuclear energy system (GEN-IV) are in progress.

Nuclear nonproliferation requirements and the expected new role of IAEA for an advanced nuclear fuel cycle were studied. The expected role of IAEA to certify and maintain “nuclear material control and accountancy,” “transparency,” and “proliferation resistance” will improve reliability of nuclear material control and accountancy in Japan, and globally, thereby decreasing proliferation risk. High reliability will therefore increase international confidence in the peaceful nuclear programs. Low proliferation risk could drastically reduce IAEA's inspection efforts for future, large-scale advanced nuclear fuel cycle facilities and allow more effective use of inspection resources to strengthen the international nonproliferation regime.

1. Introduction

Generally, “transparency” has the role of increasing confidence in activities; and “proliferation resistance” increases confidence in the capability of the facilities by limiting possibilities for misuse.

International confidence in nuclear nonproliferation and peaceful nuclear energy programs have been maintained by “nuclear material control and accountancy” and “international safeguards” implemented by the International Atomic Energy Agency (IAEA) under Nuclear Nonproliferation Treaty (NPT) framework.

International nuclear non-proliferation efforts have been undergoing change since the early 1990s. United Nation inspections in Iraq after the first Gulf War raised doubts about the scope of international safeguards at that point, on how to best deal with undeclared activities. Several recent issues related to nuclear nonproliferation, such as North Korea's announcement to withdraw from the NPT, Iran's

enrichment program and related safeguards failures, and the proliferation of sensitive nuclear fuel cycle technologies, are increasing global concerns and are still under discussion.

A non-proliferation regime that is not completely effective arouses further proliferation concerns and decreases the confidence in peaceful nuclear energy programs. Unfortunately, a country like Japan, which has a full-scale nuclear fuel cycle program, could be exposed to unwarranted criticism. Therefore to proactively respond to this serious situation, JAEA has conducted studies and research on “transparency” and “proliferation resistance” to build international confidence in its peaceful nuclear programs. These activities have been ongoing since 1993.

A diagram that graphically illustrates the loss of confidence in the current nonproliferation regimes is shown in Figure 1. As can be seen, recent world events have lowered the perceived confidence in the traditional nonproliferation regime. In the future, addition of transparency and proliferation resistance measures now under development can restore the level of confidence.

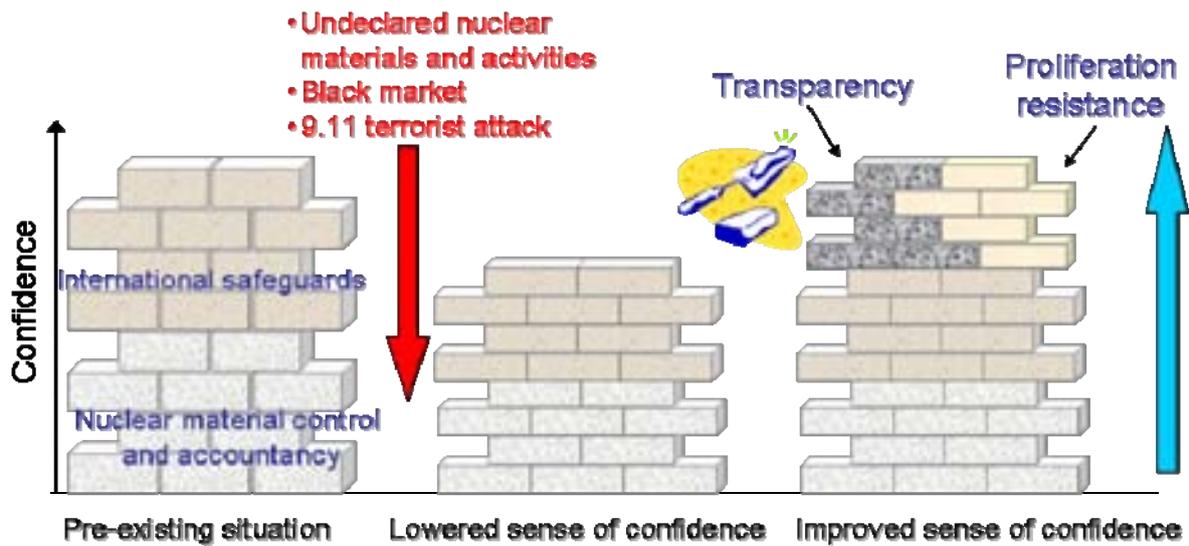


FIG. 1. “Transparency” and “proliferation resistance” to improve international confidence.

2. Efforts to Improve Transparency

There are three ways to improve transparency: systemically, technically, and through safeguards under the IAEA. These areas are discussed below.

2.1. Systemically

Several efforts to improve transparency are on going. In 1994, the US, Russia, UK, France, China, Japan, Germany, Belgium, and Switzerland began discussion on an international framework to improve transparency of plutonium utilization by publishing annual statements of each country's holdings of civil plutonium. After several discussions, basic rules of plutonium utilization in each country and the publishing of annual statements including the amount of separated plutonium in the country were agreed among the nine countries and the IAEA. In 1998, IAEA published the “Guidelines for the Management of Plutonium (INFCIRC/549)” and first statements were made by each country.

Recently, additional nuclear nonproliferation initiatives have been proposed such as the “Multilateral Nuclear Approach (MNA) to the Nuclear Fuel Cycle (INFCIRC/640)” which was proposed by Dr.

Mohamed ElBaradei of IAEA. The concept of MNA includes improved transparency through a multilateral approach.

2.2. *Technically*

New technologies such as remote monitoring, can also help increase transparency and international confidence. The Joyo Remote Monitoring System has been developed under joint study with Sandia National Laboratory (SNL) to test the technology to improve transparency [1]. Feasibility of remote monitoring technologies is being demonstrated. The remote monitoring data is currently shared between SNL and JAEA, and in the future, it will be extended to a regional level and include data from a Republic of Korea facility.

2.3. *Safeguards under IAEA (Additional Protocol)*

The long-standing concern that IAEA's safeguards under comprehensive safeguards agreements would not be capable of detecting undeclared nuclear materials or nuclear activities was confirmed with the post Gulf War revelations regarding Iraq's secret nuclear weapon development program. The response was an IAEA program, "Program 93+2", to strengthen and improve the efficiency of safeguards. The ultimate product of this program was the Additional Protocol (INFCIRC/540), which greatly improves the IAEA's capabilities to detect undeclared nuclear material and activities.

In the area of international safeguards, the Additional Protocol drastically improves nonproliferation "transparency" by expanding the required declarations from States. These declarations include nuclear fuel cycle related research and development activities (article 2.a.(i)), uranium mines and concentration plants (article 2.a.(v)), source material (article 2.a.(vi)), and exports of specified equipment and nonnuclear material (article 2.a.(ix)). This information had not previously been provided to the IAEA under comprehensive safeguards agreements (NPT safeguards). Under the additional protocol, IAEA inspectors have much broader rights of access to facilities and other locations.

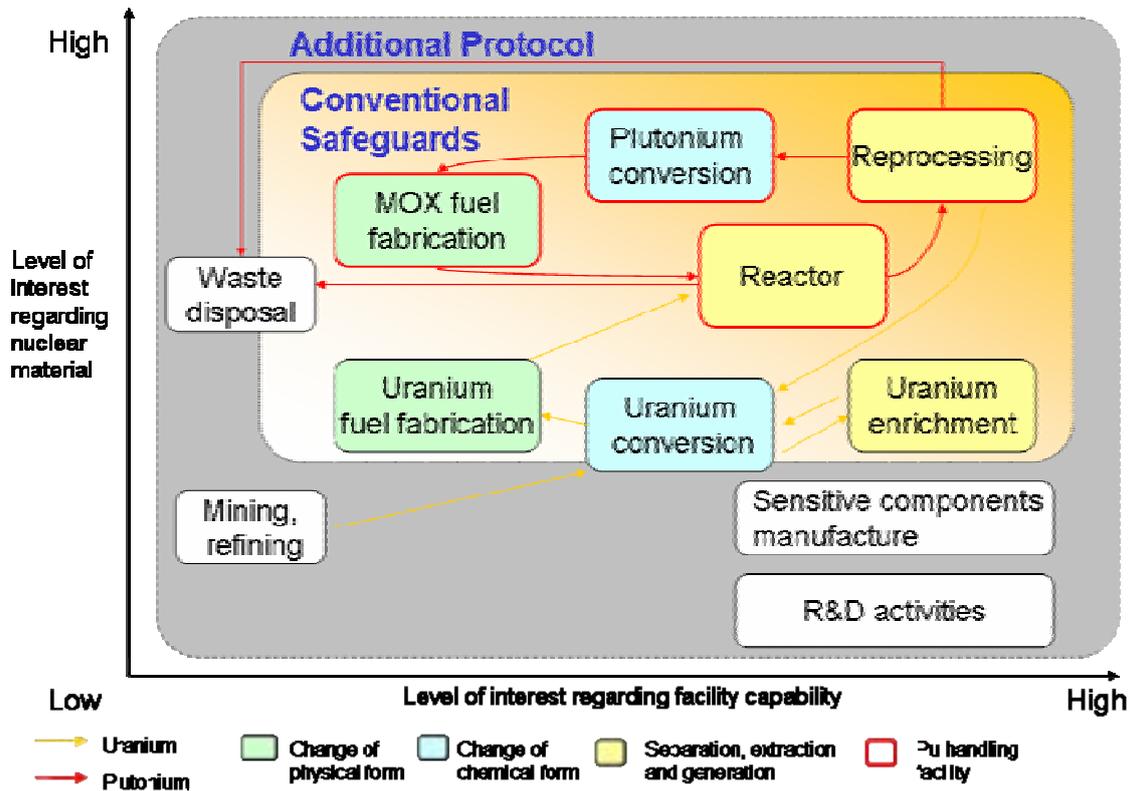


FIG. 2. Interest in facilities and nuclear materials covered by conventional safeguards and the additional protocol.

The relationships between various components of the nuclear fuel cycle and the level of interest regarding nuclear material and facility capability under conventional material accountancy safeguards and the additional protocol are depicted graphically in Figure 2.

Recent discussions on integrated safeguards, which optimize the safeguards measures comprising traditional safeguards and strengthened safeguards, enables streamlining of IAEA safeguards. Integrated safeguards also provide for a further possibility of even more efficient safeguards through improvements in “transparency.”

3. Development of Future Advanced Nuclear Fuel Cycle and Proliferation Resistance

Current efforts to develop advanced nuclear energy systems, such as the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) [2], the Generation IV Nuclear Energy Systems program (GEN-IV) [3], Global Nuclear Energy Partnership (GNEP) [4], and Japan’s feasibility studies for a commercialized fast reactor cycle system (FS) [5] have been promoted by various organization and States. In these programs, “proliferation resistance” is one of the key elements and is included in the design of the systems. As shown in Figure 3, proliferation resistance can be seen as an important element of each of these initiatives.

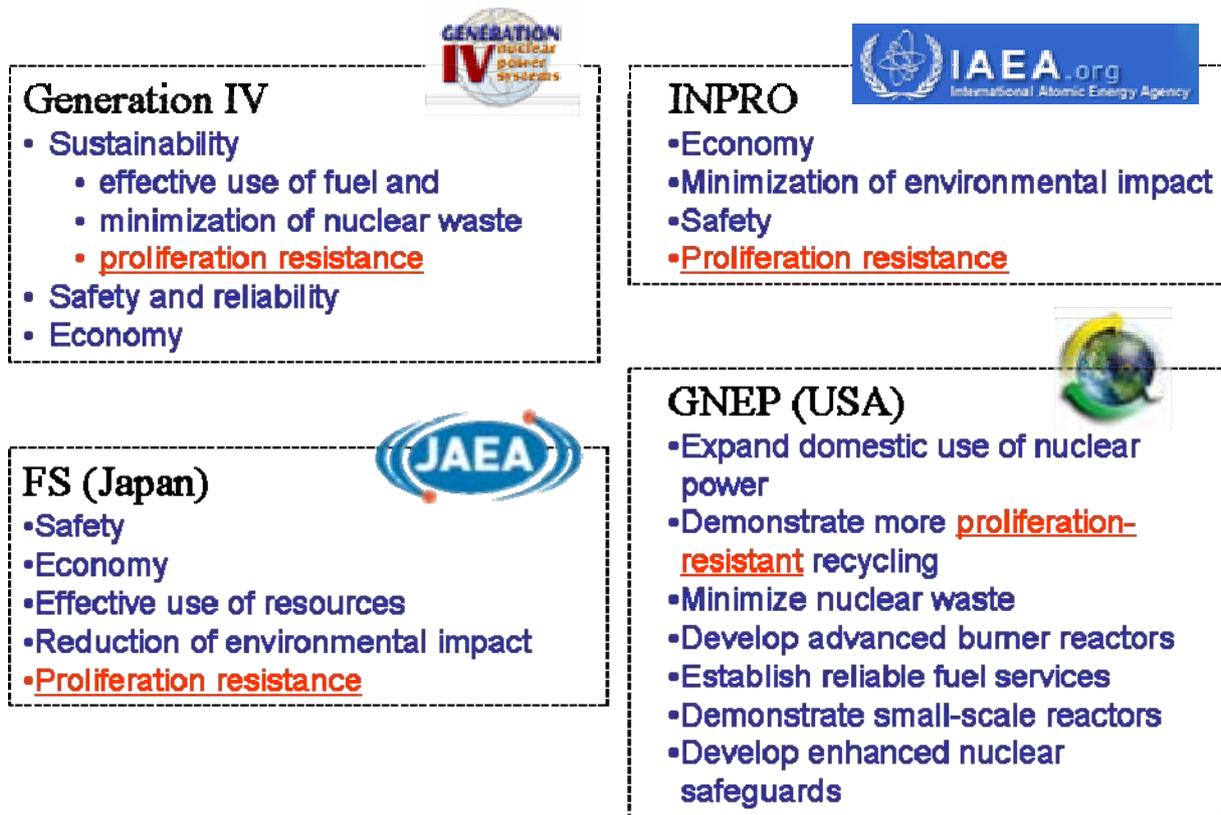


FIG. 3. Advanced nuclear energy systems and proliferation resistance.

4. Improvement of Nuclear Security

As proactive measures in response to the September eleventh 2001 terrorist attacks, the IAEA Board of Governors approved an action plan in March 2002 designed to upgrade worldwide protection against acts of terrorism involving nuclear and other radioactive materials. As part of building a nuclear security framework, work on the development of additional guidelines and recommendations to strengthen the Convention on the Physical Protection of Nuclear Material (CPPNM, INFCIRC/225/Rev.4) are on-going. These guidelines and recommendations are aimed primarily at securing nuclear facilities and materials.

5. Nuclear Nonproliferation Regime in the Future

In the future, advanced nuclear fuel cycle facilities will be designed, constructed and operated based on the results of initiatives such as INPRO, GEN-VI, GNEP, and/or FS. For these advanced facilities, the following requirements related to nuclear nonproliferation will be applied:

- ✓ In addition to the comprehensive safeguards requirements, additional protocol measures and/or integrated safeguards will be the norm;
- ✓ Improved transparency under the guidelines, MNA, and other frameworks to maintain compliance with international authorities;
- ✓ Proliferation-resistant facility design; and

- ✓ Nuclear material control and accountancy with upgraded physical protection that satisfies INFCIRC/225/Rev.4.

Under these conditions, “nuclear material control and accountancy” and “transparency” will improve the reliability of nuclear material control and accountancy in the country, and proliferation risk will be lowered through proliferation-resistance designs. The comprehensive goal is to achieve a greater sense of confidence through adherence to these requirements. This concept is illustrated in Figure 4, which depicts greater confidence as a result of improvements in the future nuclear nonproliferation regime.

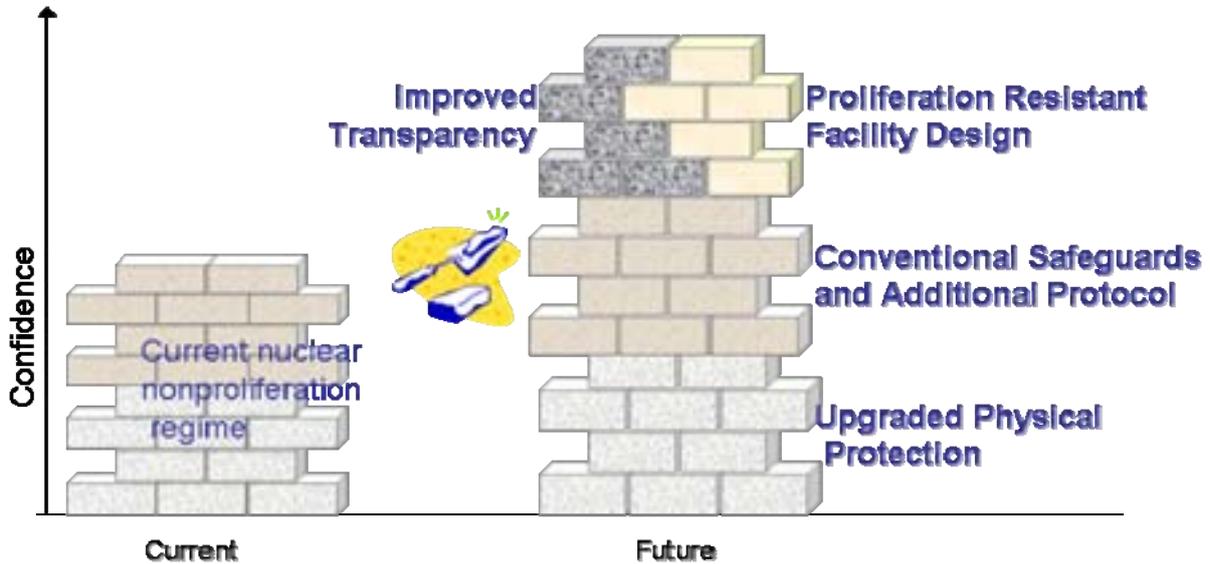


FIG. 4. New requirements applied to the advanced nuclear fuel cycle.

6. Expected New Role of IAEA

For future advanced nuclear fuel cycle facilities, several additional requirements will be applied that create a more robust regime and leads to a greater level of confidence. Activities to certify and maintain high levels of “nuclear material control and accountancy,” “transparency,” and “proliferation resistance” by international authorities will be important. Certification by the IAEA that facilities are maintained in compliance with these requirements will help assure that the proliferation risk has been reduced. As a result, confidence increases and lower proliferation risk could drastically reduce IAEA’s inspection efforts needed for future large-scale advanced nuclear fuel cycle facilities.

Expected new roles of IAEA for future advanced nuclear fuel cycle will include:

- ✓ Evaluation of a State System of Accounting for and Control of Nuclear Material (SSAC);
- ✓ Activities to certify and maintain the compliance to MNA or other guidelines related to transparency;
- ✓ Evaluation of proliferation resistant facility design to certify and maintain adequate levels of proliferation resistance;
- ✓ Activities to certify and maintain adequate levels of physical protection; and
- ✓ Small, more efficient verification activities by IAEA to detect diversion.

Increased confidence in nonproliferation and the peaceful uses of nuclear energy will be a result of the expected new role of IAEA in the advanced nuclear fuel cycle. This increase in confidence is depicted in Figure 5.

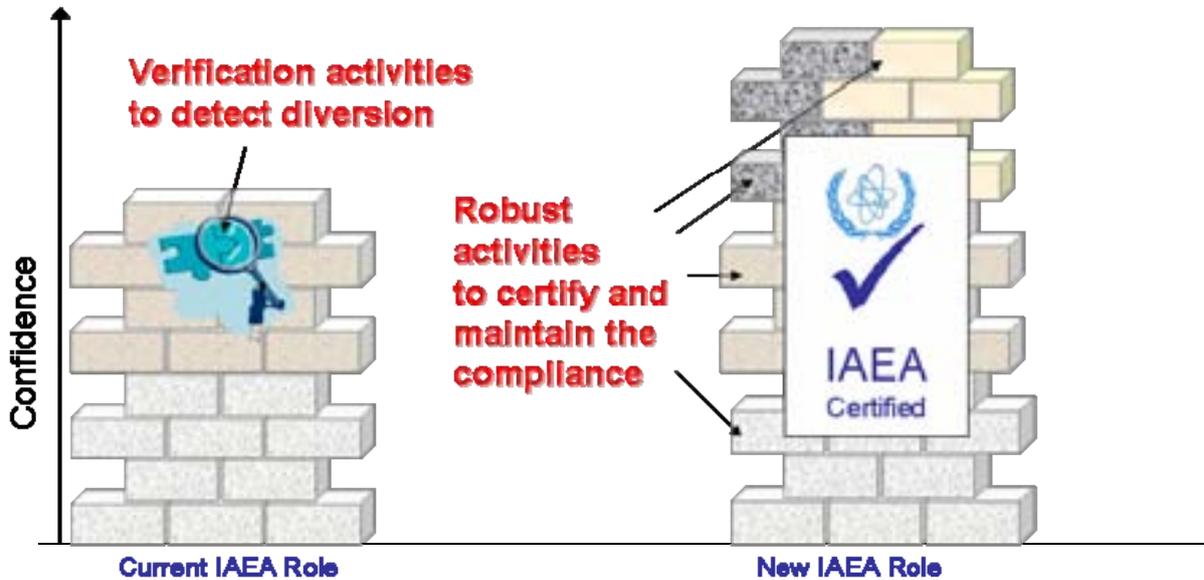


FIG. 5. Increased confidence as a result of the expected new role of the IAEA for the advanced nuclear fuel cycle.

7. Conclusion

Based on current discussions and efforts to improve safeguards, transparency, proliferation resistance, and physical protection, nuclear nonproliferation requirements for future advanced nuclear fuel cycles were studied. The additional requirements applied to the future facilities will lead to greater levels of confidence.

Activities to certify and maintain high levels of “nuclear material control and accountancy,” “transparency,” and “proliferation resistance” by international authorities will be important. Proposed new roles for the IAEA will improve the reliability of nuclear material control and accountancy within a country and decrease proliferation risk overall. As a result, high reliability will increase the international confidence in peaceful nuclear programs and lowered proliferation risk. These, could drastically reduce IAEA’s inspection effort for future, large-scale advanced nuclear fuel cycle facilities and allows in turn, effective use of inspection resources to strengthen the international nonproliferation regime.

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Giving transparency concepts a face-lift: Bridging the gap between old and new

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Presented by V. Cleary

1. Introduction

In the past, the term “transparency” has meant monitoring fuel handling activities through the use of devices such as video cameras. However, given a rapidly increasing need for power generation and an increased automation in fuel handling capabilities at nuclear facilities, current transparency techniques are no longer an efficient means of building confidence in peaceful use. Often, inspections that occur at nuclear facilities take months to assess proliferation potential, and provide no feedback to facilities or other involved parties.

2. Why should transparency be re-defined?

To support a proliferation resistant fuel cycle, the following tasks must be performed:

1. Increase confidence among nations and regulatory agencies that nuclear materials are used in a peaceful manner.
2. Design a system to support non-proliferation efforts during and following global deployment of nuclear power.
3. Optimize time required for inspections.
4. Optimize the cost involved with inspections.
5. Better enforce the current regulations (and agreements among nations) regarding the nuclear fuel cycle.

As stated previously, current “transparent” techniques cannot capably perform these tasks. A redeveloped transparency system could perform all of the tasks listed above, but must include the following capabilities:

1. The transparency system must operate in real-time.
2. Must provide a final quantitative assessment of proliferation-risk.
3. Must utilize plant process and design data.
4. Must utilize declared plant processes.
5. Must have a secure link among the facility and authorized parties.

3. Real-time analysis of proliferation risk

Ultimately, proliferation risk can be defined as a function of the following:

- Material attractiveness: obtained from quantification of factors relating to proliferation of a specific material.
- The static (baseline) risk: obtained from standard process and monitor information. This value is calculated from **expected signals** from a particular process, *declared* plant process information,

and plant design information. This is considered the normal operational risk, or the risk level that can be reasonably expected.

- The dynamic (changing) risk: obtained from real-time processes and measurements. This value is calculated from **observed signals** from a particular process, plant process information, and plant design information. This is considered the real-time risk obtained from observed deviations to expected signals.

During normal plant operations (no proliferation activities), after accounting for material attractiveness, the static risk will equal the dynamic risk.

By continuously comparing these values, a numeric value for facility proliferation risk can be determined. When the proliferation risk is acceptable to all parties involved, and there is a method to continuously calculate and monitor the dynamic risk, the system can be considered transparent.

4. Confidence building

In the current world political climate, there has been extensive debate regarding the tasks listed above. Current transparency techniques involve recording fuel handling activities at nuclear facilities, and then weeks or months later, reviewing the tapes and comparing them to declared activities. This process can be slow, as well as subjective.

To increase confidence that nuclear power is used for peaceful purposes, transparency techniques must allow for a real-time assessment of proliferation risk as described above. New (innovative) power plants and fuel cycles have increased automation capabilities, which provides a wealth of facility process data. By designing a transparency system that continuously monitors *observed* plant process data and compares this to *expected* declared activities, a rapid assessment of proliferation risk can be made. When an undeclared process is performed, the real time proliferation risk is increased.

To further increase confidence, a transparency system can be constructed such that only authorized regulatory parties would have access to sensitive information.

5. Optimizing inspection time and cost

Through use of a transparency system which analyzes automated processes at a nuclear facility, the time required to assess proliferation risk is cut dramatically. Currently, an inspection can take months to complete. A real-time transparency system could produce a numeric value of proliferation risk on a continuous basis, which could be monitored from off-site.

Additionally, the overall cost associated with determining proliferation risk at a facility is decreased by use of transparency techniques. While equipment costs may increase, the manpower associated with the assessment decreases.

6. Conclusion

It is important to note that this research is not intended to replace existing IAEA protocols. It is expected that augmentation of current transparency ideology will support the mission of the IAEA to ensure safe and peaceful use of nuclear technology.

The intent of this research is to develop a transparency system capable of continuously assessing proliferation risk. A continuous, or real-time, analysis is important due to the speed at which proliferation can occur.

By incorporating new ideas into fuel cycle transparency, increased confidence and optimized resources can be utilized to transfer nuclear technology to developing nations, optimize inspections, and enforce agreements.

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Nuclear fuel cycle transparency: An approach to support the global deployment of nuclear power

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1. Introduction

The increasing global demand for energy has created a renewed interest in the expansion of nuclear power. In order to create secure, sustainable and independent energy resources, many countries are considering complete fuel cycles that allow for the initial use of fissionable uranium in light water reactors and move toward breeder cycles when uranium resources become limited. These fuel cycles give rise to increased safety and proliferation concerns that have initiated a new approach to the global deployment of nuclear technology. The new approach is focused on building trust relationships between the nuclear operating country and other countries that are concerned about that country's operations. Transparency concepts can help build these trust relationships through applied technology and sound analysis techniques.

2. Nuclear fuel cycle transparency

The term “transparency” is used in many different applications. In the context of the nuclear fuel cycle, we define it as:

“...a high-level concept, defined as a confidence building approach among political entities, possibly in support of multi-lateral agreements, to ensure civilian nuclear facilities are not being used for the development of nuclear weapons. Additionally, nuclear fuel cycle 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 the parties involved feel that the *proliferation risk* is at an acceptable level. For this to occur, proliferation risk should be monitored in a continuous fashion”[1].

Nuclear Fuel Cycle Transparency can be further categorized into 4 accumulating levels:

1. Bilateral or multilateral agreements on the operation, inspection, and verification of nuclear operations within a host country.
2. Added surveillance and remote monitoring of nuclear operations usually at random or without notification.
3. Direct monitoring of nuclear operations in real time.
4. Ability to remotely secure and inhibit operations.

These highest levels of transparency imply multilateral control of nuclear facilities and processes.

A framework for Nuclear Fuel Cycle Transparency has been developed that is shown conceptually in Figure 1. This framework provides a basis for implementing the top 3 levels of transparency. This

framework is proposed to be demonstrated at level 3 at the training facility for the Monju Prototype Fast Breeder Reactor in Japan.

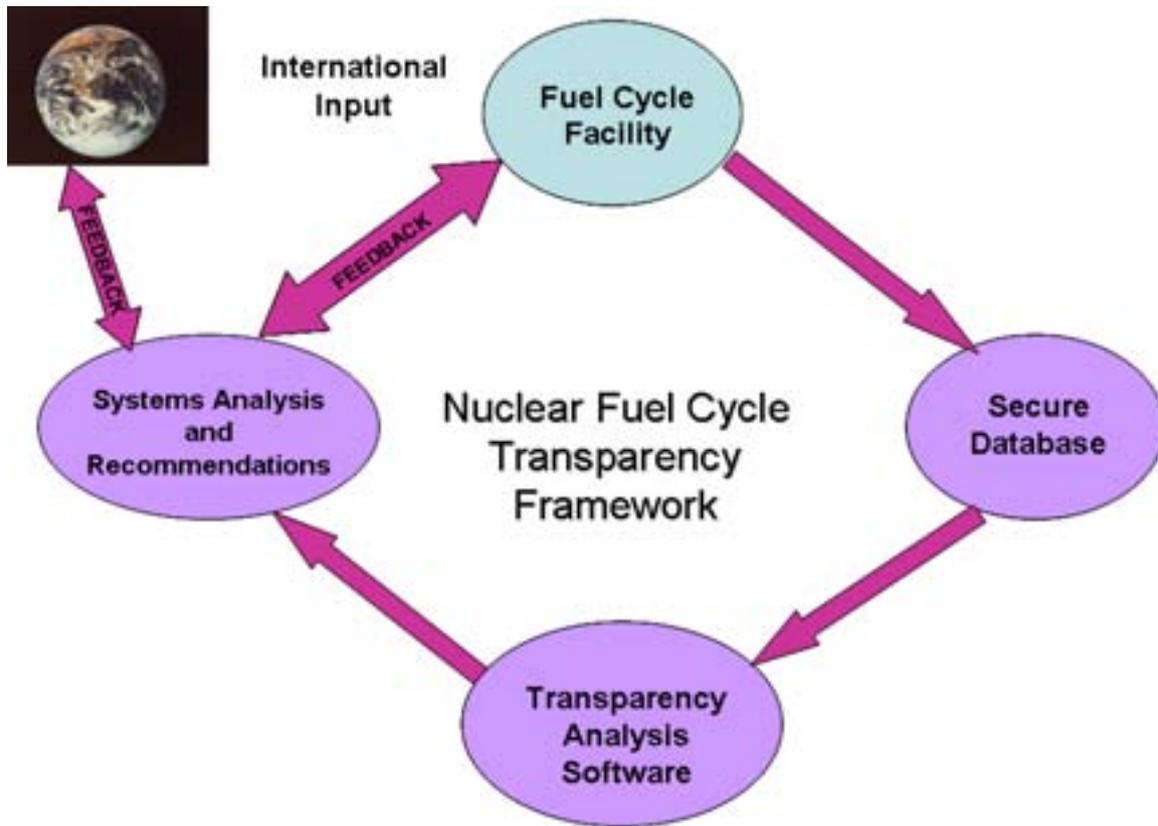


FIG. 1. A conceptual framework for Nuclear Fuel Cycle Transparency.

3. Achieving higher levels of transparency

This control, if possible or useful, needs to be carefully studied and can be applied to several areas of concern for nuclear operations: nonproliferation, safety, and environmental monitoring. If these increasing levels of control are applied as additional equipment and oversight, the cost is likely to be prohibitive.

We suggest that a new paradigm be considered for the development of deployable nuclear technology. This paradigm would have the highest levels of transparency built into all aspects of the nuclear fuel cycle by design rather than “added-on” after the facilities are designed and deployed.

This approach offers the possibility to:

1. Take advantage of the increasing level of process automation in new fuel cycle facility designs.
2. Define a need for a uniform standard for transparency that is independent of the hosting country.
3. Provide a means for uniformly gathering verified data for building trust relationships between countries.

To implement this approach will, at a minimum, require:

1. New technology to provide secured plant process data.
2. Secure information systems to distribute the data without providing data to potential adversaries.
3. Analysis tools to digest the data provided into a uniform standard.
4. Approaches to secure material and processes safely and securely.

The new paradigm will allow nuclear suppliers a means of designing systems that are demonstrably safe, proliferation resistant, and physically protected. However, the highest levels of transparency place new design requirements on the nuclear suppliers that have not been previously considered. When implemented, a complete package will be available that can be reviewed internationally, can be qualified for international deployment, and will have the confidence that every aspect of concern has been addressed by the design.

4. Preliminary conclusions

We believe that higher levels of transparency, when engineered into nuclear facilities, can provide a level of assurance for nations to verify that civilian nuclear facilities are being used safely and that the materials are being legitimately used for civilian purposes. Further, advanced approaches to transparency can be used to secure nuclear facilities and materials in the event of detected misuse, possibly before the material is diverted or a severe accident occurs.

This paper will further explore a new paradigm for transparency and will provide some examples of how it may be implemented in nuclear fuel cycle facilities.

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Implementing a new reporting system to improve the security and the processing of accounting information

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Abstract. With the objective of improving the security and speed up the processing of the accounting information received at the Nuclear Regulatory Authority, it was established that the accounting reports must be made in text files (ASCII format) and must be sent using secure e-mail. This new procedure enters into force for all Argentine facilities in May 2006. Besides, during the ABACC/IAEA joint inspection, some accounting information (inventories changes updating and itemized list) is required by the agencies in text files in a format previously established to be used in its auditing software. Consequently also from May 2006, it is mandatory for the operators to prepare these files and give them to the international agency inspectors. In order to help the operators with these tasks, a software was developed to automate the generation of the mentioned files and, at the same time, to diminish the quantity of errors contained in them. The software was installed in all Argentina's installation last February with satisfactory results.

1. Introduction

In December 1991, the Agreement between Brazil and Argentina for the Exclusively Peaceful Use of Nuclear Energy entered into force [1]. The agreement, determines the control over all nuclear materials in all nuclear activities in both countries. To verify this commitment, the Common System of Accounting and Control of Nuclear Materials (SCCC) was established and the Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials (ABACC) was created to administer and apply the SCCC [2].

The Quadripartite Agreement (INFCIRC/435), signed among Argentina, Brazil, the International Atomic Energy Agency (IAEA) and Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials (ABACC), entered into force on 4 March 1994 [3]. It is essentially a comprehensive safeguards agreement, based on the INFCIRC/153 model with some modifications due to the existence and the role of ABACC as a binational organization.

2. Accounting reports

According to the provisions of the SCCC, the National Authority (NA)¹ must send the Inventory Change Reports (ICRs) to ABACC within 20 days of the end of the month in which the inventory change occurred. The Argentine National Regulations established that the Facility Operators must send the ICRs to the NA, within 5 working days of the end of the month in which the inventory change occurred. The Material Balance Reports (MBRs) and Physical Inventory Listings (PILs) have the same time limits but with respect to the completion of the Physical Inventory Taking (PIT). This means that NA has, in the best of the cases, a maximum of 10 working days to input, process, evaluate and send the accounting reports to ABACC without introducing delay. It must be considered that the report evaluation

¹ In Argentina, this roll is assumed by Safeguards Division of the Nuclear Regulatory Authority

includes the detection of mistakes in order to make, whenever possible and with the approval of the respective operators, the necessary correction before sending them to ABACC.

Until the first semester 1996 the reports were sent to ABACC in hard copy by diplomatic mail. In the second semester of 1996 it was implemented the SCMN software (**Sistema de Control de Material Nuclear**) with the capabilities of error detection and the generation of the reports in ASCII format. Consequently the reports began to be sent in diskette until the ending 1999 and since then by secure e-mail [4].

Although the SCMN software has the capacity to input the accounting reports in ASCII format, the facilities operators continue sending them by mail or fax. Consequently a significant part of the 10 working days that NA have for sending the reports to ABACC is used to enter the data manually with the possibility of introducing transcription errors.

Also it is important to point out that the level of security of the accounting reports sent to the NA (mail or fax) is considerably inferior to the one used to transmit the same information to ABACC (diplomatic mail or secure e-mail).

With the objective to solve these disadvantages, in June of 2005, it was decided by the NA:

- to receive the accountings reports only in text files (ASCII formats) according the structure established in the Code 10 of General Procedures of the INFCIRC 435 (the same format that is utilized to send the reports to ABACC),
- to receive by secure mail the text files with the objective to improve its level of security and to make it compatible with the used one to send the same information to ABACC. This means, that the NA must receive the accounting reports only by email encrypted and with the operator's digital signature, using the concept of Public Key Cryptography [5] and the Certificate Authority of the Argentine National Government [6]

To make this points mandatory to the facility operators an official safeguards requirement (RQ SV 01/06) was generated, which was distributed in February to be applied from. May , 2006.

3. Auditing files

In June 1994, the inspections at the Argentine Facilities began under the Quadripartite Agreement. The inspections have been carried out jointly by ABACC and the IAEA including the records auditing activity applying each organization their own procedures.

In 1999 ABACC developed the SARA (**S**oftware de **A**uditoria de **R**egistros de **ABACC**) software for records auditing to be utilized in field during the inspections using a notebook and started to be used in routine basis in the beginning of 2000 [7,8]. By the ending of 2000, ABACC and IAEA decided introduce some modifications in the SARA software in order to implement a joint system for auditing accounting records during inspections. The new **S**oftware for **J**oint **A**uditing of **R**ecords (SJAR) was fully implemented on September of 2004 and is utilized in all joint inspection carried out since then by the agencies [9].

The SARA/SJAR offer the possibility to introduce some accounting information (inventories changes updating and list of items) in text files in order to reduce significantly the time

consumed in this activity and diminish the number of errors due to manually inputting data. In the beginning of 2000, the NA agreed to induce the operators to provide these files in a diskette to ABACC/IAEA inspectors, but only a few installations have been able to give these archives since then.

By June 2005, the NA considered that it is very convenient to provide the auditing files in text format, therefore, it was decided to turn the supply of such files mandatory for the facility operators. Consequently, an official safeguards requirement (RQ SV 02/06) was generated and distributed in February to be applied from May 2006.

4. Implementation

As it was said in the preceding sections, in June 2005 it was decided to implement the requirements RQ SV01/06 and RQ SV02/06 to be effective as of May 2006. The time between the decision and the effective put into force of the requirements is related to the fact that it was decided to use that period to make a software to provide the operators with a tool for the generation of the files necessary to fulfill the RQs.

The ICAIFE software (**I**nformes **C**ontables **A**rchivos **I**nspección en **F**ormato **E**lectrónico) was developed between June and October of 2005 and was fully implemented using Fox Pro. The main function of ICAIFE is to generate the accountings reports (ICRs, MBRs, PILs and Concise Notes) and the auditing files (inventories changes updating and itemized list) in the appropriate ASCII formats from the operators general ledgers. The ICAIFE main characteristics are:

- to define the number and period of the reports,
- to check all the information entered, for example, MBA codes, sign of element/isotope weight, enrichment, etc.,
- to check if the input fields fulfill the Code 10,
- to generate the MBRs automatically,
- to generate the PILs automatically from an itemized list,
- to display the inventory for comparison with the declared one in the operator ledger,
- to calculate the value of physical inventory for each element,
- to check the relationship between different fields to assure that the numeric information entered is correct, for example, the correlation between gross, net and tare weights,
- to avoid duplication input of equivalent information.

In November 2005, two workshops for facility operators were made with the purpose of:

- clarifying the requirements RQs SV 01-02/06,
- presenting the ICAIFE software giving training on its use,
- explaining how to send secure e-mails and how to obtain the public and private keys from Certificate Authority of the Argentine National Government.

In February 2006, the NA completed the installation of the ICAIFE in all Argentine facilities and started a test period. During the workshops and the test period the operators made comments and suggestions that lead to further improvements in the software.

It is important to point out:

- all the accounting reports are received in text files via e-mail from March 2006 with a significant reduction of the errors,
- the auditing files are being given to the ABACC/IAEA inspectors from July 2006,
- the use of secure e-mail was not completely fulfill (approximately 30 % of the facilities were not able to implement it but the NA consider that they can resolve this issue by the end of October).

5. Conclusion

The development of ICAIFE software was possible in a considerable short time and has been permanently improved taking into account the comments and suggestion of the facility operators. The software developed has shown to be highly satisfactory to fulfill the objective of the requirements RQ SV01/06 and RQ SV02/06

More than 70 % of the accounting reports are received by secure e-mails increasing significantly their security level. It is expected than in October 2006, all the reports will be received by this mean.

The reception of the accounting reports in text files (ASCII format) make it possible to reduce the time used in processing and evaluating them, from some days to some hours, allowing the NA to send the reports to ABACC without introduce any delay.

The supply of the auditing files to the ABACC/IAEA inspectors has shown to be effective to reduce significantly the time consumed in the auditing activity during the inspections.

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Safeguards approach to a new nuclear power plant in Argentina

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Abstract. Atucha II is an Argentinean Nuclear Power Plant (NPP) under construction which is planned to be finished and operating in about four years. To understand the progress it would imply to Argentina the construction of a new NPP, it is worthy to make a general design comparison between Atucha I and Atucha II as they both are “On Load Pressured Heavy Water Reactor (PHWR)”. When a Comprehensive Safeguards Agreement entered into force in Argentina, in April 1994 some problems were faced at Atucha I due to the application of IAEA Safeguards Criteria 1991-1995. This reactor had partially attained the quantity component of the inspection goal because with the safeguard scheme available the scenario of unrecorded Pu production was not fully covered. To solve this issue it was necessary to acquire a system to qualitatively measure irradiated fuels moved between the reactor and the pond and vice versa. The solution to this problem was finally found and implemented in 2003. Based on the hard experience described and considering the importance to think in advance, the safeguards approach at Atucha II is being started to be considered from the construction phase in order to put in place every single device to meet safeguards objectives. The main purpose of the Argentinean National Authority on this new NPP is to view the possible alternatives aiming at arriving at the best conclusion to comply with the IAEA and ABACC requirements without falling in undesirable constructive modifications to the plant once it has started operation.

1. Introduction

Atucha II is placed in the city of Lima, to the north of Buenos Aires, and next to Atucha I, and it is expected to generate 745 Megawattios. 80 % of the construction has been done and the reactor and the steam generator are installed. The agreement for the construction of Atucha II was signed in 1980 and hard work was conducted up to 1994, then the construction was stopped. The last big construction step was performed in 1999, when the pressurized vessel was placed. Atucha II is expected to have a useful life of 40 years.

It is expected that Atucha II will be ready to start to produce 5% of the electricity used in the country. Experts and Professionals have been in charge during last ten years of maintaining properly the parts and plans of Atucha II. The parts and pieces, such as electronic circuits and hydraulic devices, have been stored for many years in 18 heated warehouses to avoid humidity and so corrosion of the parts. It is estimated that about 93% of the NPP components are gathered between what is stored in the warehouses and what has been installed in the NPP.

Taking into account the future introduction of this NPP to the fuel cycle of Argentina, it is foreseen that it would be necessary to increase the production of Uranium and Heavy Water. For the acquisition of Uranium, the possibilities could be to import or it is also feasible the reactivation of “Sierra Partida” Mine, in Mendoza Province, and “Cerro Solo”, in Chubut Province. Besides, it is very important to consider the necessity of acquiring Heavy Water that could be produced at the Heavy Water Industrial Plant. In this sense it is also necessary the review of the design and real capacities for the Fuel Fabrication Plant (FFP) and the Conversion Plant.

In relation to the FFP, nowadays, there are two separate lines one for Low Enriched Uranium (LEU) for Atucha I fuel assemblies and the other one for Natural Uranium for Candú fuel bundles. Before Atucha II

starts operation, it would be necessary to count with a Natural Uranium line for its fuel assemblies. By now, the impact on the increase in the future production for the FFP is being analyzed, and two possibilities arose, one of them is the reactivation of the Natural Uranium production line used for Atucha I assemblies, and the other one is the creation of a completely new line depending on the fuel assembly design for Atucha II, which is still under consideration.

In connection to the Conversion Plant at Cordoba Province, it would be necessary to analyze the production capacity as probably the expected production would be doubled or at least increased abruptly.

Now the political decision was made and Atucha II will be finished, if possible, in four years time. But this long postponed decision implies, by sure, the revision, updating or adjustment of all the steps of the current fuel cycle in Argentina to deal with an increasing demand.

2. NPPs in Argentina

2.1. Embalse NPP

Embalse NPP works commercially since the 20th of January 1984. It is placed by the South cost of the Embalse of Tercero River, nearby Embalse in Cordoba Province. The designer and constructor were Atomic Energy of Canada Limited (AECL) from Canadá e Italmimpianti from Italy. Embalse is a PHWR, with heavy water and 380 channels to place Natural Uranium fuel bundles. The Brut Power is of 648 Mwe. A by product of the NPP is the production of Co 60 used in nuclear medicine and in industry and a big amount of this production is exported.

Experience has been gained at this facility in the verification by gross defects of irradiated fuel bundles. Many advantages in the use of the core discharge monitor are recognized, and eventually, it could be considered the use of it at Atucha II. The change in the channels (back fitting) for this NPP is foreseen for next year to increase the useful life.

It works commercially since 24th of June, 1974 and it was the first NPP installed in Latin America. It is situated by the Paraná de las Palmas River in Buenos Aires Province at 100 km from Buenos Aires city and its main designer was SIEMENS.

Atucha I NPP is an On Load Pressured Heavy Water Reactor (PHWR) built in the 70's. The power of this NPP is 357 Mwe. The reactor started operation with natural uranium fuels but now it works with low enriched uranium fuel assemblies (0,85% U^{235}). The core grid capacity is of 252 fuels. The reactor is refrigerated and moderated by D₂O. In this Plant there are two houses with a total of six pools for storing the spent fuels and two maneuvering pools, one in each house. The re-fuelling frequency is of 0,72 fuels per day when the reactor is working at high power. There is only one channel between the reactor building and the maneuvering pools of the two houses and the flow of fuels (fresh, burnt up and semi burnt up) is through it.

In the ponds, the fuels are stored in two layers and in hangers of different design for the upper and the lower layer. Basically, the differences consist of constructive dimensions. Due to that reason, the fuels are not aligned and so the fact of placing a detector between the fuels in the lower layer turned into a difficult challenge at the time of re- verification.

In this NPP all the channels have been changed to extend the useful life of the facility.

	Embalse NPP	Atucha I NPP
Reactor Type	Candu Type	Siemens - PHWR
Thermal Power	2,109 MWt	1,179 MWt
Electrical Brut Power	648 Mwe	357 Mwe
Moderator and Refrigerator	Heavy Water	Heavy Water
Fuel	Natural Uranium	LEU (0,85%)
Vapour Generator	Four	Two
Turbine	1 step of high pressure, three steps of low pressure	1 step of high pressure, three steps of low pressure
Electric Generator	Four poles 22 Kv, 50Hz	Two poles 21 Kv, 50Hz

2.3 Atucha II NPP

Atucha II is the third NPP in the country of a SIEMENS design. This NPP is based on Atucha I prototype, but with a brut power higher than 745 Mwe. The fact that it is placed next to Atucha I, will help in the use of big part of the infrastructure and services.

	Atucha II NPP
Reactor Type	Siemens - PHWR
Thermal Power	2,175 MWt
Electrical Brut Power	745 Mwe
Moderator and Refrigerator	Heavy Water
Fuel	Natural Uranium
Vapour Generator	Two
Turbine	1 step of high pressure, two steps of low pressure
Electric Generator	Four poles 21 Kv, 50Hz

3. Experience gained at Atucha I

According to IAEA Safeguards Criteria issued in 1991 and in particular since 1995, Atucha I had only partially attained one of the technical inspection goals due to the requirement of having a fuel flow monitor to count and verify by gross defect the fuels discharged from the reactor core before being placed under IAEA and ABACC surveillance at the storage ponds. This criteria basically based on Candú reactor model, could not be easily extrapolated to On Load Reactors types (OLRs), in particular for those in operation.

At that moment, for ARN, the goal was to find a suitable method to fulfill this requirement minimizing the intrusiveness in normal operation of the facility and avoiding substantive construction modifications together with appropriate NDA techniques to verify the spent fuel inventory at the ponds before the VIFM were left working. This method could also be used in case of re verification purposes when C&S failure occurred.

An alternative to the installation of flow monitors to fulfill safeguards requirement could be verifying by gross defect with medium detection probability the spent fuel inventory at every interim inspection. This would have implied to perform physical inventory verifications every three months and therefore the inspection effort would have been enormous. That was the main reason for looking for technical solutions to allow maintaining the continuity of knowledge of verified items.

4. Description of the adopted technical solution at Atucha I:

For the verification by gross defects of fuel flow between the reactor and the ponds the fuel monitor installed at Atucha I NPP included a subsystem of semi submergible cameras to avoid tampering of the detectors.

For measuring the spent fuels difficult to access stored in the lower layer and for those with long cooling time, a spent fuel neutron counter (SFNC) of suitable dimensions was used.

It was projected the installation of the fuel monitoring system during the first middle of the year 2003 and towards the end of that year the testing period was finished. The objectives complied during the year 2003 were to implement a suitable system to complete the safeguards approach for this NPP; and to select complementary methods to the Improved Cerenkov Viewing Device (ICVD) to re-verify the inventory at the storage ponds.

The adaptation of an “in operation facility” to Safeguards Criteria afterwards developed, was a tough task that could be successfully completed.

4.1. Description of the selected monitoring system at Atucha I

The unattended system installed consisted of gross gamma detectors and a surveillance system semi submerged at the pond, focusing to the transfer channel. The detectors are positioned in such a way as to register the direction of each transfer. The underwater cameras record all the movements through the transfer channel into and out of the bay. The digital signals from the gamma detectors and the images from the surveillance cameras are sent to a server that would process the data and generate reports and images for a set safeguards period. This server also has redundant functions for power supply and data storage.

The National Authority is conscious that the detectors could not be installed at strategic points to avoid tampering; instead the positions were selected according to operating limitations. The failure probability of the whole system is not low as the flow monitor and the semi submergible cameras are two independent sub systems. If the underwater surveillance fails it can not be assured that not tampering of the detectors were made and If the detectors fail the verification by gross defect of the fuels also fails and so re verification of the pools is necessary.

Now, during the construction phase, in Atucha II, it is possible to choose the more convenient points to install the detectors for the flow monitor without operational or construction limitations. Now, it is possible to install the flow monitors in the transfer channel with enough inaccessibility to consider tampering unviable before the start of operation. In case of failure of the detectors, a potential solution could be considered for the next stop of the NPP, where the flow monitor system could be repaired or replaced.

4.2 Description of the NDA techniques selected to re-verify the spent fuel inventory at Atucha I

The spent fuels stored in the lower layer of the ponds were mostly inaccessible for in situ measurements because of the limited space between the assemblies. Another feature that was considered was the long cooling time of many of the fuels with low burn up that prevented from using the ICVD. Additionally, isolation of the fuels for measurement was extremely difficult due to the number of assemblies and the difficulties in moving and re-arranging the fuels due to safety and operational reasons. Due to the fact that the hangers were no aligned between the lower and upper layer of the ponds a specific device was designed and constructed to drive the neutron detector towards the randomly selected fuel.

Finally, for the re verification of the spent fuels in the ponds was defined, after subsequent field trials, the use of a SFNC and, when applicable, to the irradiated fuel assemblies with low cooling time, an ICVD .

4.2.2 Spent fuel neutron counter (SFNC)

A prototype neutron detector system, which employed a fission chamber (12,7 cm active length, 2,54 cm diameter) and a preamplifier moderated by a polyethylene cylinder (3,5 cm thick polyethylene) was fabricated to be used at Atucha I. The system was designed to work with MMCA and software widely used by the IAEA. The SFNC was used to collect neutron signals from the assemblies with burn ups of 5000 to 8000 MWD/TU. The measurement results, while limited in the number of data points, showed that the calibration curve was sufficiently linear for the purpose. The detector system was placed 2.4 m below the top of the fuel assemblies and this vertical position was maintained for measurements.

Now in the construction phase of Atucha II, it is possible to design a suitable hanger to optimize the storage capacity allowing alignment of the two layers maintaining optimum storage capacity and minimizing difficulties in the introduction of detectors in case re-verification is required. The National Authority is interacting with the Operator to introduce modifications in the pool design to resolve the problem that exists at the prototype NPP, tempting at optimizing the safeguards activities. In that sense, the hangers should be designed allowing seals to be applied leaving difficult to access fuels under dual, containment and surveillance system.

5. Foreseen safeguards measures at Atucha II NPP

According to the strengthening of safeguards evolution in the world, it is necessary to think in advance that when Atucha II starts operation in about 4 – 5 years, Integrated Safeguards would be in force in Argentina. In that sense, it should be considered in the safeguards approach, the application of an announced inspection regime, remote monitoring systems, state of health transmission, etc.

According to the current safeguards criteria, it is also foreseen the installation of a flow monitor to verify by gross defect irradiated fuel assemblies before they turned into difficult to access nuclear material in the ponds. At the beginning of the NPP operation it is not expected to have a great amount of irradiated fuels stored and so the verification effort would not be very high but high in comparison with the spent fuels generated in a light water reactor type. It is necessary to think in advance, as the spent fuels stored increase the inventory during normal operation, how to minimize re verification efforts. Besides, it is also necessary to think about surveillance failure. Due to the experience gained in Atucha I, conditions such as minimize the re verification effort in case of surveillance failure is an important issue for all the inspectorates.

The re verification effort could be minimize if, for instance, the verified spent fuels are kept under seals. So it turns into a necessity to work in the suitable design of the hangers to include this technical capability. This could not be implemented at Atucha I due to the fact that, the little space in the pools did not allow the movement of the vast majority of the fuels and so no change of the hangers was possible. Besides, the design of hangers that could be aligned would facilitate the introduction of the detectors in case of conventional surveillance failure.

However, in case of failure of the detectors of the flow monitor system, the measures proposed by the National Authority regarding storage of spent fuel assemblies, would allow that only a portion of unverified spent fuels would be subject to verification by gross defect. Consequently, it is necessary the adoption of a storage strategy by subdividing the inventory at the pond by design, while in the storage process. In addition, a mailbox system complemented with an announced inspection regime could be

approved to be implemented as soon as the failure is detected to last up to the repair of the failure. This also will conduct to the re verification of a minimum amount of irradiated fuels reducing inspection efforts. Identical analyze could be made considering the case of surveillance system failure at the pool house.

Atucha II is under the construction phase, and so it is also possible to think about the possibility of having a dried storage built similar to the existing one in Embalse NPP and complementary to the wet storage. For Atucha II, the design of the silos, baskets, transportation flasks, welding cells, etc. are strongly influenced by the size of the spent fuel assemblies and it should be considered in advance.

6. Conclusion

- It is very important to think about the safeguards measures for Atucha II NPP during the construction phase to minimize operational intrusiveness and construction modifications, once the facility is in operation.
- It must be consider to minimize and easy by design, the activities to be performed at routine or special inspections (eg. re verification inspections)
- In case of failure of fuels flow detectors or surveillance it should be considered that the recovering of the continuity of the knowledge should be the less possible intrusive to the normal operation of the NPP. It is worthy considering a storage strategy to maintain segregated, by design, the unverified or out of surveillance irradiated fuel assemblies, aiming at reducing the re verification effort.
- During the construction phase, it is necessary to study the design of the hangers to include the possibility by design of applying seals. This will lead to dual measures on difficult to access nuclear material.
- According to the experience gained in finding a technical solution at Atucha I, co-operation between ABACC, IAEA, ARN and also the Operator, brings positive results and it is worthy to start simultaneous work during the construction phase and as soon as possible.
- As it is possible that in 4-5 years time Integrated Safeguards wouldl be in force, in the safeguards measure to be applied to this new NPP, the new technologies would have impact to meet safeguards goals, like transmission of state of health, remote monitoring application or unannounced inspection regime complemented with a mail box system, etc.

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Determination of uranium contents and isotopic compositions in estuarine sediment standard reference material candidates by isotope dilution thermal ionization mass spectrometry^{*}

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Abstract. Environmental sampling as one of the safeguards strengthening measures has become an essential tool for detecting undeclared nuclear material or activities, which technique has been introduced by the IAEA publication [1]. Many laboratories have developed similar analytical methods for analysis of environmental samples [2] and become the members of network analytical laboratories for nuclear safeguards. However, the IAEA environmental sampling is mainly focusing on swipe sampling. Additional techniques for such purpose are encouraged for improving detection capabilities. Sediment is an optional environmental carrier. Positive findings in sediments collected at various places from stream can be traced upstream, establishing a record of what has come down stream [3], providing additional information to detect possible nuclear signature. Estuary is the down stream endpoint, which sediment may keep more information originated from upstream activities. To guarantee analytical quality for uranium and other elements in estuarine sediments, three estuarine sediment standard reference material candidates have been produced. This paper gives the methodology for analysis of uranium contents and isotopic compositions in those SRM candidates by isotope dilution thermal ionization mass spectrometry (ID-TIMS).

The materials were taken from the Yellow River Estuary and the SRM candidates are prepared under the guidance of document for preparation of national standard reference materials.

Chemical procedure for the samples is carried out in our Class-100 clean laboratory (designed by SINO-USA Joint LabTech Ltd). High pure water and acids, microwave oven (MILESTONE, Ethos PRO) are used for sample digestion. Weigh (nearest 0.001g) amount of a SRM sample (about 0.2 g) and HEU spike (estimated amount based on the ID function) into the same microwave vessel for mix uranium isotopic ratio measurement. Weigh (nearest 0.001g) another fraction of the SRM sample (about 0.2 g) into another microwave vessel for original uranium isotopic measurement. Add 10.0 mL HNO₃ to each vessel in a fume hood. Cap the vessel, torque the cap to a proper position, place the vessels in the microwave carousel, irradiate the sample vessels for 10 minutes between temperature 170-180 °C according to the manufacturer's directions and the guidance of EPA 3051. At the end of the microwave program, allow the vessels to cool for a minimum of 5 minutes before removing them from the microwave unit. When the vessels have cooled to room temperature, uncap and vent each vessel in the fume hood. Transfer the samples into the clean PTFE beakers, add 1mL HF and 0.5 mL HClO₄, and place the beakers on the hot-plate for heating till almost dryness. Add 1.0 mL 7 mol/L HNO₃ to each beaker, warm up at 80°C for 2 minutes, and then cool and transfer the samples to the centrifuge tubes for centrifugation at 5000-6000rpm for 10 minutes to clear the supernatant. Load the supernatant of each sample on the micro separation column that is filled with anion exchange resin to remove matrix. 1.0mL deionized water is applied to elute uranium from the column and the final sample is evaporated to dryness for isotopic ratio measurement.

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

D. Guo et al.

Thermal ionization mass spectrometer (TIMS, model Isoprobe-T, GV Instruments Ltd., UK) is applied to determine uranium isotopic ratios. In the measurement of uranium isotopic analysis, SRM U010 solution of $1\mu\text{g}/\mu\text{L}$ is used to calibrate the TIMS and triple filaments (inner Tantalum, centre Rhenium and outer Tantalum) are employed to enhance the stability of ion beams. After cleaning the filaments in a degas device, put the filaments in a loading device and then load $1\mu\text{L}$ SRM U010 solution and $10\mu\text{L}$ treated-sample onto the centre of filaments under the current less than 1A, respectively. Allow the sample to dryness almost totally and then heat the filaments to approximately 1.5A, leave for no more than 5 seconds then turn the current off. The samples are then loaded into the magazine and analyzed in the TIMS instrument. Measure the intensities of uranium isotopes using Faraday cups for ^{235}U and ^{238}U as well as EPT for ^{234}U and ^{236}U . The $^{234}\text{U}/^{238}\text{U}$, $^{235}\text{U}/^{238}\text{U}$ and $^{236}\text{U}/^{238}\text{U}$ ratios are automatically given.

Uranium contents and isotopic compositions with natural uranium origin in 3 SRM candidates are given in Table 1. The detection limit (D.L.) of isotope is given by the abundance sensitivity of 3ppm. The uncertainties of measurement are evaluated at 95% confidence level, indicating the results are quite satisfactory.

ID-TIMS is powerful tool for determination of uranium contents and isotopic compositions in environmental samples in our laboratory. As an example, we determined uranium contents and isotopic compositions in estuarine sediment SRM candidates, in which natural uranium only at environmental level and no detectable ^{236}U are found.

Table 1. Uranium contents and isotopic compositions in 3 SRM candidates.

Sample	U ($\mu\text{g}/\text{g}$)	^{234}U , atom %	^{235}U , atom %	^{236}U , atom %	^{238}U , atom %
HKBW1	2.169 ± 0.009	0.00535 ± 0.00028	0.724 ± 0.028	<D.L.	99.270 ± 0.011
HKBW2	2.261 ± 0.058	0.00505 ± 0.00026	0.727 ± 0.028	<D.L.	99.268 ± 0.011
HKBW3	1.381 ± 0.034	0.00551 ± 0.00029	0.725 ± 0.028	<D.L.	99.269 ± 0.011

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Approaches to strengthen China nuclear material control system

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1. Introduction

In order to ensure the security and lawful use of nuclear material, to protect against theft, loss, unlawful taking of nuclear material and sabotage of nuclear material and nuclear facilities, the *Regulations on Nuclear Materials Control of the People's Republic of China* was issued by the state council in 1987. In 1990, the *Rules for the Implementation of Regulations on Nuclear Materials Control of the People's Republic of China* was promulgated. These documents formed the legal basis of nuclear material control in China.

2. Basic requirements for licensee

As defined in the Regulation, The materials containing the Plutonium-239, Uranium-233 and 235, Tritium and Lithium-6 are nuclear materials to be controlled. licensing system has been adopted. Any organization or individuals intending to use, produce or store nuclear materials is required to apply the license and get the approval from the state competent authorities. The licensees should establish the nuclear material accounting and control system and physical protection system.

To meet the requirements for nuclear material accounting and control, licensee should establish and maintain records and accounts on the quantities of nuclear material in each MBA, establish nuclear material measurement system to provide the quantities of nuclear material including inventory, shipment, receipts and loss etc. conduct physical inventory to determine the quantities of nuclear material present in each MBA periodically, and evaluate MUF. $MUF = BI + A - EI - RL - KL$, MUF: The material unaccounted for, BI: The beginning inventory, A: The additions of material, EI: Ending inventory, R: Material removals, KL: The known loss of nuclear material.

A professional security organization at facility is required to be set up in charge of protecting the security of nuclear material and facilities. The physical protection system should be based on the evaluation of the threat. The principles of detection balance and defence in depth are required to be applied in the physical protection system design. IAEA document INFCIRC/225 Rev.4 is recommended as a reference for reviewing and designing physical protection system of nuclear material and facilities.

3. Inspection

Inspection is a measure in China national nuclear material control system to verify the compliance of licensee and to evaluate the reliability and effectiveness of nuclear material control measures at facilities. China Atomic Energy Agency (CAEA) issued the Rules for Inspection of Nuclear Material Control in

1997, and formulated some inspection procedures. Based on the Rules, Office of Nuclear Material Control is in charge of working out inspection plan, organizing the inspector team, implementing inspection activity, and evaluating and approving the inspection reports.

4. Measures to further strengthen China nuclear material control system

With the development of nuclear energy , Measures to strengthen China nuclear material control system have been developing.

- The regulations is to be revised to meet the new requirement. Some new guidances will be developed. More attention will be paid to the effectiveness of nuclear material control system.
- Advanced assay techniques and software related to nuclear material accounting have been developed.
- Methodologies of threat analysis and vulnerability assessment in physical protection are under developing.
- Verification Measurement used for Inspection such as *γ spectroscopy and neutron counting* are developed. Rules and Procedures on inspection will be revised.
- International cooperation is needed for improving the effectiveness of national nuclear material control system. Two pilot workshop on nuclear material accounting and control at facilities have been held in Beijing, China in May 2004 and May 2006 respectively. In August,2006, a regional training course on physical protection will be held in Beijing, China. In October 2005, an joint US-China integrated nuclear material managemaint technology demonstration was held in Beijing, China.

5. Summary

China has attached great importance to nuclear material control and has established an effective national system to ensure the safety in lawful use of nuclear material, to prevent theft, sabotage, loss, diversion and use of nuclear material without lawful authority.

Study on plutonium isotopic analysis in environmental samples *

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Abstract. A method for determination of plutonium isotope ratio in environmental samples was investigated in our laboratory. Plutonium was separated with complex matrix and purified by using TEVA extraction chromatography resin, and determined by MC-ICP-MS.

The elution of Pu from TEVA column was tested. Plutonium could be eluted from TEVA column effectively by $0.025\text{mol}\cdot\text{l}^{-1}\text{H}_2\text{C}_2\text{O}_4\text{-}0.15\text{mol}\cdot\text{l}^{-1}\text{HNO}_3$ or $10^{-2}\text{mol}\cdot\text{l}^{-1}$ ascorbic acid- $0.2\text{mol}\cdot\text{l}^{-1}\text{HNO}_3$.

For one TEVA column, the decontamination factors of main environmental matrix, such as Na, K, Ca, Al, Fe, Mg, were between $10^4\text{-}10^5$, and the concentration of total cation ions in the final elution was less than $10\text{ }\mu\text{g}\cdot\text{g}^{-1}$, which could be determined by MC-ICP-MS directly. The decontamination of U, Th and Am was also examined, $DF_{\text{U/Pu}}=1.97\times 10^3$, $DF_{\text{Th/Pu}}=7.19\times 10^4$, $DF_{\text{Am/Pu}}=1.52\times 10^4$ were obtained. The chemical recovery of plutonium during the whole process was ~30%.

Because of the tailing of ^{238}U and the formation of $^{238}\text{UH}^+$ while the determination of plutonium isotope by MS, it was not enough for the decontamination of uranium by using one TEVA column. A UTEVA column was needed before the solution loading in the TEVA resin. The decontamination factor of uranium could reach 1.33×10^6 .

The detection limit of Pu by MC-ICP-MS was about $2.5\text{ fg}\cdot\text{ml}^{-1}$. The influence of U on the determination of plutonium isotope ratio was also examined. For the solution containing less than $1\text{ ng}\cdot\text{g}^{-1}$ uranium, Aridus sample introduction system was adopted to improve the sensitivity. Meinhard PFA nebulizer was used for solutions of high uranium concentration ($1\text{-}500\text{ ng}\cdot\text{g}^{-1}$). The results show that, the $^{238}\text{UH}^+/\text{U}^+$ is about 8.9×10^{-5} for the high content of uranium, and about 3.6×10^{-4} for the low content.

According to the method mentioned above, two IAEA reference materials, IAEA-368 and SOIL-6 were analyzed. Purified Pu fractions were obtained by using UTEVA+TEVA extraction chromatography; $^{240}\text{Pu}/^{239}\text{Pu}$ were determined by MC-ICP-MS. The results were showed in table 1.

Ocean sediment IAEA-368 from Mururoa atoll[1], which was the former nuclear weapon test site of France, had a very small value of $^{240}\text{Pu}/^{239}\text{Pu}$, that is, the plutonium was weapon grade. SOIL-6 was sampled from Ebensee in Upper Austria, the $^{240}\text{Pu}/^{239}\text{Pu}$ was consistent with the global fallout value of 0.176 ± 0.014 .

Table 1. $^{240}\text{Pu}/^{239}\text{Pu}$ in two IAEA reference materials.

	$^{240}\text{Pu}/^{239}\text{Pu}$ This study	$^{240}\text{Pu}/^{239}\text{Pu}$ Reference[2]
IAEA-368	0.035 ± 0.015	0.043 ± 0.008
SOIL-6	0.186 ± 0.015	0.191 ± 0.005

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

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A multilingual text mining based content gathering system for open source intelligence

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Abstract. The number of documents available in electronic format has grown dramatically in the recent years, whilst the information that States provide to the IAEA is not always complete or clear. Independent information sources can balance the limited State-reported information, particularly if related to non-cooperative targets. The process of accessing all these raw data, heterogeneous both for source and language used, and transforming them into information, is therefore inextricably linked to the concepts of automatic textual analysis and synthesis, hinging greatly on the ability to master the problems of multilinguality. This paper describes a multilingual indexing, searching and clustering system, whose main goal is managing huge collections of data coming from different and geographically distributed information sources, providing language independent searches and dynamic classification facilities. The automatic linguistic analysis of documents is based on Morpho-Syntactic, Functional and Statistical criteria. This phase is intended to identify only the significant expressions from the whole raw text: the system analyzes each sentence, cycling through all the possible sentence constructions. Using a series of word relationship tests to establish context, the system tries to determine the meaning of the sentence. Once reduced to its *Part Of Speech and Functional* tagged base form, referred to its language independent entry inside a sectorial multilingual dictionary, each tagged lemma is used as descriptor and possible seed of clustering. The automatic classification of results is made by Unsupervised Classification schema. By Multilingual Text Mining, analysts can get an overview of great volumes of textual data having a highly readable grid, which helps them discover meaningful similarities among documents and find any nuclear proliferation and safeguard related information. Multilingual Text Mining permits to overcome linguistic barriers, allowing the automatic indexation and classification of documents, whatever it might be their language, letting international agents cut through the information labyrinth.

1. Introduction

The number of documents available in electronic format has grown dramatically in the recent years, whilst the information that States provide to the IAEA is not always complete or clear. Generally speaking, up to 80% of electronic data is textual and most valuable information is often hidden and encoded in pages which are neither structured, nor classified. The availability of huge amount of data available in the open sources leads to the well-identified nowadays paradox: an overload of information means no usable knowledge. Besides, open source texts are - and will be - written in various native languages, but these documents are relevant even to non-native IAEA speakers. Independent information sources can balance the limited State-reported information, particularly if related to non-cooperative targets. The process of accessing all these raw data, heterogeneous both for type (scientific article, patent, free textual document), source (Internet/Intranet, database, etc), protocol (HTTP/HTTPS, FTP, GOPHER, IRC, NNTP, etc) and language used, and transforming them into information, is therefore inextricably linked to the concepts of focused crawling, textual analysis and synthesis, hinging greatly on the ability to master the problems of multilinguality. This task can require undoubtedly remarkable efforts.

2. The infrastructure

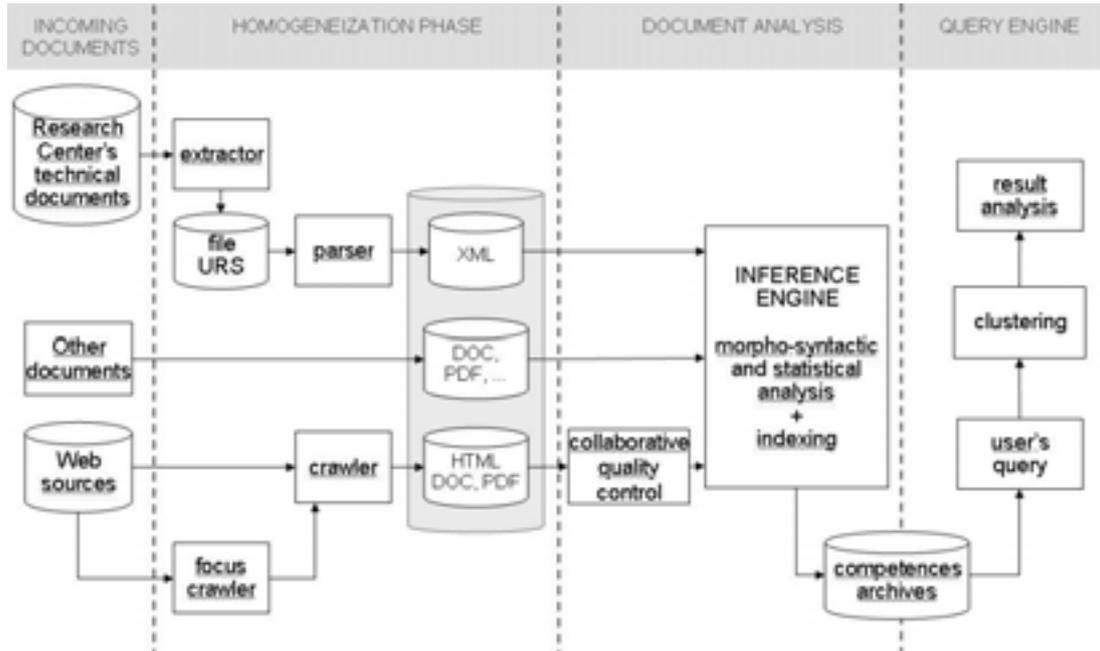


FIG. 1. Overall system infrastructure.

2.1. The gathering system: Searchbox

In any large company or public administration the goal of aggregating contents from different and heterogeneous sources (even if they are located and managed by the company itself) is really hard to be accomplished. Exporting data from an existing database means that all the people providing and using the content has to obtain the necessary authorizations, or some human resources have to be allocated in order to write the sw procedures needed to get the data. In this scenario, a crawling technology can enormously simplify the integration task, because the crawler acts exactly like any other authorized user whose accessing procedures are already defined and accepted by all departments inside the organization.

The crawler aggregates many different information sources and provides some standard application services to access them, as shown in Fig. 2.

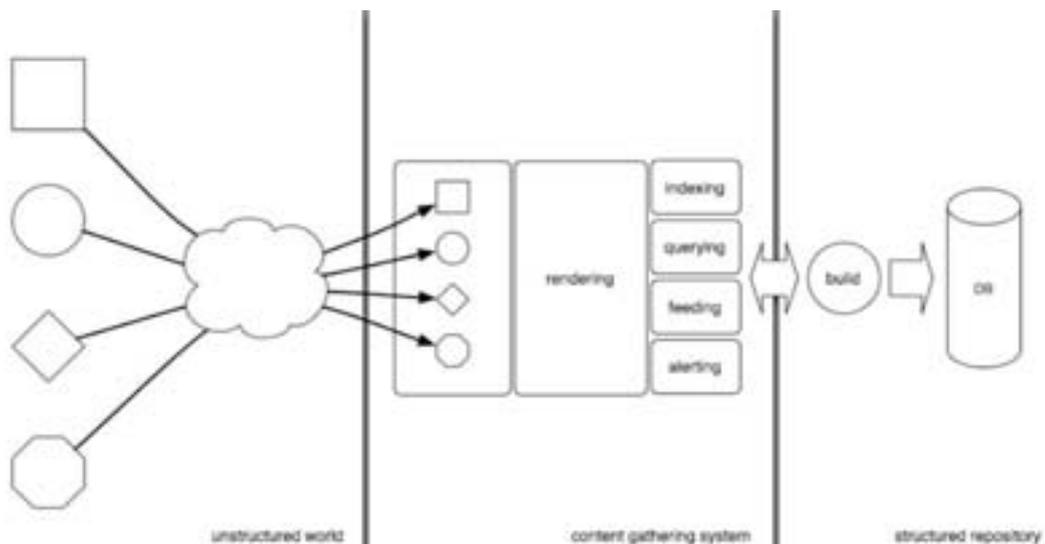


FIG. 2. The content gathering component.

In fact, on the left side of Fig. 2, the heterogeneous world of contents providers is sketched: the different shapes represent the different protocols and formats used to access documents. Then the content gathering module – in the middle - chooses the right adapter to gather information from any content provider, filling the structured repository on the right.

Searchbox is a multimedia content gathering and indexing system [3], whose main goal is managing huge collections of data coming from different and geographically distributed information sources. Searchbox, whose architecture has been conceived as a layer for information retrieval services in large enterprises, government institution, and Internet vertical portals, provides a very flexible and high performance dynamic indexing for content retrieval.

In Searchbox (see Fig. 2), the *gatherer* is the coordinator of a pool of agents whose task is to acquire new data from an information source, as soon as it is available. For instance, a noticeable example of a gathering agent is the focused Web crawler, which starts from a set of initial Web pages - the seeds - and performs intelligent navigation on the basis of appropriate classifiers. The gathering activities of the Searchbox, however, are not limited to the standard Web, but operate also with other sources like remote databases by ODBC, Web sources by FTP-Gopher, Usenet news by NNTP, WebDav and SMB shares, mailboxes by POP3-POP3/S-IMAP-IMAP/S, file systems and other proprietary sources [2].

The *renderer* is a central component in the Searchbox architecture. Searchbox indexing and retrieval system does not work on the original version of data, but on the "rendered version". Any piece of information (e.g. a document) is then processed to produce a set of features using appropriate algorithms. For instance, the features extracted from a portion of text might be a list of keywords/lemmas/concepts, while the extraction of features from a bitmap image might be extremely sophisticated. Even more complex sources, like video, might be suitably processed so as to extract a textual-based labeling, which can be based on both the recognition of speech and sounds. All extracted features are then compiled in an internal XML format and passed to the indexing module. The extraction process of the renderer component is done by a pipeline of plug-ins, which provide the compilation of the final XML representation.

The *indexer* creates the index of the collection of information gathered from multiple sources, while the querying module offers a complete query language for retrieving original contents, wading through millions of documents. The index is fully dynamic in the sense that any indexed content is almost-immediately available for queries. This is a crucial feature when the system is used on highly dynamic sources.

Searchbox indexer module can manage any feature that a specific renderer plug-in is able to extract from the original raw content. All of the extracted and indexed features can be combined in the query language which is available in the user interface. Searchbox provides default plug-ins to extract text from most common types of documents, like HTML, XML, TXT, PDF, PS and DOC. Other formats can be supported using specific plugins. Finally, a multilevel cache is available: the possibility to "historicize" different versions of the same document is a relevant practical feature, which turns out to be especially interesting for the implementation of the watch and alert concepts, when managing tons of documents.

2.1.1. Focused crawling

Focused crawling aims to crawl only the subset of the Web pages related to a specific category. The major problem in focused crawling is performing the appropriate credit assignment to different documents along a crawl path, such that short-term gains are not pursued at the expense of less-obvious crawl paths that ultimately yield larger sets of valuable pages. To address this problem the focused crawling algorithm builds a model for the context within which topically relevant pages occur on the Web. This algorithm shows significant performance improvements in crawling efficiency over standard focused crawling. In fact, the credit assignment can be significantly improved by equipping the crawler with the capability of modelling the context within which the topical materials is usually found on the Web [1]. Such a context model has to capture typical link hierarchies within which valuable pages occur, as well as describe off-topic content that co-occurs in documents that are frequently closely associated with relevant pages. The general framework and the specific implementation of such a context model are called Context Graph. It has a rapid and efficient initialization phase, being suitable for real-time services. The Context Focused Crawler (CFC) uses the

limited capability of search engines like AltaVista or Google to allow users to query for pages linking to a specified document. This data can be used to construct a representation of pages that occur within a certain link distance (defined as the minimum number of link traversals necessary to move from one page to another) of the target documents. This representation is used to train a set of classifiers, which are optimized to detect and assign documents to different categories based on the expected link distance from the document to the target document. During the crawling stage the classifiers are used to predict how many steps far from a target document the current retrieved document is likely to be. This information is then used to optimize the search. There are two distinct stages to using the algorithm when performing a focused crawl session:

- (1) An initialization phase when a set of context graphs and associated classifiers are constructed for each of the seed documents
- (2) A crawling phase that uses the classifiers to guide the search, and performs online updating of the context graphs.

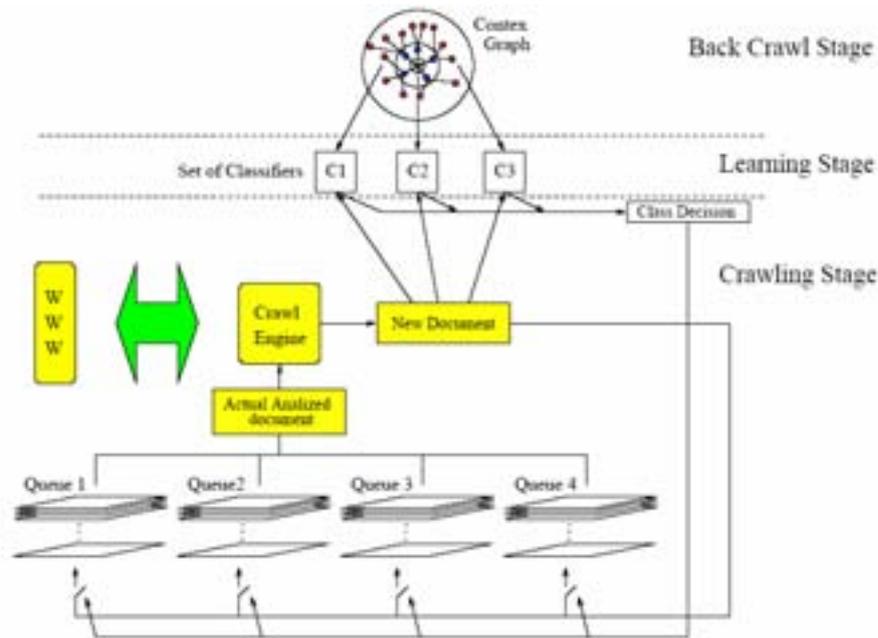


FIG. 3. Graphical representation of the Context Focused Crawler.

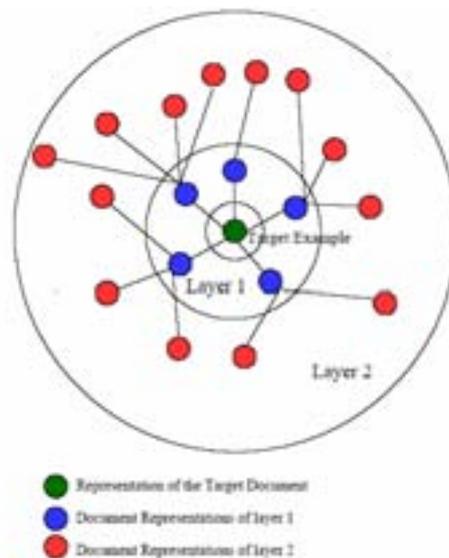


FIG. 4. A Context Graph represents how a target document can be accessed from the Web.

The detected terms and phrases are then extracted, reduced to their *Part Of Speech*¹ and *Functional*² tagged base form [5]. Once referred to their language independent entry inside the sectorial multilingual dictionary, they are used as descriptors for documents [7][8][9] and possible seeds of clustering. In multilingual dictionaries, each lemma is referenced to syntax or domain dependent translated terms, so that each entry can represent multiple senses. Besides, the multilingual dictionaries contain lemmas together with simple binary features, as well as sophisticated tree-to-tree translation models, which map - node by node - whole sub-trees [7].

2.3. The search and clustering system

Users can search document by keywords combined by boolean operators, or by typing³ their own query in Natural Language, expressed using normal conversational syntax. Traditional Boolean queries, while precise, require strict interpretation that can often exclude information that is relevant to user interests. The system analyzes the query, identifying the most relevant terms contained, their semantic and functional interpretation, expanding terms and concepts to all the languages supported by the system. The search engine returns as result all the documents which contain the query lemmas/concepts, having the same functional role [10] (See Fig. 6).

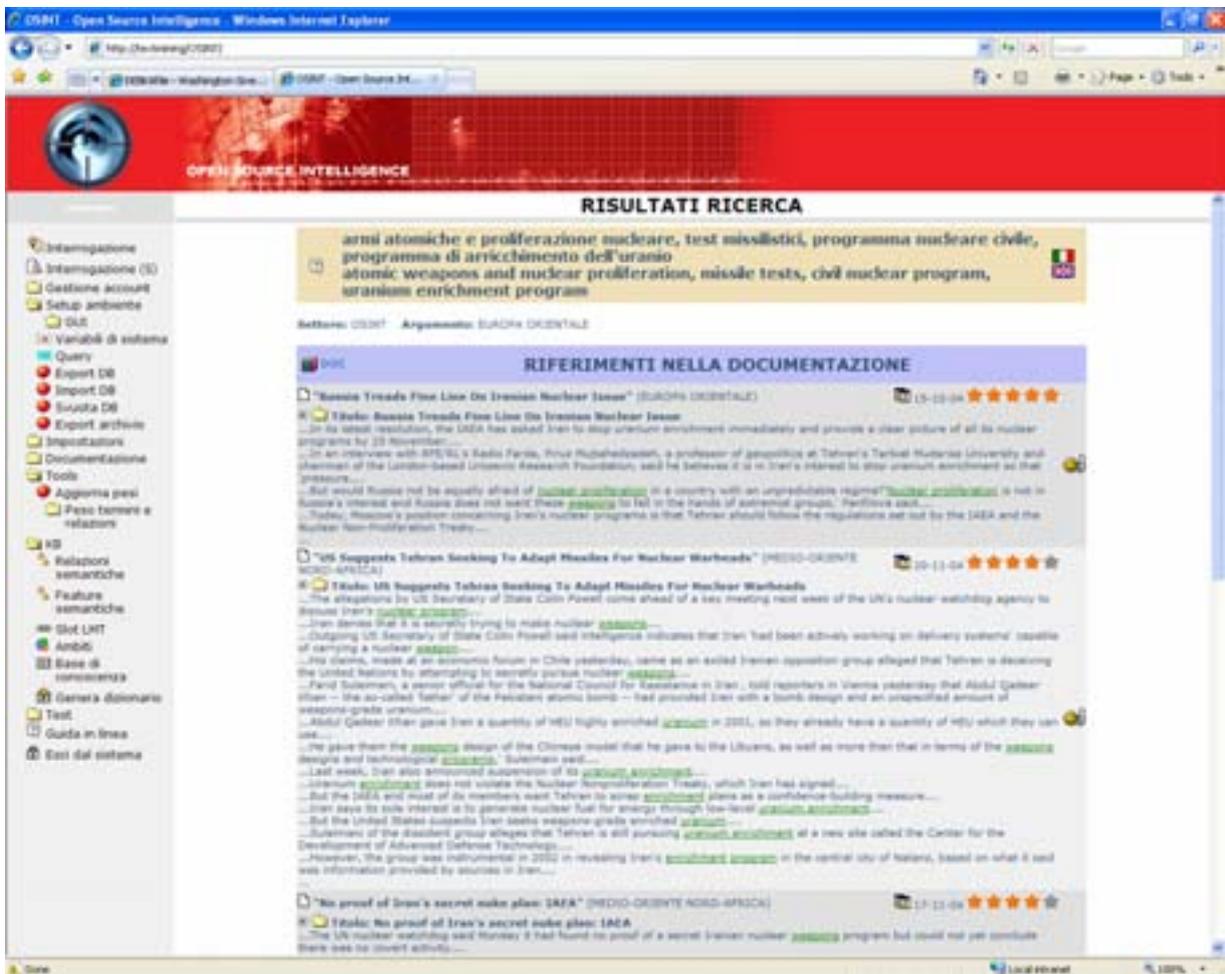


FIG. 6. Multilingual searching.

¹ NOUN, VERB, ADJECTIVE, ADVERB, etc.
² AGENT, OBJECT, WHERE, CAUSE, etc.
³ Speech recognition plugin is optionally available.

The automatic classification of results is then made by Online Miner Light, which is an application developed by TEMIS⁴ jointly with SYNTHEMA. It fulfils the Unsupervised Classification schema: the application dynamically discovers the thematic groups that best describe the detected documents, according to the K-Mean approach. This phase allows analysts to access documents by topics, not simply by keywords. In fact, the system provides a visual summary of the analysis (See Fig. 7), as well an ordered list of clusters. A map shows the different groups of documents as differently sized bubbles (the size depends on the number of documents the bubble contains) and the meaningful correlation among them as lines drawn with different thickness (that is level of correlation). Document analysts can search inside topics and have a look of the documents populating the clusters. The output results can be viewed by a simple Web browser, like Ms Internet Explorer.

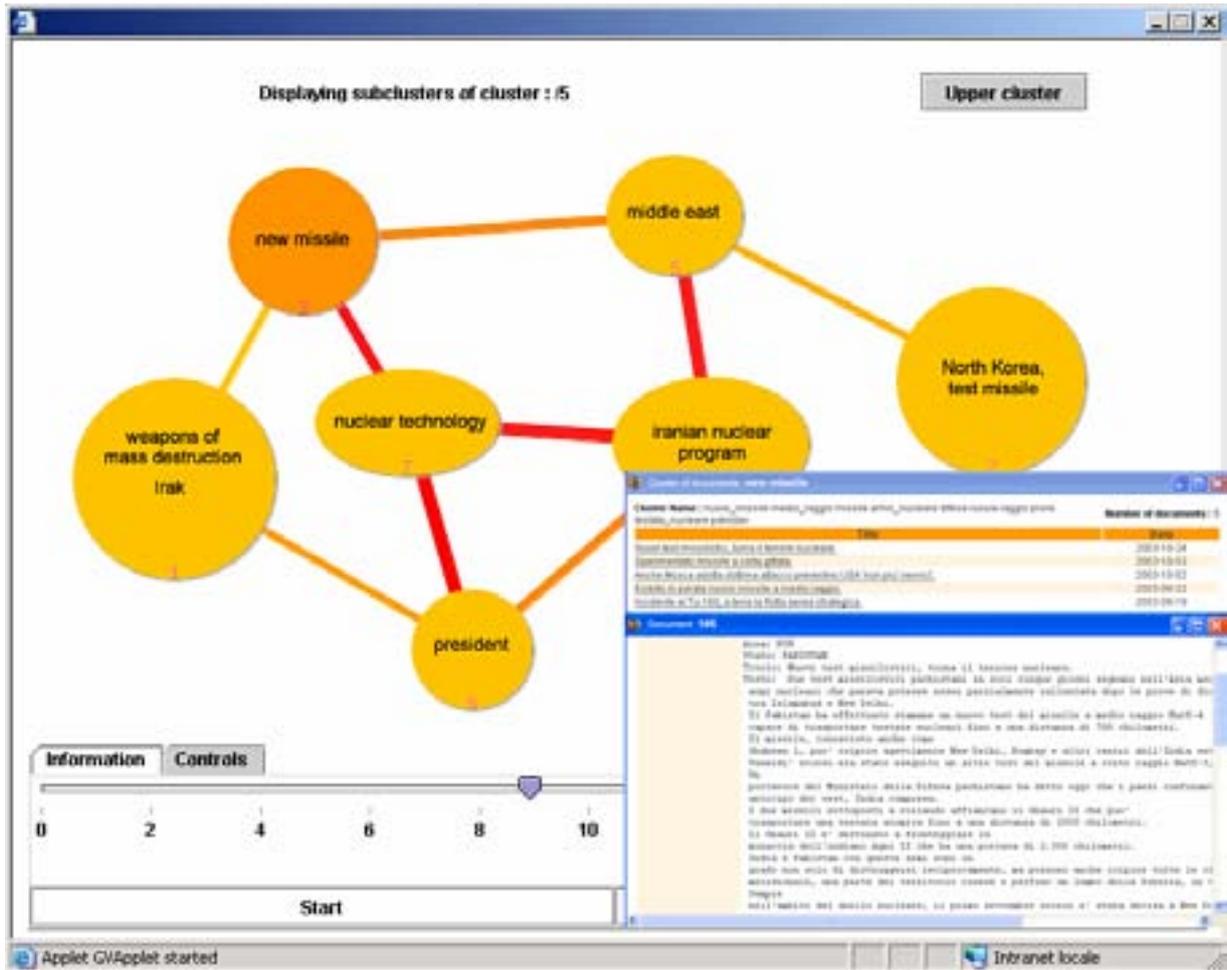


FIG. 7. Thematic map and documents visualization.

⁴ TEMIS was established in 2000 as a Technology & Consulting Company, specialized in Text Intelligence and Advanced Computational Linguistics to develop applications related to Competitive Intelligence, Customer Relationship Management and Knowledge Management.

3. Conclusions

This paper describes a Multilingual Text Mining application: Lexical analysis and Translation Memories permit to overcome linguistic barriers, allowing the automatic indexation and classification of documents, whatever it might be their language, or the source they are collected from. This new approach enables the research, the analysis, the classification of great volumes of heterogeneous documents, helping open sources analysts to cut through the information labyrinth.

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RADAR and CRISP - Standard Tools of the European Commission for remote and unattended data acquisition and analysis for nuclear safeguards

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Abstract Modern software packages for unattended, automatic measurement and analysis of safeguards relevant data are presented. RADAR (Remote Acquisition of Data and Review) is applicable to virtually all relevant installed sensors applied by nuclear inspectors in the field and CRISP (Central RADAR Inspection Support Package) analyses large amounts of data in automatic or semiautomatic ways. The remote transmission of data is inherently built into RADAR and used on sites. The transmission to headquarters has been successfully tested. The data acquisition systems are implemented in all large facilities within the European Union with a particular focus on facilities handling direct use material in bulk form. RADAR also helps to acquire data in a number of storage locations and contributes to save inspector resources in spent fuel loading campaigns. Several implementations are in shared use with the IAEA. Future aims are discussed.

1. Introduction

The Nuclear Safeguards Directorate of the European Commission has a very significant requirement for physical verification of nuclear materials. Within the EU, the Commission is inspecting more than 1000 nuclear installations in nuclear weapons states as well as in non nuclear weapons countries. Some of the largest installations within the civilian nuclear fuel cycle are under EURATOM safeguards, including large reprocessing plants and mixed oxide fuel fabrication facilities as well as large Plutonium stores. These installations, most of which are located in the United Kingdom and France, present a double challenge. On one hand they have large throughput of strategically important materials and on the other the advanced automation of the processes and security requirements lead to a very reduced accessibility of the material to be inspected. The physical verification requirements in many large plants can only be satisfied with automatic data acquisition systems integrated into the plants in a tailor made fashion. DG TREN has a vast experience in this field

The member states of the European Union and the European Commission attach a high significance to the nuclear material safeguards of direct use material. The related requirements are addressed with the utmost care.

In order to safeguard even the most advanced facilities in the nuclear fuel cycle, the European Commission started already in the 1980s to deploy unattended non destructive assay systems (further discussed in (1)). A variety of specialized systems was developed. They ran successfully but became more and more difficult and expensive to maintain and due to their complexity required inspectors dedicated to specific facilities. To improve the situation it was decided at the end of the 1990s to develop a standard system combining high flexibility, modularity and userfriendliness. Standardisation of user interfaces should, apart from economic and other advantages, allow inspectors to quickly adapt to the safeguards systems in different facilities. The software package RADAR (Remote Acquisition of Data and Review) was designed and built. Today RADAR combines a modular data acquisition platform for many sensors with a data analysis software (CRISP: Central RADAR inspection support package) which has user interfaces designed for automatic treatment of large amounts of data. Thus RADAR can be tailored to changing needs with relatively small effort .

Modern inspection regimes require the highest efficiency and effectiveness of inspections. One of the central elements to achieve this is to make the relevant data available to inspectors in their offices. RADAR is TCP/IP based and allows remote data transmission by design and can satisfy this need. Another requirement is to analyse large amounts of data in an efficient way to reduce the burden of mechanistic verification by inspectors. With automatic and semiautomatic evaluation routines CRISP contributes significantly to this aim.

2. RADAR – The data acquisition platform

Conceptually, RADAR is a fully modular software. It was developed for Windows NT and ported to Windows XP. RADAR was first mainly written in Visual Basic (as well as C++) and then ported to Microsoft .NET.

For each sensor type, RADAR has a specific Data Acquisition Module (DAM). For a given measurement task, the required DAMs are activated. DAMs deal with proprietary communication protocols and control the specific parameters of a sensor (e.g. power, high voltage, periodicity, thresholds, communication protocols, configuration, regions of interest etc.). DAMs are being controlled by a watchdog process. A service, ensuring reliable acquisition. DAMs create data files but also log and setting files with technical information as well as alarm files. Fig 1 gives an overview of the modular structure.

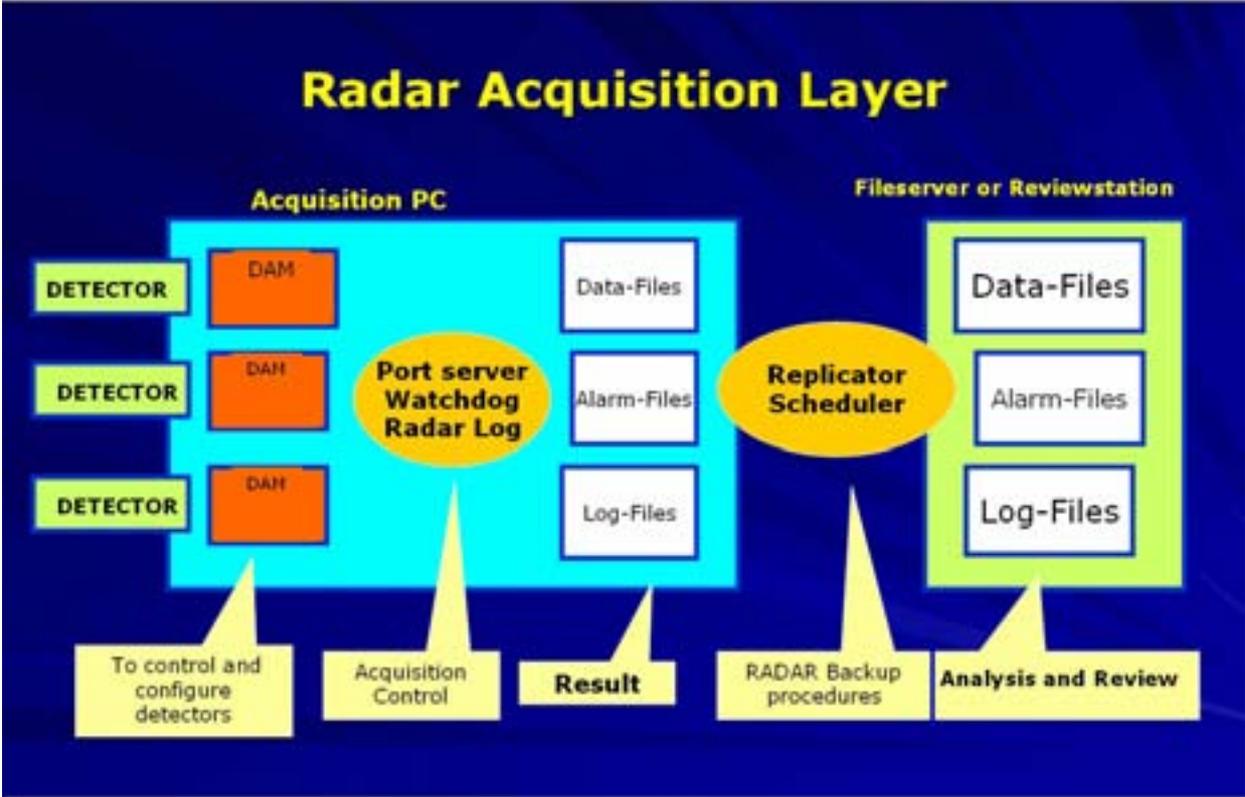


FIG 1 – the structure of the RADAR data acquisition layer

The complete data acquisition layer for a facility may consist of a small or large network of PCs, all services communicate via TCP/IP. Existing installations range from two PCs to more than 20 data acquisition computers running more than 50 DAMs in one facility. Specific processes, Scheduler and Replicator, can be used to support the administration of the whole data acquisition system. E.g. to schedule certain activities, copy files from a local data acquisition computer to a server or via a router to headquarters. Housekeeping functions (e.g. archiving, deletion of files) can be scheduled like this as well.

DAMs have been produced for many different instruments and are available for practically all important sensors, including neutron counting equipment (shift register JSR12), gamma spectroscopy (multi channel analyser MCA166), electronic seals (Vacoss, EOSS in preparation), 4-10mA signals (e.g. balances, tank height and density), for fork electronics (GRAND 3 and FST Mikromesskanal), for digital I/O cards, for specific ID readers, for the 3D Laser Range Finder for design verification, for acquisition via modem etc. For the DCM14 camera system, a DAM is in preparation.

Most important to note is that due to the modular design structure of the software, it is fairly simple to create a new DAM if a new sensor needs to be integrated into the data acquisition package.

3. CRISP – The data analysis

The data acquisition layer of RADAR is complemented by a data analysis package known under the acronym CRISP – for Central RADAR Inspection Package. CRISP is a database application which can be run on Microsoft Access for smaller and standard applications or, if the treatment of large amounts of data is required, on an Oracle database.

3.1 Structure of the data analysis and Algorithms

3.1.1 The evaluation layers

CRISP has several evaluation layers, shown in fig 2.

The analysis is done per Point of Interest (POI). A typical example of a Point of Interest would be a measurement station which combines e.g. a neutron coincidence counter, a gamma spectrometer and an identity reader. Fig 3 shows typical sample data for such a Point of Interest.

The first layer of CRISP takes care of raw data reduction. The raw data of each sensor are analysed and the relevant events are extracted.

In a second step, the events of all sensors belonging to a POI are evaluated. This means, as far as feasible the relevant analysis algorithms are applied. E.g., for a gamma spectrometer measuring Plutonium, the code MGA is called and the isotopic vector of the sample is determined. This is done for all sensors with the relevant algorithm. CRISP has a standardized interface which allows simple integration of external, third party algorithms for the analysis of any kind of sensor data.

In the third step, the events are correlated. This correlation is done through known information which is parametrized in the configuration of the analysis algorithm.

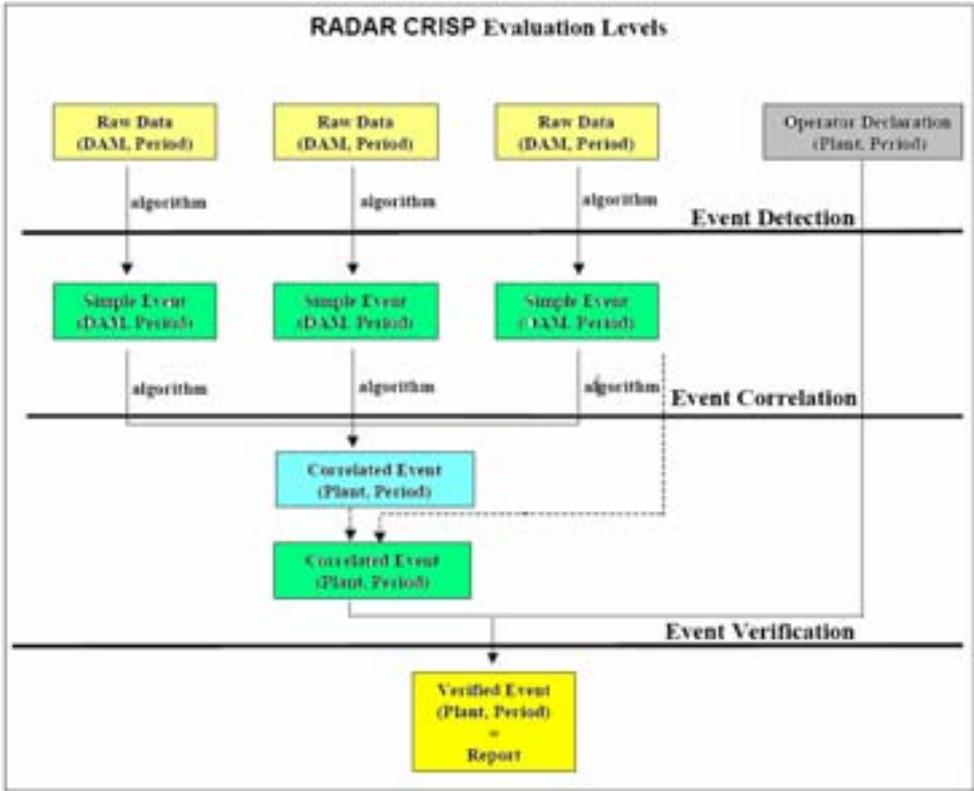


Fig 2 The evaluation layer structure of the data analysis package in CRISP

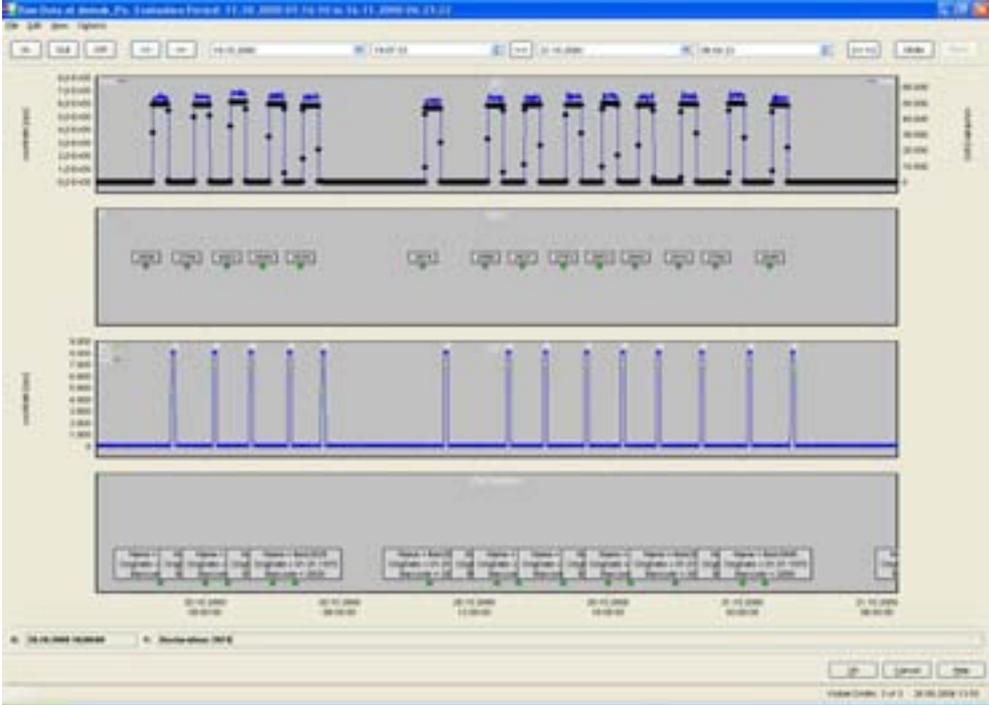


Fig 3 A typical example of raw data at a for the determination of the plutonium mass of an item. The traces show (from top to bottom) neutron data, identity measurements, gamma data and operator declarations

Time differences between measurements are the most common correlation parameters. In automatic production cycles, items are moved along transport systems with standard timing and on pre-defined paths. This behaviour allows for relatively simple configuration of the correlation algorithm. The analysis can then typically be fully automatic. In this step, further analysis routines are run as well. For example, the calculation of the plutonium mass of a sample requires as input the isotopic vector which has been determined one step before.

The last step of the analysis is the comparison with operator declared data. This step is then followed by the creation of a report, which is made available to the user.

3.1.1.1 Flow verification

The example above described the analysis flow for a qualitative analysis. However, CRISP can also be used for containment and surveillance purposes or for the more complex application to verify the flow of an item in a facility equipped with numerous sensors.

A typical complex nonlinear topology to be considered can be described as follows. Items may come from a production facility and move to one of several storage areas or to an export location. They may also move from a storage area to another storage area or from a storage area to the export location. A variety of sensors may exist – for example neutron monitors as part of the C/S system, NDA equipment to determine content and identity of the items. The time sequence of the sensors allows to determine the path which the item has taken and allows to make an assessment whether this is an allowed path or not and whether it is in line with the declarations.

We have developed a specific algorithm for CRISP which can be used at the stage of the event correlation to run through the necessary analysis and to prepare a full report for the inspector. The largest system to which this algorithm so far has been applied provides for four possible input/output points and nine sensors. To reduce inspection workload, the data are treated up to a level where a consolidated report is provided to the inspector giving only the most condensed ‘allowed/not allowed move’ information. A detailed description will be given in a forthcoming paper.

3.1.2 The algorithms

The modular structure of the data acquisition part of RADAR finds its counterpart in the modular structure of the analysis package. In principle, for each sensor type acquiring data a specific algorithm is required to analyse the data. Whereas some algorithms are trivial – e.g. the extraction of numbers from an ID reader – others can be very sophisticated, as for example MGA for the deconvolution of gamma spectra to extract the isotopic vector of Pu samples.

It has been recognized early on in the development of CRISP that it would be advantageous to have at disposal a standardized interface allowing the integration of external algorithms and thus make optimal use of the know-how available in the safeguards community. This interface has been used e.g. for the integration of MGA (3), to link a program to calculate the mass of Plutonium containing items (4), to apply the heavy metal correction (5), to calculate item identities from binary codes (6) etc. The interface lends itself to the integration of more programs in the future, e.g. to make use of spent fuel burn up calculations for the analysis of fork measurement data or integrate algorithms to evaluate neutron multiplicity counting data. CRISP is open to include in a relatively simple manner algorithms provided by others.

3.2 The configuration of CRISP by an expert user

Each facility has specific Points of Interest. Typically they are at boundaries of relevance. These can be e.g. the limits of the material balance area, or the input and output routes of storage areas or specific work areas etc. For each Point of Interest, CRISP needs to be configured once in order to know which sensors exist, which files need to be analysed and which algorithms to use. Also parameters of the algorithms need to be tuned once, e.g. threshold levels for event extraction or background parameters. This work needs normally to be done only once and requires the work of an expert user. He has at his disposal the CRISP Administration Tool CAT for configuration and the Report Administration Tool REAT to create reports adapted to the individual needs at a specific facility.

3.3 Interfaces of CRISP – for the standard user, for declarations, for reporting

3.3.1 The user interface

The standard user works with prepared configurations and algorithms. The user interface **SEAT** (Smart Evaluation and Analysis Tool) is a wizard like application which requires only a minimum of user input. The user selects the facility, the MBA and the Point of Interest he is interested in and chooses for which time period he would like the analysis to run.

In situations where the practically fully automatic SEAT cannot be applied, the interface **X-SEAT** is used. X-SEAT allows for evaluations in more complex situations where manual interference is occasionally or frequently required. This can be the case if the automatic production mode of the facility is disturbed, or if the background situation in a given POI is very dynamic, or if the variation between items is too large to be covered with a standard set of analysis parameters. X-SEAT also needs to be applied if a sensor fails or other anomalies need to be resolved. The user can interfere with the analysis routines and he is guided through the evaluation process with screens which give him a multitude of graphical representations and interaction options. In order to understand the need for X-SEAT we would like to mention that the design basis for RADAR and CRISP was a facility where at one single Point of Interest several hundred moves per day were expected.

3.3.2 Operator declarations input

As a rule, CRISP works with XML interfaces for data input and output.

Declarations are entered through **IOD**, the module for Input of Operator Declarations. As operator declarations per Point of Interest do not come in a standard way, converters have been prepared which can treat a variety of formats – files with comma separated variables, Excel spreadsheets, database excerpts, text files etc. The converter filters out the relevant data and prepares an XML output file. If this procedure is not applicable, the user can also input the declarations manually. IOD even allows the user to view existing measurement data and create manual timestamps to declarations should the exact time information not be available from the operator.

3.3.3 Report output

Reports are created in XML format files and are presented in a graphical way adapted to the needs at a Point of Interest. Depending on the needs, individual and summary reports can be created and printed. The adaptation of the reports to specific or changing needs is supported by REAT, the Report Administration Tool.

4. Technical Review, ARNI, auxiliary tools

With complex and automatic data acquisition systems as RADAR it is important to be informed of the status of the system. RADAR services like DAMs, scheduler, watchdog etc produce log and alarm files. They as well as the data files themselves contain the status information of the system. Already in the early days of the development a program called 'Global Surveyor' was developed to analyse the data files for completeness and condense the information contained in the log and alarm files for easy technical review. For the analysis of status of health files for the Gemini surveillance system this program has been successfully in use for several years. An improved version is now in the planning phase and is expected to be able to handle all technical status files arriving at headquarters.

Another tool which is of importance for use in large installations is ARNI, the advanced RADAR network investigator. ARNI is a technical surveillance tool allowing to easily control the status of the processes and machines in the RADAR data acquisition network.

5. Status of Remote data transmission

RADAR is fully remotely operable. The transmission of data from plants to on site offices has been utilized for many years. However, the transmission to headquarters needs to take into account the specific security requirements of the operators and the Member States as well as those of the Commission. Tests where technical data have been transmitted were run successfully for several years from two installations in two EU countries. The full transmission of data from a UK facility is at the time of writing almost fully prepared. This scheme was described earlier (7).

The full utilization has a tremendous savings potential for the IAEA as well as for DG TREN. The details of cooperation between the two organisations are under discussion and will take into account the specificities of the regional character of the Euratom Comprehensive Verification Agreement INFCIRC 193.

6. Training

A suite of training courses exists and is further developed in order to have well trained users make full use of the capabilities of the system RADAR and CRISP. The full suite of courses consists of three levels.

Level one presents RADAR and CRISP concepts and overview. The user is familiarized with the acquisition interfaces as well as the standard analysis interface and is trained on the network investigation tool ARNI. Target audience are beginners and infrequent users of simpler (few sensor) systems. Level one courses have been held in Luxembourg and at the IAEA HQ in Vienna.

Level 2 courses are held in cooperation with the PERLA laboratory of the European Commission's Joint Research Centre (IPSC) at Ispra. There an in-depth understanding of the internal machineries of CRISP is being developed. Training measurement systems with real sensors (neutron, gamma, ID and others) are used to demonstrate how to configure the evaluation routines, reporting etc and SEAT and X-SEAT are made familiar in practical exercises. This level addresses inspectors of large installations.

Level 3 courses are under development for expert users who frequently need to configure systems or adapt existing configurations.

7. Conclusions and outlook

RADAR and CRISP are modern, fully modular automatic data acquisition and analysis systems designed and built to allow inspectors to work with higher efficiency and effectiveness. The packages are available to date. They are both continuously upgraded and modernized. Concerning RADAR, the integration of DCM14 surveillance sensors is in preparation as well as the inclusion of the new standard electronic seal EOSS. Concerning CRISP, the data analysis has a huge potential of further enhancement, which also leads to reduction of inspector analysis workload and adds to the quality of information derived from inspections. One focus is in the analysis of spent fuel measurements, another large area of potential development is in the more systematic statistical analysis of the data acquired and extracted from measurements.

Acknowledgements

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Significant contributions have been made by many colleagues, including DG TREN safeguards inspectors as users, who helped to make the software a useful tool, but also by staff of the JRC who contributed to the development of the training courses and are also currently involved in the definition of new data acquisition modules. For particularly significant contributions we thank Martyn Swinhoe, Alain Dutrannois, Ludwig Holzleitner, Jan Loeschner and Willi Koehne.

A cooperation with the US Department of Energy resulted in the DLL of MGA which is integrated into CRISP.

In Memoriam

We wish to dedicate this paper to Wayne Ruhter, whose early loss we regret. Some of his invaluable contributions to safeguards techniques are part of this work.

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Tunable diode laser spectroscopy as UF₆ monitoring technique

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Abstract. Tunable Diode Laser Spectroscopy (TDLS) uses diode lasers tuned to access specific regions of the mid-infrared of the electromagnetic spectrum; spectral regions where gases of interest for detection such as Uranium Hexafluoride (UF₆) have strong spectral absorption. The Natural Sciences Centre of the General Physics Institute in Moscow, along with its joint venture partner Canberra Albuquerque, Inc., has developed a portable system that uses TDLS to detect the presence of UF₆ isotopes in-situ as well as, more importantly, the percentage of enrichment of the involved process.

This technology is appealing for application towards safeguards for two reasons. It provides a non-invasive method of isotopic measurement, and can be performed in-situ with real-time data processing. The in-situ use of TDLS is of interest to traditional safeguards (e.g., UF₆ samples are sent to International Atomic Energy Agency (IAEA) laboratory facilities for destructive analysis) as well as under the IAEA's expanded responsibilities under the Additional Protocol.

Optical monitoring methods such as diode laser spectroscopy allow non-contact isotopic measurement to be combined with real-time data processing. This potential means that new applications for safeguards use, such as a TDLS-based enrichment monitoring system, are possible. Although such development efforts are technically feasible in principle and are encouraged by the IAEA, their political impact on safeguards implementation and their economic feasibility compared to the current establishment monitoring system must first be addressed. If such concerns are properly managed, TDLS can become the future enrichment safeguards system for enrichment facilities.

Introduction

To fulfill its mission of verifying that international safeguards agreements signatory parties are in compliance with their non-proliferation commitments, the International Atomic Energy Agency (IAEA) applies a strict inspection regime. In addition to IAEA inspectors visiting declared nuclear facilities with a 'minimum time needed to divert' (timeliness) approach, safeguards instrumentation is installed to gather information between inspection visits, establishing Continuity of Knowledge about the compliance of Non-proliferation Treaty (NPT) Member States.

To make best use of emerging technologies and to keep abreast with potential adversaries using state-of-the-art instrumentation and approaches to defeat the inspection system, the

IAEA constantly monitors and analyses technology innovations for possible application within its inspection regime, starting at the early research and development level. One such research area that has seen significant development in the last few years is Tunable Diode Laser Spectroscopy (TDLS). TDLS-based detection systems apply diode lasers that are tuned to investigate specific regions of the mid-infrared of the electromagnetic spectrum where gases of interest (such as Hydrogen Fluoride (HF) and Uranium Hexafluoride (UF₆)) have strong spectral absorption. Thus, TDLS can be applied to detect the presence of such molecules in an air sample, as well as, in direct connection with UF₆, to identify the percentage of enrichment of the involved process.

This technology is appealing for application towards safeguards for two reasons. First, it provides a non-invasive method of isotopic measurement. Second, it can be performed in-situ with real-time data processing. This second advantage is not only of interest to traditional safeguards where, for example, UF₆ samples are sent to IAEA laboratory facilities for destructive analysis, but also in the framework of the expanding responsibilities of the Agency under the Additional Protocol. Short- and no-notice inspections have a need to be supported by rapid analysis instrumentation to give inspectors the tools needed to quickly conclude on the status of the inspected site for further action. Also, TDLS could potentially serve as an enrichment monitoring tool for safeguards conclusions if political implications and the complex technical environment of enrichment facilities can be properly addressed.

The present paper will describe the status of the technology today, the basics of TDLS, and the efforts undertaken by the IAEA to make use of this technology to date. Next, it will outline how the current technology can be extrapolated into a permanent safeguards installation at centrifugation enrichment facilities. The impact of such an application on both the operator and safeguards authorities, not only with regards to the Hexapartite agreement but also with regards to the economic feasibility when compared to the existing safeguards approach will then be highlighted. Recommendations on how proposed technologies and solutions can be implemented will conclude the paper.

TDLS Technique and UF₆ in the nuclear industry

Today, TDLS has become one of the most intensively developing areas of analytic applications of lasers, especially in spectroscopic detection of molecular species in the atmosphere. Systems based on tunable diode lasers are known to be among the most sensitive apparatus for detecting molecular species. These special lasers have been developed in the USA, Germany, France, Italy, UK, Japan, and Russia with very good results. Tunable diode laser based systems are especially effective due to their unique parameters, such as high sensitivity, discrete selectivity, and extremely short response time. Thus, they show great promise in applications such as measuring trace concentrations for environmental protection, medicine, explosive detection, drug detection, enrichment measurements, among others. More than 300 different molecules, radicals, and ions have been detected using TDLS, essentially by simply changing the frequency spectrum of the laser in order to analyze the specific molecular species.

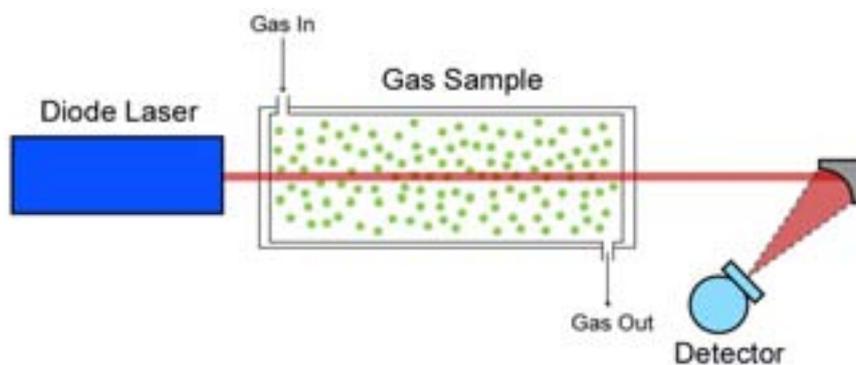


FIG. 1. Simplified TDLS Detection Process.

TDLS is related to a method of measuring the isotopic ratio in a gas based on precise excitation of isotope species, and more particularly to a method wherein the selective excitation is initiated by laser means.

The TDLS system (as shown in figure 1) draws the sample gas into a container. The tunable diode laser shines through the sample gas, hitting a detector on the other side. The detector then analyzes the absorption lines caused by the gas in the cell to identify the composition of the sample. For the safeguards inspection regime, this process is particularly applicable towards measuring the isotopic enrichment of uranium hexafluoride.

The UF_6 molecule, the single known gaseous uranium component, is used in the nuclear industry for uranium isotope separation purposes (separation of ^{235}U from the ^{238}U). The ^{235}U isotope is enhanced using various enrichment technologies. The monitoring of the ^{235}U isotope concentration is very important during the enrichment process, as well as during periodic verification on site of stored UF_6 in containers, and the IAEA measures the enrichment of UF_6 stocks as a potential indicator of weapons production.

Uranium is found naturally as two isotopes: ^{235}U and ^{238}U . Naturally occurring uranium has a composition of 0.7% ^{235}U and 99.3% ^{238}U . When processed for use as nuclear fuel, the uranium is “enriched” by various means to a composition of 5% ^{235}U ; and when processed for use in nuclear weapons it is enriched to more than 20% ^{235}U . Both enrichment processes also produce “depleted” uranium as a by-product, having less than 0.2% ^{235}U . Measuring the enrichment level with precision is a key element for the IAEA.

There are two methods available to non-destructively measure UF_6 on-line. One method uses ^{241}Am as a source of 60 keV gamma radiation and a low resolution NaI (TI) detector for the measurement of ^{235}U and ^{238}U by detecting the 60 keV and 186 keV gamma radiation, the last is specific only for ^{235}U . The content of ^{235}U is calculated from the ratio of the two measurements. The other method for the measurement of ^{235}U at low pressure uses ^{57}Co as a source of 122 keV gamma radiation. It should be noted that at low pressure of UF_6 both methods have poor precision (about 20%) and can only be used to determine whether the enrichment of ^{235}U exceeds or is less than 20%. Furthermore, special measures need be taken to distinguish between the uranium in gaseous phase and the uranium deposited on the container walls.

The standard destructive analysis method applied by the IAEA is to measure individual gas samples of UF_6 with a mass spectrometer. Samples need to be collected in special gas bottles and transported to a suitable mass spectrometer. The precision of such enrichment measurement is on the order of 1%. The main advantage of optical methods, and in particular

TDLS, is the possibility of non-contact isotopic measurement combined with real data processing. Another feature is high selectivity, as different molecules and isotopes have different infrared spectra.

While mass spectroscopy can achieve the required measurement uncertainty of about 1%, the non-destructive methods have measurement uncertainties on the order of 20%, for example, $0.7\% \pm 20\%$. Consequently, these methods cannot distinguish between natural and depleted uranium, and are not reliable enough to identify the presence of weapons-grade uranium. The TDLS method has high potential to significantly increase this precision as results presented in this paper will indicate. In addition to the advantages as an optical method, this precision outlines the potential interest of the IAEA of using this safeguard method.

As a result, two potential developments can be investigated, one of which, the portable application, has already been pursued by the IAEA. A permanent installation, for example at the end of an enrichment cascade is another application of great potential from a technical point of view, but also one that has to be carefully investigated as it is heavily influenced by operator concerns, safeguards application policies, and other factors.

IAEA TDLS UF₆ detector

Due to the fact that UF₆ has many absorption bands located in the mid- and far-infrared and that mid-infrared diode laser spectroscopy can be applied to the measurement of uranium isotope concentrations in gaseous UF₆, the IAEA has conducted a joint research and development project for a TDLS-based UF₆ detector. A prototype detector was developed and successfully demonstrated. Figure 2 represents a schematic overview of the system.

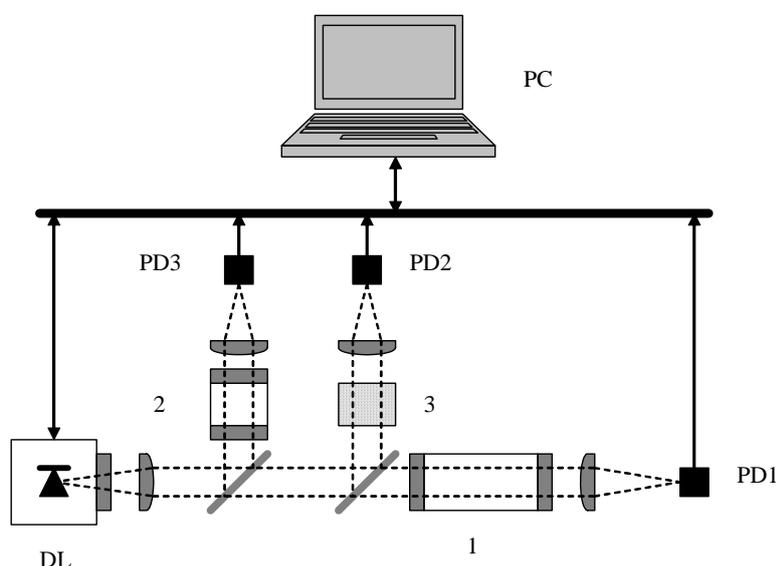


FIG. 2. IAEA UF₆ Detector Schematics.

The Diode Laser (DL) which is tuned to the range of the spectrum where UF₆ has high absorption is installed in a cryostat that provides cooling. The laser radiation from the DL is collimated into a parallel beam by the lens that is installed in front of cryostat. Light is guided into three instruments channels via beam splitters. All windows, splitters and lenses are made of Barium Fluoride (BaF₂) that is transparent in the mid-infrared spectral range. In each channel, the laser radiation is focused by lenses on the sensitive area of Mercury Cadmium

Telluride (HgCdTe) photodiodes that are installed in additional cryostats. The amount of Cadmium in the HgCdTe alloy can be chosen so as to tune the optical absorption of the material to the desired infrared wavelength.

Reference channels (2) and (3) are used for the diode laser frequency tuning calibration. In channel (2), a cell with reference gases (CH_4 , C_2H_2) is installed to determine the absolute calibration of DL frequency tuning as well as to provide stabilization. The reference channel (3) is designated to determine the frequency tuning curve of the DL. It consists of a Fabry-Perot interferometer with the free dispersion region of 0.095 cm^{-1} .

In the analytical channel (1), a sample optical cell with the UF_6 gas mixture under investigation is installed. Channel (1) is designated to carry out routine measurements of the UF_6 concentration and the value of the $^{238}\text{UF}_6/^{235}\text{UF}_6$ isotopic ratio in an unknown gas sample of uranium hexafluoride. It contains a stainless steel sample cell of a length of ten centimetres that is displayed in Figure 3. The laser radiation from the DL passes through the gas sample using a window made of BaF_2 on each end of the sample cell. UF_6 gas sample under investigation is introduced into the sample cell through a valve which is controlled by a valve handle. The sample cell is attached to a stabilizing base by two u-shaped brackets.

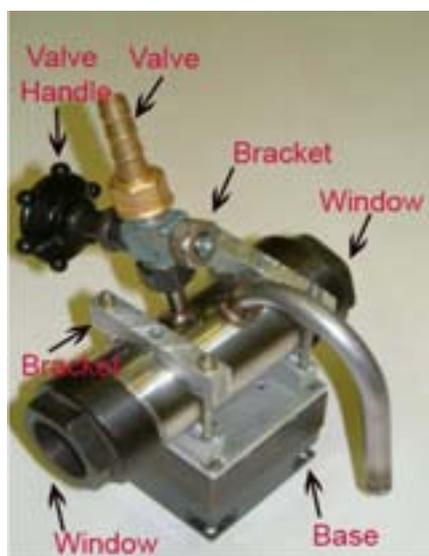


FIG. 3. TDLS Detector Sample Cell.

Signals received from photodiodes were amplified by specially developed preamplifiers and recorded by a multifunctional National Instruments DAC/ADC board (PCI-MIO-16E-1) installed in a Personal Computer which controlled all system operations. It produced the laser excitation current, stabilized the DL temperature, and recorded signals from the three photodiodes.

The UF_6 gas is irradiated with laser radiation that is tuned across all frequencies in the region of the absorption band that exhibits shift for the isotopes of interest. All three channels operate at a frequency of $\nu=1,290 \text{ cm}^{-1}$ which is most suitable for the detection of gaseous UF_6 . As the frequency of the laser light is varied, the irradiation excites detectable vibrations in the gas molecules of each isotope at the characteristic frequency for that isotope. Precise measurement of the isotopically-broadened absorption peak provides a measure of the relative contribution of each isotope to the broadened peak, and therefore the relative concentration of each isotope in the gas.

The achieved measurement precision was analyzed and error sources were identified. The relative precision of the content of isotope ^{235}U is characterized by a root-mean-square spread of about 30%. Straightforward engineering refinements hold the potential to improve this precision to the required 1%.

Application of TDLS for UF_6 on-site monitoring

Applications for such a detector exist in portable devices for the validation of UF_6 storage container declarations: measure of enrichment level in UF_6 cylinder gas samples. This is currently performed via time-consuming destructive analysis of the samples. The on-site laser technique would allow the IAEA to considerably increase the number of bias detect tests on containers with instant data analysis.

In addition to a portable TDLS detector that could be carried by IAEA inspectors into the field for measurements, its application of a similar instrument is envisioned: a continuous enrichment meter permanently installed at gas centrifuge enrichment facilities or more specifically outside of the enrichment room to control the input and the outputs of UF_6 . With TDLS technology advancements to this date, such an instrument could apply a similar tunable diode lasers technology as it was proven to the IAEA to measure the enrichment level of the gas.

Similar to gaseous diffusion, the centrifuge process makes use of the differing masses of ^{235}U and ^{238}U . The UF_6 feed is channeled into a series of vacuum centrifuges that spin at between 50,000 and 70,000 rounds per minute. The heavier molecules (^{238}U) concentrate towards the outside of the spinning centrifuge whereas the lighter molecules (^{235}U) concentrate towards the center. The enriched gas serves as feed for the next stages in the centrifuge cascade while the depleted UF_6 gas is channeled back to the previous stages. At the end of the cascade, uranium of the desired enrichment level can be drawn from the process.

Enrichment using gas centrifuges was invented in the 1940s but initially abandoned in favor of the simpler diffusion process. As a second generation enrichment technology, centrifuge enrichment was further developed and fielded in the 1960s. It has been commercially deployed by the industrial group of German, Dutch, and British companies that formed URENCO. Russia also operates four centrifuge enrichment facilities that account for about 40% of the world's capacities.

Because enrichment technologies are capable of producing uranium suitable for both the civil nuclear fuel cycle and for nuclear weapons programs, the control of enrichment technologies and the safeguarding of such facilities are of great importance for the non-proliferation community. In 1983, the Hexapartite Safeguards Project was signed by the technology holding countries Australia, Germany, Japan, the Netherlands, the United Kingdom and the United States as well as the regulatory authorities IAEA and EURATOM. The Hexapartite agreement regulates the inspection of enrichment facilities even in Nuclear Weapons States on a Limited Frequency – Unannounced Access (LFUA) basis.

To support safeguards at declared facilities, a TDLS enrichment meter could be installed at in the product line to perform real-time, online measurements. Gathered data can be stored securely until retrieved by an inspector during regular inspection visits or, where agreements permit, transmitted securely to the IAEA or other safeguards authorities. In addition to the measurement of the enrichment level and the UF_6 total pressure which would follow the same

principle as for the portable application outlined above, a permanent TDLS enrichment meter could measure one more parameter: the flow rate.

Even though the measurement of all three parameters – enrichment level, pressure, and flow rate – is technically feasible, its actual application at enrichment facilities is impacted by policy decisions and confidentiality concerns of both operators and safeguards authorities. It has to be understood that the current safeguards regime at enrichment facilities is internationally accepted and has proven itself, and the impact of a new technology on this established system has to be closely investigated.

Impact on safeguards authorities and operators

From a technical point of view, TDLS-based permanent enrichment meters could be installed at any appropriate point in the enrichment process and provide real-time information on the three parameters enrichment level, flow rate, and pressure. Considering the operation of enrichment cascades, applying TDLS meters in the cascade itself (in-between centrifuges) might lead to diluted readings due to deposits in the pipes or due to artifact spikes when the cascade is brought on-line. Installing the system at the product line outside of the enrichment room might thus be the most preferable solution, even though it might not be capable of detecting UF₆ that is diverted somewhere in the process prior to the measurement. Also, the application of this technique would work best when complemented by other safeguards measurements.

There are, however, non-technical considerations that impact the actual use of the TDLS capabilities. While all three parameters might be used by safeguards authorities to obtain information about the declaration compliance of the facility, especially the pressure parameter is sensitive information and not subject to disclosure from an operator's point of view. Also, the application of the technology is not easily applied to existing facilities as the diode laser needs to be capable of 'looking into' the piping it is monitoring. Enrichment facilities are highly complicated and sensitive environments, and the design changes that would be necessary to implement the system would have serious impact on operational procedures as well as established safeguards inspection routine.

The economic value of a regime change is another factor that would impact the selection of TDLS as an enrichment monitoring tool. The design changes to any given facility, the development and fielding costs, as well as operational costs of a TDLS monitor might outweigh the advantages gained by reducing the destructive analysis costs and gaining the benefit of faster data availability. It might turn out that TDLS-based units only make sense when they can be designed into the facility plans of new facilities when there are no operational concerns of impacting production. Also, TDLS-based units might prove to be of greater value to operators as alternative to their mass spectrometry units rather than for safeguards authorities.

If the regulatory boundaries are identified and properly addressed, TDLS can serve as a continuous enrichment meter that will function reliably at low pressure rates. Even if only the enrichment parameter is measured, the gathered information could significantly increase the ability of a safeguards authority to draw safeguards conclusions beyond the measurement methods that are currently applied.

Also, the information gathered could potentially be used as a transparency building measure if applied to the concept of multi- or international fuel cycle models. Under the assumption that countries that provide fuel enrichment services for other countries in exchange for these countries foregoing the development of own enrichment capabilities, the issue of assurance of supply becomes of utmost importance for such models to become feasible. The operation of enrichment facilities will be under close scrutiny of the fuel recipients in such models and sensitivity cleared data from the TDLS system could provide transparency that the facility is operating as declared and not misused for covert activities.

Conclusion

The use of the Tunable Diode Laser Spectroscopy as UF₆ Monitoring Technique has shown promising results. The method has been successfully demonstrated for identifying and calculated the level of enrichment in a given gas sample. With the use of new lasers with better tuning characteristic, the precision of the measurements could be further increased. Also, the application of lasers such as Quantum Cascade Lasers would remove the requirement to cool the laser and make the technique much easier to field as a portable instrument.

To bring TDLS closer to application within the safeguards community, it would be recommended that the IAEA further pursues the development of fieldable units of the portable TDLS UF₆ detector for actual use by IAEA inspectors. This could be driven by both applications under traditional safeguards to make use of the real-time measurement of UF₆ containers content as well as Additional Protocol measures for the detection of undeclared activities as undeclared sites. Funding for the development activities could be requested from Member States Support Programs.