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Return of Research Reactor Spent Fuel to the Country of Origin: Requirements for Technical and Administrative Preparations and National Experiences

Proceedings of a technical meeting held in Vienna, August 28–31, 2006

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FOREWORD

The back-end of the Research Reactor (RR) nuclear fuel cycle is not solely a technical issue. Non-proliferation, physical security and environmental concerns are equally as important, if not more so, as technical concerns.

At present, all aspects of the back-end of the RR nuclear fuel cycle are capturing increasing interest in Member States that operate RRs and are being extensively discussed in the major meetings of the international RR community. International activities in the back-end of the RR nuclear fuel cycle are at present dominated by the RR spent fuel take back programmes. The major goal of the separate take-back programmes for the United States of America and the Russian Federation origin fuels is to eliminate inventories of HEU by returning RR spent nuclear fuel to the country where the fuel was originally enriched.

Continuous attention is being given by the IAEA to this issue since 1996 when IAEA Director General, sent a letter to the United States of America Energy Secretary that helped to revive the U.S. “take-back” programme, that, from 1963 to 1989, shipped back to the United States of America more than 12.000 spent fuel elements from countries in Africa, Europe, America and Asia.

This report gathers, in one single IAEA publication, the requirements for technical and administrative preparations for shipment of Research Reactor Spent Fuel to the United States of America, and an important number of national experiences in shipping back fuel to the country of origin.

The main purposes of this publication are to disseminate information on good practices on the subject and make available to operators and managers of research reactors orientation on the basic methods and activities that serve as the preparatory framework for implementing the shipments, and to capture the lessons learned from previous successful shipments of research reactor spent fuel to the U.S.A and the Russian Federation.

The IAEA wishes to thank all those who participated in the Technical Meeting and contributed with their presentations and discussions for this publication. The IAEA officials responsible for this publication were P. Adelfang, A. Soares and D. Jinchuk from the Division of Nuclear Fuel Cycle and Waste Technology.

EDITORIAL NOTE

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SUMMARY

1. INTRODUCTION

The back-end of the Research Reactor (RR) nuclear fuel cycle is not solely a technical issue. Non-proliferation, physical security and environmental concerns are equally as important, if not more so, as technical concerns such as safe management of Research Reactor Spent Nuclear Fuel (RRSNF), storage capacity, availability of qualified high-density reprocessable fuel, and national self-sufficiency to deal with the domestic turnover of RRSNF.

At present, all aspects of the back-end of the RR nuclear fuel cycle are capturing increasing interest in Member States that operate RRs and are being extensively discussed in the major meetings of the international RR community.

International activities in the back-end of the RR nuclear fuel cycle are at present dominated by the RR spent fuel take back programmes, the United States of America Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) acceptance programme and the Russian Research Reactor Fuel Return (RRRFR) programme. The major goal of the separate take-back programmes for USA and Russian origin fuels is to eliminate inventories of Highly Enriched Uranium (HEU) by returning RR spent nuclear fuel to the country where the fuel was originally enriched.

The US FRRSNF acceptance programme is a longstanding initiative launched originally in 1996 to accept US enriched fuel irradiated by May 2006, and returned by May 2009. However, a revised record of decision extended these dates to May 2016, and May 2019, respectively. At the end of 2007, the programme had completed safely and successfully 41 shipments. Twenty-seven countries have participated, returning a total of 8078 spent nuclear fuel elements to the United States, most of it of highly enriched Uranium.

The programme has been fully supported by the IAEA since its inception, when the Agency assisted in the preparation of a “Guidelines Document on Technical and Administrative Preparations Required for Shipment of Research Reactor Fuel to its Country of Origin” and organized two interregional training courses on the subject in cooperation with Argonne National Laboratory, one in January 1997 and the other in May 1999. The courses provided participants with technical, organizational, and administrative information needed to prepare spent research reactor fuel for shipment.

The more recent RRRFR programme that accepts fresh or spent fuel enriched in the former Soviet Union or the Russian Federation is also in operation at present, after a previous programme to return Soviet/Russian origin RR spent fuel to Mayak reprocessing facility was stopped in 1991. The RRRFR Programme originated in 1999 from a tri-partite initiative of the Russian Federation, the United States of America, and the International Atomic Energy Agency (IAEA). The programme experienced frequent challenges due to important modifications of the Russian laws, regulations, and procedures since the last shipments of RR spent fuel under the former programme in 1991. The initiative was complemented in May 2004 with an agreement signed between the Government of the United States of America and the Government of the Russian Federation Concerning Cooperation for the Transfer of Russian-Produced Research Reactor Nuclear Fuel to the Russian Federation.

In cooperation with the RRRFR, under the IAEA Technical Cooperation Programme, a total of 446 kg of fresh HEU fuel have been removed from Serbia, Romania, Bulgaria, Libya, Uzbekistan, Czech Republic, Poland Germany and Vietnam returned to the Russian

Federation at the time of writing. The IAEA successfully completed, also under the Technical Cooperation Programme a 4.000.000 Euro procurement of 10 dual purpose (transportation and storage) high capacity casks that are at present available on a "lease-free" basis to the RRRFR programme.

In 2006 the first shipment of RR spent fuel under RRRFR was completed from a research reactor in Uzbekistan, 252 spent fuel assemblies, containing over 63 kg of HEU, were safely returned to the Russian Federation. The second shipment of spent fuel under the RRRFR was carried out on December 2007 from the Nuclear Research Institute in Rez, Czech Republic and consisted of 80 kg of spent HEU fuel and 280 kg of spent LEU fuel. The cargo arrived safe on 8 December 2007 at Mayak in the Russian Federation, after passing through Slovakia, the Ukraine and the European part of Russia. This is the first research reactor spent fuel shipment where the high capacity casks purchased by the IAEA for the RRRFR have been used.

There are still a large number of fuel assemblies eligible to be returned under these two programmes and it is envisaged that many more shipments will take place before they cease.

2. ORGANIZATION OF THE PUBLICATION

The publication is divided in two parts, **the first part** includes two papers, one covering generic issues to be considered by countries interested in shipping spent fuel within the framework of the US FRRSNF acceptance programme, and the other with an update of the well established FRRSNF acceptance programme document "APPENDIX A", including instructions on how to complete it. These documents provide key information for planning and conducting shipments of research reactor spent nuclear fuel back to the United States of America (country where it was originally enriched) and are intended for use by all parties involved in the planning, preparations, coordination and operations associated with returning spent nuclear fuel to the US.

Appendix A", is the document used by the US Department of Energy (DOE) for a fully characterization of spent nuclear fuel to be returned to the United States under the FRRSNF acceptance programme. The document is used to identify the physical, chemical, and isotopic characteristics of the fuel, Research reactor personnel have many decisions to make and activities to complete long before making a shipment of spent nuclear fuel. In many ways, the completed "Appendix A" is the starting point from which these decisions and activities are based. For instance, a completed "Appendix A" will clearly help in performing, among other tasks, a thorough evaluation of a potential shipping cask. It contains all necessary information on the spent fuel required to determine if a cask license authorizes, or needs to be amended to transport the spent fuel in question. Besides, a complete and accurate Appendix A contains essential information for the interim storage facility in the US. Data in Appendix A is used as input to carry out the analysis for safe handling and interim storage of the fuel at the interim storage facility. Ultimately, the Appendix A may be used as an information source for the data package for final disposition of the spent nuclear fuel in the United States of America.

The second part consists in the proceedings of the Technical Meeting on National Experiences on Return of Research Reactor Spent Fuel to the Country of Origin held on 28 – 31 August 2006, in Vienna, which captures almost all the experience accumulated so far in shipping back research reactor spent fuel to the country of origin worldwide and identifies and discusses the basic methods and activities that serve as the preparatory framework for implementing the shipments.

3. PROCEEDINGS OF THE MEETING

Under the regular budget activity on updating the guidelines document on the technical and administrative procedures required for the shipment of spent fuel from RRs a Technical Meeting on National Experiences on Return of Research Reactor Spent Fuel to the Country of Origin was held in Vienna on August 28–31, 2006. The main purpose of the meeting, that was partially supported by an extrabudgetary contribution from US DOE National Nuclear Security Administration (NNSA), was to allow operators and managers of research reactors that have shipped nuclear fuel to the country of origin to describe their experiences, exchange information and transfer lessons learned to managers and operators of research reactors that have not make any shipment yet and are considering the return of their spent nuclear fuel to the country of origin.

The meting was attended by 46 experts from 27 Member States The participants provided thirty seven technical presentations that can be classified into 6 categories: The history of the U.S. research reactor spent nuclear fuel (RRSNF) programmes since 1956 (1 presentation); characteristics of the U.S. foreign research reactor spent nuclear fuel (FRRSNF) acceptance programme (5 presentations); national experiences on shipments of RRSNF to the U. S. (20 presentations); methodology and infrastructure available for shipment of RRSNF to the U. S. (5 presentations); characteristics of RRSNF and infrastructure available at facilities that have fuel eligible for the FRRSNF acceptance programme, but have not made any shipment yet (5 presentations); and the role of the IAEA on assisting member states on activities related to returning RRSNF to the country of origin (1 presentation).

4. CONCLUSIONS

Although each shipment operation is unique and has its own peculiarities, some common lessons can be extracted from the different papers presented:

- The licensing activities usually require lead time, and it is necessary to initiate the process well in advance;
- It is essential to identify from the beginning of the process the different local authorities who are responsible for the decisions on the diverse issues related to the operation;
- It is very important to establish a very good communication process among all participants and authorities involved;
- Special attention should be devoted to “non technical” issues such as negotiation of contractual matters, safeguards, managerial activities, security, budget, cost scheduling and public relation with the media;
- Key to success of a shipment operation are strong coordination and collaboration between the local Organization and DOE staff;
- Due to the diversity of tasks to be undertaken by the reactor operator it is recommended to work under a centralized and vertical organizational scheme, appointing a general manager of the operation strongly supported by the reactor manager.
- Whenever possible it is better to integrate various shipments in a common campaign. A positive effect of such logistic approach is the reduction in transport costs for each involved reactor station.

It is evident from the individual presentations that every shipment operation is unique and likely to face different technical and administrative challenges. Different scenarios will require specific approaches in relation to licensing processes, multinational agreements, casks loading issues, transport procedures, public acceptance and also political aspects. Nevertheless, based on the experience accumulated so far worldwide, it is possible to conclude that both programmes, FRRSNF and RRRFR are being quite successful in safely transporting RRSNF back to the country of origin. In this way, these programmes are efficiently contributing to the global objective of minimizing and eventually eliminating the use of non-proliferation concerning nuclear materials, especially HEU, in civilian applications.

Presentations

Foreign Research Reactor (FRR) Spent Nuclear Fuel (SNF) Acceptance Program

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Abstract. The Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel, adopted by The United States Department of Energy (DOE), in consultation with the Department of State (DOS) in May 1996, has been extended to expire May 12, 2016, providing an additional 10 years to return fuel to the U. S. This paper provides a brief update on the program, now transitioned to the National Nuclear Security Administration (NNSA), and discusses program initiatives and future activities. The goal of the program continues to be recovery of nuclear materials, which could otherwise be used in weapons, while assisting other countries to enjoy the benefits of nuclear technology. The NNSA is seeking feedback from research reactor (RR) operators to help us understand ways to include eligible RRs who have not yet participated in the program.

1. Introduction

This paper presents the Foreign Research Reactor (FRR) Spent Nuclear Fuel (SNF) Acceptance Program, (the Acceptance Program). After an initial discussion of program history, contract extension and compliance are discussed. Planning issues are then set out to incorporate lessons learned from recent shipments in order to help FRRs understand issues which may assist in achieving their objective of proper disposition of SNF. The final discussion topic is DOE efforts to advance the goals of the Acceptance Program, with a conclusion that the Acceptance Program wants to work with FRRs to plan for shipment of their eligible spent fuel as early as possible.

2. Acceptance Program Metrics

The Acceptance Program, now in the tenth year of implementation, has completed 35 shipments to date, safely and successfully. Twenty-seven countries have participated so far, returning a total of 7 145 spent nuclear fuel elements to the United States for management at Department of Energy (DOE) sites in South Carolina and Idaho, pending final disposition in a geologic repository. Twenty eight (28) of the 35 shipments contained aluminium-based spent nuclear fuel from research reactors and were placed into storage at the Savannah River Site (SRS) in South Carolina. One shipment was forwarded on to the Y-12 National Security Complex, since the fuel was only slightly irradiated and eligible for receipt at that facility. The remaining six (6) shipments were placed into storage at the Idaho National Laboratory (INL) (Fig 1).

3. Contractual requirements

3.1. Contract extensions

DOE believes that all contract extensions, required to support reactor conversion and continued operation after May 2006, have now been signed. Other research reactors which have already converted to LEU fuel will need a contract extension to authorize shipments wanted after May, 2009. DOE intends to modify these contracts with priority given to those who are scheduled to ship in the near future. Reactor Operators in this situation are strongly encouraged to coordinate with the Acceptance Program office to negotiate the extension of the FRR-DOE contract to authorize continued Acceptance Program participation.

FIG. 1. Shipments to the United States of America.

3.2. Contract implementation

DOE enters into a contract with each of the customers who return SNF to the United States. It is very important that the contracting parties clearly understand all of the provisions in the contract. Contract requirements are usually described in detail prior to the first shipment. Significant contractual and programmatic requirements are shown in Annex I to this paper. As time passes and personnel change, some understanding may be lost. Further discussions on contract requirements can always be addressed to the Acceptance Program office. Compliance with all contract requirements must be maintained. One important article which has recently been misunderstood covers compliance with government regulations concerning public disclosure of any shipping plans or shipment information, or the individual details comprising such plans or information. Compliance with this article is an important obligation to support security for any shipment activity. During a recent shipment, a press release was made after the ship reached international waters on the way to the United States. DOE believes this is an unwarranted violation of the contract which made the security of the shipment more vulnerable. This premature release of information also violated the United States Nuclear Regulatory Commission regulations under which the shipments are authorized. Further, The Convention on the Physical Protection of Nuclear Material entered into by states which support the Acceptance Program requires that each state protect the confidentiality of this information. Our ability to continue this program depends on our customers following the agreed process.

3.3. Contract appendices

All FRR-DOE contracts contain one or more appendices. Contracts used to ship aluminium-based Material Test reactor type SNF to the DOE-Savannah River Site contain an Appendix A, Spent Nuclear Fuel Acceptance Criteria, and Appendix B, Transport Package (Cask) Acceptance Criteria. These documents have been updated and should be used on future shipment planning activities. The following paper in this part of the publication includes Appendix A, Revision 9 and Appendix B, Revision 10, and an extensive explanation how to fill the information requiered.

4. Focus on advance planning

The FRR SNF Acceptance Program focuses on the planning and implementation of these shipments of research reactor spent fuel to the United States in support of worldwide nuclear nonproliferation efforts, while assisting other countries to enjoy the benefits of nuclear technology. Along with shipment logistics, the DOE Office of Global Threat Reduction (GTR) continues to address many other issues of importance to the program.

4.1. Shipment scheduling

The most critical barrier to smooth operation associated with the program remains early scheduling and coordination of planned shipments. It is always important that NNSA clearly understands each Reactor Operator's intentions so that our planning can be well integrated and supported to meet the Reactor Operator's needs. It is also important to submit the required fuel data as early as possible in order to allow the receiving site adequate time to perform necessary reviews and prepare for receipt and storage.

Early availability of this data is also important for use in verifying transport package license requirements or submitting for a license amendment. Budget limitations could challenge implementation of shipping plans while NNSA and the Department of Energy receiving facilities also face increasing challenges in preparing to receive material, particularly when shipping plans are not well known.

As requested by many FRRs the program was extended to allow additional time for further development to LEU fuels and planning for back end solutions in the fuel cycle. The change was made to benefit the FRRs that needed justifiable relief. Some other FRRs are now taking advantage of these benefits by extending their shipping schedules to defer costs.

4.2. End –user assurances

Some countries require the issuance of an End-Use or Dual-Use Undertaking in order to obtain an export license. In the past, DOE provided that document to the reactor operator when requested. DOE no longer provides that document. However, assurances are already provided to those countries through the Agreements for Cooperation between each country and the United States when one exists or other avenues. The U.S. Department of State can validate those assurances to the participating country as necessary. It is recommended that these requirements be identified and resolved by the reactor operators as early as possible to ensure this political process is completed without shipment delays.

4.3. Insurance issues

One issue has been noted to be a problem for reactor operators in high-income economy countries who participate in joint shipments. Nuclear liability insurance associated with the ocean transport has the potential to adversely affect the total cost of shipping. This is because the shippers are sometimes required to have overlapping insurance coverage and also may have different requirements for minimum coverage. It is important for reactor operators to plan early for the required coverage and how to provide coverage in the least expensive manner. Consideration should be given for reactor operators entering into a joint shipment to coordinate in obtaining their nuclear liability insurance with the same pool or under a joint contract, where possible, in order to mitigate overlapping insurance costs. It is also important to be conscious of this potential problem and budget for any added cost that cannot be mitigated.

4.4. Cask license review

The Acceptance Program enjoys a very good working relationship with Nuclear Regulatory Commission (NRC) staff and wishes to take every measure possible to respect this relationship by ensuring that cask applications are timely and complete. DOE has been meeting periodically with NRC to discuss planned shipments and forecasted support required to meet the needs of the Acceptance Program and our customers. However, because there are limited resources for review of cask licenses, it is necessary for our customers to provide adequate time in the preparation process, scheduling for early application for review and approval of cask licenses.

5. Efforts to improve and accelerate

The Acceptance Program has now passed its approximate midpoint. More than ever before, DOE and reactor operators need to work together to schedule shipments as soon as possible, to optimise shipment efficiency over the remaining years of the program. Countries interested in participating in the Acceptance Program should express their interest as soon as possible so that fuel and facility assessments can be scheduled and shipments may be entered in the long-term shipment forecast. New and current Acceptance Program participants should also coordinate with DOE approximately 18 – 24 months in advance to ensure DOE can meet the Reactor Operator's plans and needs. Accelerated schedules are possible if there are no significant issues over past shipments. However, decreasing resources and coordination requirements with other agencies such as the Nuclear Regulatory Commission and Department of Transportation have the potential to limit DOE's capability to support these accelerated schedules. Specifically, the Acceptance Program may not be able to accommodate a large number of requests at the end of the program, particularly from geographically isolated regions.

5.1. Reorganization

The Office of Global Threat Reduction has reorganized in order to better use available resources and align the offices within three global regions and three cross-cutting program pillars. The regions include The Office of North & South American Threat Reduction (NA-211), Office of European & African Threat Reduction (NA-212), Office of Former Soviet Union and Asian Threat Reduction (NA-213).The organizational program pillars include Convert, Protect, and Remove. The FRR SNF Acceptance program, as a Remove function, is located under the Office of FSU and Asian Threat Reduction.

Although the program is managed under the Office of FSU and Asian Threat Reduction, the program operates globally across all regions. The program Technical Lead will continue to implement the program and will be the primary point-of-contact for this program. Regional Country Officers will assist in program coordination and shipment implementation. This reorganization should be essentially transparent to the reactor operator and other supporting shipment participants.

5.2. Material disposition

The DOE Environmental Management (DOE-EM) organization that used to manage the FRR SNF Acceptance Program is making strides to further disposition the repatriated spent nuclear fuel. The DOE-EM organization is considering continuing with the DOE Programmatic Spent Nuclear Fuel Environmental Impact Statement [1] and associated Record of Decision [2]. This decision included transporting fuel to place all aluminium clad spent fuel at the SRS and stainless steel fuel such as TRIGA fuel at INL. This allows for a potential decision to further treat the aluminium clad fuel in the H-Canyon facilities at SRS for disposition as waste in the same fashion as other high level waste material within the DOE complex. Any decision to further treat the material would be subject to further evaluation under the National Environmental Policy Act.

5.3. Potential fee changes

NNSA continues to evaluate ways to accelerate repatriation activities. Therefore, fees may change in the future and/or other changes may be implemented, if DOE believes the changes will positively influence program goals. DOE is also continuing to try to keep the reactor operator's cost to participate in the Acceptance Program low as possible. Any suggestions of methods to accelerate repatriation of SNF, especially Highly Enriched Uranium (HEU), would be welcomed and given all due consideration.

5.4. Coordination with other programs

A primary goal of the Acceptance Program is to support worldwide nonproliferation efforts by disposition of HEU which contains uranium enriched in the United States. Integral to this process is the U.S. assistance offered in helping reactor operators convert their cores to low enriched uranium (LEU) as the reduced enrichment fuels become qualified and available. In addition, DOE plays a strategic role in ensuring a supply of enriched uranium for fuel fabrication. In the Acceptance Program, the primary goal is intertwined with the missions of the Reduced Enrichment for Research and Test Reactors (RERTR) Program and the Enriched Uranium Operations group from DOE's Y-12 National Nuclear Security Complex in Oak Ridge, Tennessee. DOE Acceptance Program staff remain committed to working with staff in these other program offices within DOE and to do whatever is possible to assist in smooth transitions of core enrichment level and a steady supply of fuel.

6. Conclusion

The United States remains committed to supporting worldwide nonproliferation goals while assisting other countries to enjoy the benefits of nuclear technology such as those for which this program was designed. The programmatic goal is to accept eligible fuel sooner rather than later. Reactor operators are strongly encouraged to work closely with technical points-of-contact in order to ensure shipping schedules are accurate and achievable. The GTR staff hopes to work with all remaining eligible research reactors to plan for shipments of their eligible spent fuel as early as possible. NNSA continues to support research reactor operators' needs and would be happy to meet any interested parties to discuss the program.

REFERENCES

[1] Final Environmental Impact Statement for Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs DOE/EIS-0203-F (60 FR 20979, April 28, 1995).

[2] Record of Decision on the Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement (60 FR 28680, June 1, 1995).

ANNEX I

Foreign Research Reactor Spent Nuclear Fuel Acceptance Program

Requirements and Contracting Conditions

1. GENERAL REQUIREMENTS

Acceptance Policy

- Eligibility
  - Reactor should use fuel in which the uranium was enriched in the United States
  - Reactor startup before May 1996
  - Agree to participation based on the country's economic status as determined by the World Bank (including changes to economic status)
  - Enter into a contract outlining detailed responsibilities
  - Target material treated separately from fuel
  - Reactor Status:
    - Operate on or converting to LEU fuel
    - Reactors that are shut down
    - Operate HEU, but formally agree to convert to LEU within the policy period
    - Operate on HEU lifetime cores
    - With HEU cores that will shut down during policy
    - With HEU cores for which no suitable LEU fuel exists
    - Unirradiated HEU or LEU
    - All HEU must be received prior to receipt of LEU (except under extenuating circumstances)

Identification and use of HEU after Execution of the Contract

- If HEU fuel will continue to be used in the reactor after signing of the contract, specific milestones or HEU irradiation dates must be included in the contract
- Identification of all eligible material
  - Dependent on operating status HEU/LEU
  - Reactor operator should identify and request DOE to accept all eligible HEU spent nuclear fuel and HEU fresh nuclear fuel
  - Target material could qualify for return within a very limited time
  - Other stored fuel of U.S. origin

Fuel Condition

- Reactor operator shall identify material that is degraded, failed, or materially damaged
- DOE and Reactor operator shall determine shipping method
- Many transport packages (casks) are licensed to transport corroded or damaged fuel without encapsulation

Ineligible Material

- Cadmium
- Stainless steel components
- Thermocouples
- Other material that cannot be qualified as authorized material
- Some small non-aluminum materials as part of the fuel assembly may be accepted, but must be identified.

Joint Shipments

- Multiple casks from different countries are shipped on one vessel
- Important to minimize shipment costs
- Ship may pass through reactor operator's port with fuel from other countries
- Necessary to meet DOE's commitment to U.S. to minimize the number of total shipments into U.S.
- Nuclear Liability Insurance issues may be problematic for reactor operators in countries with high income economies
- Expected to be a significant cost savings to reactor operators in high-income economy countries

Program Constraints

- Acceptance program ends May 12, 2019
- All material to be shipped must be removed from reactor or no longer irradiated by May 12, 2016
- If reactor operator has not shipped before May 2016, DOE will make every effort to accept eligible material but cannot guarantee the ability to receive all fuel in the last few months of the program
- Reactor operator is expected to stop using HEU as soon as practical with continued shipments of LEU as desired through the end of the program

2. REQUIREMENTS FOR COUNTRIES WITH OTHER-THAN-HIGH-INCOME ECONOMIES

Taking Title/Liability

- Title normally changes at the reactor operator's port of embarkation
- Title can be transferred at a location from the departure of the reactor facility to off-loading at the U.S. port, to be determined on a case-by-case basis
- Specified in contract
- **Price-Anderson Act** In effect;
    - Within the United States territory
    - Outside the U.S. if the material is owned and controlled by the United States Government

DOE Subsidizes Cost for Shipment, Acceptance, and Management

- DOE subsidizes most activities associated with transport of the spent fuel
- Transportation subsidy is based upon reactor operator's capabilities
- DOE will not charge a DOE management fee for fuel storage and disposition
- Reactor operator performs its requirements without charge to DOE

Graduation to a High Income Economy

- Designation based on World Bank Development Report issued annually
- The reactor operator is expected to agree to modify the contract to incorporate clauses provided for countries with high-income economies to include:
    - Reactor operator shall pay for all packaging and transportation costs
    - Reactor operator shall pay DOE a management fee currently at the rate of $4,500 USD for TRIGA and MTR HEU (total mass) and $3,750 USD for MTR LEU (total mass)
        - Not changed since program began
        - Subject to change annually
        - Fee shall be set at the time the reactor operator notifies DOE at the designated shipping agent, but no more than 90 days prior to commencement of shipping
        - The reactor operator will have about one year after graduation to a high-income economy to complete any shipments under an other-than-high-income economy status

Responsibilities

- **Reactor operator's responsibilities**
    - Facility operational cost
    - Facility equipment maintenance cost
    - Facility security

    - Security during transit from facility to port
    - Assistance in coordination for the shipment of the loaded casks to port
    - Import/Export license
    - Loading personnel
    - Crane service in the facility
    - De-ionized water, electricity, compressed air
- **Reactor operators will assist DOE in:**
    - Determining fuel condition
    - Making arrangements for preparation of fuel
    - Completing transportation and fuel acceptance process
    - Coordinating within the reactor operator's country
    - In country transport, security, and regulatory assistance
    - In some cases, contract with local vendors on behalf of DOE

Work Activities

- Reactor operator shall crop (cut) and load SNF with assistance from the transportation services contractor that is provided by DOE
- Reactor operator shall provide a certification of physical condition of the fuel prior to final shipment planning and immediately upon fuel loading in the transport package
- DOE will provide casks, specialized loading equipment, and fuel cutting equipment
- DOE normally will provide any extra equipment necessary that is not available to the reactor operator

Physical Protection

- Reactor operator shall assist DOE in security planning within their country
- Reactor operator is responsible for coordinating and providing normal physical protection of the shipment while in country
- DOE will provide additional protection, coordinated through the reactor operator's security point-of-contact or organization, if required based on the results of the security plan

Import/Export Licenses

- Reactor operator will obtain required licenses in country
- Shipper may assist
- DOE disposes of SNF and considers that the material has no value and therefore no value should be assigned to export licenses

- Reactor operators should identify any safeguards requirements under any applicable agreement for cooperation documents, such as with the IAEA.

Shipment Not Yet in Transit (not departed the reactor site)

- DOE may postpone a shipment for a reasonable cause
- Shipment rescheduled for earliest practical date
- Reactor operator cannot charge for temporary storage at facility
- DOE will provide for all other reasonable cost associated with delay

Shipment In Transit (shipment still in-country)

- If delayed, reactor operator and DOE will find a location for temporary storage
- May be at reactor operator's facility
- Reactor operator cannot charge for temporary storage
- DOE will provide all other reasonable cost associated with the delay

Shipment In Transit (shipment departed country)

- If delayed, reactor operator and DOE will find a location for temporary storage
- Once entered into international waters, shipment is not expected to return to the reactor operator's country unless emergency circumstances require vessel assistance in accordance with international rules
- DOE will provide all cost associated with the delay

3. REQUIREMENTS FOR COUNTRIES WITH HIGH-INCOME ECONOMIES

Taking Title/Liability

- Title changes at U.S. port of entry
  - Naval Weapons Station-Charleston when arriving by ship
  - At the U.S. border when arriving by truck
- Price-Anderson Act In effect;
  - Within the United States territory
  - Outside the U.S. if the material is owned and controlled by the United States Government
- Reactor operators are responsible until shipment arrives at the DOE receiving site
- Reactor operators contracts appropriate transport contractors for delivery to the receiving site.

Financing

- Up to $4,500 per kilogram of total mass for TRIGA and MTR HEU spent fuel and $3,750 for MTR LEU spent fuel
- Transportation costs borne by reactor operator

- No change to the DOE Management Fee since the start of the program
- Subject to change annually
- Fee shall be set at the time the reactor operator notifies DOE of the designated shipping agent, but no more than 90 days prior to commencement of shipping
- DOE looking to lessen the burden of transportation costs

Responsibility

- The reactor operator is responsible for shipping the material to the Savannah River Site (SRS) or Idaho National Laboratory (INL) in the U.S. and must meet all packaging and shipping requirements
- Some fresh or slightly irradiated fuel may be shipped directly to the Y-12 National Security Complex

Physical Protection

- Reactor operator is responsible for physical protection during shipment in accordance with all regulations
- DOE provides security inside the United States of America without charge to the reactor operator

Work Activities

- Reactor operator shall crop (cut) and load
- Reactor operator's contractor or cask vendor usually can assist

Shipment Not Yet in Transit (not departed the reactor site)

- DOE may postpone a shipment for a reasonable cause
- Reactor operator shall pay for costs associated with postponement
- Reactor operator cannot charge for temporary storage at facility
- Shipment will be rescheduled at the earliest practical date

Shipment in Transit

- Reactor operator and DOE will work together to identify a location for temporary storage
- Temporary storage at the reactor operator's facility may be required
- Reactor operator and DOE share cost delays if not a legal impediment
- DOE pays for delay if directed by DOE
- DOE is not responsible for another reactor operator's delay
- Once title is transferred, Authorized material will not be returned to the reactor operator

US Department of Energy “Appendix A” spent nuclear fuel characterization

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Abstract. The US Department of Energy uses the *Appendix A, Spent Nuclear Fuel Acceptance Criteria*, to identify the physical, chemical, and isotopic characteristics of spent nuclear fuel to be returned to the United States under the Foreign Research Reactor Spent Nuclear Fuel Acceptance Program. The purpose of this paper is to provide guidance to reactor facility personnel responsible for preparing an Appendix A. Reactor facility personnel have many decisions to make and activities to complete long before making a shipment of spent nuclear fuel. In many ways, the completed Appendix A is the starting point from which these decisions and activities are based. Reactor facility personnel will find the overall shipment process easier by investing time early to complete the Appendix A. For instance, a completed Appendix A will clearly aid in a more thorough evaluation of potential shipping casks. This is because it contains all the necessary fuel characteristics required to determine if a cask license authorizes, or can be amended to include, the fuel as contents. Completing the Appendix A early in the overall shipment process will help the reactor facility make decisions, like cask selection, which may result in avoiding unnecessary costs or schedule delays. Completing an Appendix A can appear to be a daunting task, however, Washington Savannah River Company (WSRC) will work with reactor facility technical personnel during the Appendix A development and acceptance process. A complete and accurate Appendix A is essential to WSRC because the data is used as input to analysis for safe handling and interim storage of the fuel at the Savannah River Site. Ultimately, the Appendix A will be used as part of a data package for final disposition of the spent nuclear fuel in the United States.

1. Introduction

Since 1996, the Savannah River Site (SRS) has successfully received 169 spent fuel shipping casks containing over 5,900 aluminium-based assemblies in support of the Foreign Research Reactor (FRR) Spent nuclear Fuel (SNF) Acceptance Program. The success of this program is built upon the advance planning and coordination of many experts in field of nuclear material transportation. The common denominator throughout shipment planning and execution process is the requirement to clearly understand the characteristics of the spent nuclear fuel to be transported. The US Department of Energy uses the Appendix A, Spent Nuclear Fuel Acceptance Criteria [1], to identify the physical, chemical, and isotopic characteristics of spent nuclear fuel to be returned to the United States under the Foreign Research Reactor Spent Nuclear Fuel Acceptance Program. Over 250 Appendix A documents have been generated describing SNF from 23 countries. Washington Savannah River Company (WSRC) is responsible for the review and acceptance of Appendix A data for aluminium-based fuel to be received, stored and ultimately dispositioned at the Savannah River Site.

This paper provides detailed guidance on how to complete the Appendix A and summarizes the sequence of activities leading to WSRC acceptance of the document. Figure 1 shows the activities starting after the FRR operator and the DOE sign a contract agreeing to the terms and conditions for acceptance of SNF at the Savannah River Site. The paper concludes with Washington Savannah River Company (WSRC) acceptance of the Appendix A.

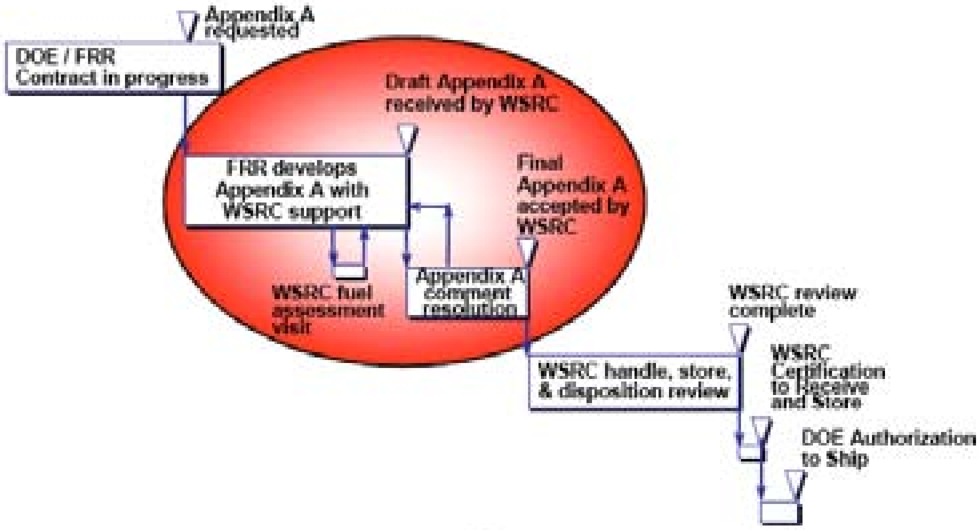

FIG. 1. Generic Appendix A preparation activities.

2. Generic Appendix A preparation activities

Each shipment of SNF has unique issues that need to be identified as early in the process as possible so shipment plans and schedules are not impacted. Research reactors participating for the first time, or those resuming shipments after many years, or those planning to ship new fuel types, should ensure they have a contract in place with DOE-SR 12 to 18 months in advance of the shipment. After a research reactor completes a shipment, preparation time for subsequent shipments can be significantly reduced if more of the same fuel types are to be shipped. Research reactor management has several opportunities each year to meet in person with DOE-NNSA FRR Program management to discuss proposed shipping schedules and SNF technical issues. Both DOE and WSRC FRR Program management attend the Reduced Enrichment for Research and Test Reactors (RERTR) and the Research Reactor Fuel Management (RRFM) conferences every year.

2.1. WSRC fuel assessment visit

WSRC personnel provide Appendix A development assistance to reactor facilities shipping SNF to SRS under the FRR program. A key activity for reactor facilities making first time shipments is the WSRC fuel assessment visit. Timing of the visit and facility access requirements are agreed to in advance between facility management and WSRC. WSRC will work with a facility point of contact in advance of the visit to collect preliminary fuel data and determine the scope of the assessment. The purpose of the visit is to visually inspect candidate fuel, prior to finalizing an Appendix A, to ensure it will be accurately described and to verify its condition. The amount of fuel to be inspected is determined based on the quantity and types to be shipped. Single shipments of small quantities of fuel are inspected 100% whereas large quantities of fuel to be shipped in multiple shipments utilize a sampling plan. A typical visit to assess fuel known to be in good condition would require three days, one day of discussions and two days of fuel assessment. The WSRC assessment team will normally only bring visual aids such as binoculars and digital cameras to assess and document the condition of the fuel. If necessary, WSRC can conduct extensive fuel assessments when the condition of the fuel is known to be poor or highly suspect. In such cases, WSRC can use specialized visual and remote inspection equipment, additional assessment personnel, and a customized fuel assessment plan based on specific facility needs. Support from the reactor facility during the assessment typically includes an operator to move fuel, health physics coverage, and an engineer for technical discussions and data collection.

2.2. *WSRC fuel assessment criteria*

WSRC has two primary criteria to satisfy during a fuel assessment. One is to obtain fuel drawings and specifications containing sufficient detail to confirm Appendix A fuel data by calculation. Drawings are matched to the fuel during the visual assessment to confirm all fuel assembly component parts are represented. Drawings are also used to confirm the fuel identification scheme shown on the drawing will match the fuel, especially if the fuel is going to be cropped. In some cases, fuel identification is removed during cropping and an alternate identification scheme needs to be developed. The second is to verify the structural integrity of the fuel to ensure it can be safely handled and stored at SRS. It is essential that the fuel be structurally intact so that fuel geometry does not change during transport or subsequent handling. The structural assessment will be checking for loose fuel plates or component parts or fuel assemblies that are deformed beyond drawing tolerances. The visual assessment will check for signs of exposed fuel meat. Indication of exposed fuel meat can include punctures or cuts due to handling or cropping. Fuel will also be assessed for signs of corrosion that could indicate through clad penetration. Generally, the presence of light crevice corrosion or small (<3mm) nodules on observable exterior fuel plates does not mean the cladding has been penetrated. Heavy crevice corrosion and nodules >3mm on observable exterior fuel plates may indicate the cladding has been penetrated and additional evaluation is required prior to acceptance of this fuel by DOE. Notification of DOE by the reactor facility is required for fuel that is suspected of being failed or warped. Notification requirements are identified in both the Appendix A and B [2] to the reactor facilities contract with DOE. When suspect failed fuel is identified it is not automatically excluded from the shipment, but it does require additional review to determine how it can be handled and whether it meets the failed fuel criteria of the proposed shipping cask. Depending on cask licensing, fuel that is considered failed but structurally intact may be transported without treatment if the total surface area of the exposed fuel meat is less than the limit allowed by the cask license. Fuel that is not structurally intact, including loose plates, or is above exposed fuel meat limits for the cask may be transported if it is placed in a canister. The shipper is required to notify and obtain approval of DOE-SR if fuel is considered to be failed. Both the Appendix A and B have failed fuel criteria which must be followed in order to ship failed fuel. The specific canister design shall be approved by DOE-SR. In all cases, it is the responsibility of the shipper to ensure the fuel is within the limits of the proposed shipping package. The WSRC fuel assessment visit is an essential part of the Appendix A development process. Once completed, the Appendix A benefits both the reactor facility and WSRC because it ensures the fuel to be transported is accurately characterized for safe handling, transportation, storage and disposition.

3. Appendix A development

WSRC is prepared to provide assistance to reactor facility personnel during the Appendix A development process. Reactor facility personnel begin the process by collecting the fuel drawings, specifications, and irradiation history to be used as references for extracting the requested data. Persons preparing the Appendix A should keep notes on which references are used as the source of the data being provided. The facility should provide WSRC with a preliminary draft Appendix A, along with developmental references and Table G Irradiation History, before the fuel assessment visit. WSRC will review the draft against the references, perform calculations to confirm weights and isotopic data, and provide comments back to the facility. For first time participants, WSRC and the preparer of the Appendix A can resolve comments during the facility fuel assessment visit. Otherwise, the comment / resolution process is conducted by email. When all comments have been resolved, WSRC will inform the facility that the Appendix A has been accepted.

The following information describes the Appendix A section by section. DOE-SR is currently working to Appendix A, Revision 9 (7/07), see Attachment 1, and the instructions contained therein take precedence if there is any conflict with the guidance within this document.

3.1. General Appendix A preparation guidance

(a) Weights must be in grams.

(b) Dimensions must be in centimetres.

(c) Weights and dimensions must be nominal values.

(d) Numeric data must be presented consistently using the appropriate number of decimal places in order to bound the data being provided and minimize rounding errors.

(e) Uncertainties must be provided in grams (g) or percent (%) as indicated. The preparer must ensure uncertainties bound the data they represent. For example: if the fuel assembly description of total weight of U is 399.95g +/- 2.1g then the assembly Table G data for pre-irradiated U grams must fall within the range of 397.85g to 402.05g.

(f) Total U and ^{235}U values specified under Fuel Element Description and Fuel Assembly Description must be Beginning of Life (BOL) values.

(g) A separate Appendix A is normally prepared for each fuel element or assembly having a different number of plates, length, ^{235}U content, or uranium enrichment.

(h) Use 'Not Applicable' (N/A) in data lines that do not apply to the fuel being described.

3.2. Specific Appendix A section guidance

HEADER

DOE-SR will provide the Appendix A identifying numbers, revision numbers, and contract number information.

SECTION A. CORRESPONDENCE

1. Customer Contacts

Provide required administrative data.

SECTION B. DEFINITIONS

Appendix A Definitions

Fuel Element – The smallest integral unit of clad fuel (e.g., plate, tube, rod, disc, etc.)
Fuel Assembly – A group of elements that are combined in a structural unit. The assembly is usually the fuel structure which is removed from the reactor as an individual unit. The fuel assembly consists of elements and other components such as end fittings, side plates, combs, spacers, guide and dummy plates, screws, welding material, canning material, etc.

Additional clarifying definitions not found in the Appendix A

As-shipped configuration –This includes cropping, canning, or reassembly of loose elements into an assembly by use of fasteners or canisters to facilitate storage or transportation. Appendix A fuel element and assembly weights and dimensional data must reflect the "as-shipped" configuration of the fuel.
Assembly Cross Section - The dimensions of the smallest rectangle that contains the assembly. This includes the convex portion of the fuel plates that extends past the end of the side plates for curved plate assemblies.

Cladding – All aluminium within an element except any aluminium present in the fuel meat.
Cropping – Cutting off of the non-fuel portions of either end of the assembly done to facilitate storage or transportation of the fuel. Appendix A fuel element and assembly weights and dimensions must reflect the "cropped" condition.
Loose Elements - Individual elements (plates, tubes, etc.) that are not contained within an assembly.

SECTION C. FORM AND COMPOSITION OF SPECIFICATION MATERIAL

1. Drawing Identification

List the drawings used to identify the element and assembly data provided in the Appendix A. The list should include drawings of the complete assembly, individual fuel elements, and fuel assembly components. If the fuel is cropped, a mark up drawing showing the dimensions for the cropping lines and location of the fuel meat should be provided. Canister or clip drawings should also be provided if they are used to ship loose elements. Provide the drawing number, revision number and/or date, and title for each. If available, also list and provide the fuel fabrication specification or manufacturer's fuel data sheets. WSRC will use the documents provided to confirm the Appendix A fuel physical data.

2. Material Description

This section is used as input to perform criticality analysis for safe handling and storage of the fuel at SRS. Fuel assembly cross section and the amount of aluminium in the fuel region will effect criticality analysis results. Accurate data is needed to ensure that over 10 000 spent fuel assemblies of more than 300 types are safely stored at SRS.

2.1. Fuel 'Element' Description

Line 1: Fuel element type, i.e., curved or flat plate, disc, rod, tube, etc.
Line 2: Chemical form of fuel meat, i.e., U_3O_8-Al, UAl_x-alloy, UAl_x-Al, U_3Si_2-Al, as applicable.
Line 3: Nominal dimensions of fuel meat[1].
Line 4: Weight of ^{235}U in an element with uncertainties.
Line 5: Weight of total U in an element with uncertainties.
Line 6: Alloy or compound material weights are as follows:
- U_3O_8-Al fuel meat: Weight of O_8 in U_3O_8.
- UAl_x-alloy fuel meat: Weight of Al_x in UAl_x.
- UAl_x-Al fuel meat: Weight of Al_x-Al in UAl_x-Al[2].
- U_3Si_2-Al fuel meat: Weight of Si_2 in U_3Si_2.

Line 7: Dispersing material weights are as follows:
- U_3O_8-Al fuel meat: Weight of -Al in U_3O_8-Al.
- UAl_x-alloy fuel meat: NA
- UAl_x-Al fuel meat: NA (See Line 9 above).
- U_3Si_2-Al fuel meat: Weight of -Al in U_3Si_2-Al.

Line 8: Nominal total weight of the fuel meat. (Line 8 = Line 5 + Line 6 + Line 7)
Line 9: Cladding material (Aluminium) and method of sealing.
Line 10: Clad thickness and total clad weight. (see cladding definition)
Line 11: Bonding material, if any[3]. (Na, Al-Si, etc.)
Line 12. Bonding material thickness and weight.

---

[1] Dimensions for curved fuel meat shall be provided in the flat condition prior to any forming operations
[2] UAlx exists in this type of fuel meat in varying combinations of UAl2, UAl3, and UAl4. Without the specific contents of each, the amount of –Alx in the UAlx cannot be calculated. Therefore, for UAlx-Al fuel meat, provide under Line 6, the weight of –Alx in UAlx plus the weight of –Al (dispersing material) in UAlx-Al, which is all of the aluminium in the fuel meat. In turn, an entry under Line 7 (dispersing material) will not be required.
[3] Immediately notify DOE if Sodium is present.

Line 13: Provide quantities, material, dimensions, and weights (each) of other materials in an element, as applicable.
Line 14: Nominal dimensions of fuel element including clad and bond[4].
Line 15: Nominal total weight of fuel element[5].5 (Line 15 = Line 8 + Line 10 + Line 12 + Line 13)

2.2 Fuel 'Assembly' Description

Line 1: Number of elements per assembly. Differentiate between number of outer and inner plates where they differ in length and weight.
Line 2: Total weight of ^{235}U. (Line 2 = Line 2.1 4 x Line 1) with uncertainties.
Line 3: Total weight of U. (Line 3 = Line 2.1 5 x Line 1) with uncertainties.
Line 4: Enrichment (% +/- % uncertainty). Nominal ^{235}U enrichment % for assembly with uncertainties in %.
Line 5: Side plate material, if applicable.
Line 6: Side plate dimensions and weight per plate, [a] if applicable.
Line 7: Spacer material, i.e. combs, if applicable
Line 8: Spacer major dimensions and weight per spacer, if applicable.
Line 9: End boxes or fitting material, if applicable.
Line 10: End boxes or fitting dimensions and weight, if applicable.
Line 11: Braze or weld material, if applicable.
Line 12: Braze or weld dimensions and weight, if applicable.
Line 13: Other structural material in the assembly include material, quantity, dimensions, and weights; i.e., dummy and/or guide plates, screws, etc.c
Line 14: Canning material, [b] if applicable.
Line 15: Can dimensions and weight, [b] if applicable.
Line 16: Method of can sealing (screw, weld, etc.), if applicable.
Line 17: Over-all dimensions. [a] Assembly length (as shipped) and assembly cross-section.
Line 18: Over-all weight. [a] Assembly (as shipped) including all elements and other assembly components . (Line 18 = Line 6 + Line 8, Line + 10, Line + 12 , Line + 13, Line + 15)

Note (a): Is the assembly cropped? In many cases, the reactor facility may have already cropped the spent fuel in order to maximize their storage space. Cropping is also sometimes necessary to maximize the quantity of spent fuel that can fit into a particular transport package. Figure 2 shows a typical fuel assembly drawing marked to provide the dimensions of the cuts. A cropping drawing is important because:

(1) It documents the as-shipped configuration.

(2) The reduced amount of aluminium in the assembly is accounted for in criticality calculations for fuel storage at SRS.

(3) A cask license may have a limit on the minimum amount of non-fuel bearing material remaining at each end of the fuel.

(4) Cropping may remove part or all of the fuel assembly identification requiring the addition of identification tags or accounting for the portion of the identification remaining.

(5) The drawing identifies how structural integrity of the assembly will be maintained and how the fuel can be handled.

(6) Additional cropping may be done at SRS so the fuel meat must be accurately located.

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[4] Dimensions for curved plates shall be provided in the flat condition prior to any forming operations.
[5] Multiple entries are required when the fuel design includes outer and inner plates of different dimensions. See as-shipped configuration definition.

(7) The amount of aluminium remaining may become important if a fuel disposition decision includes chemical dissolution.

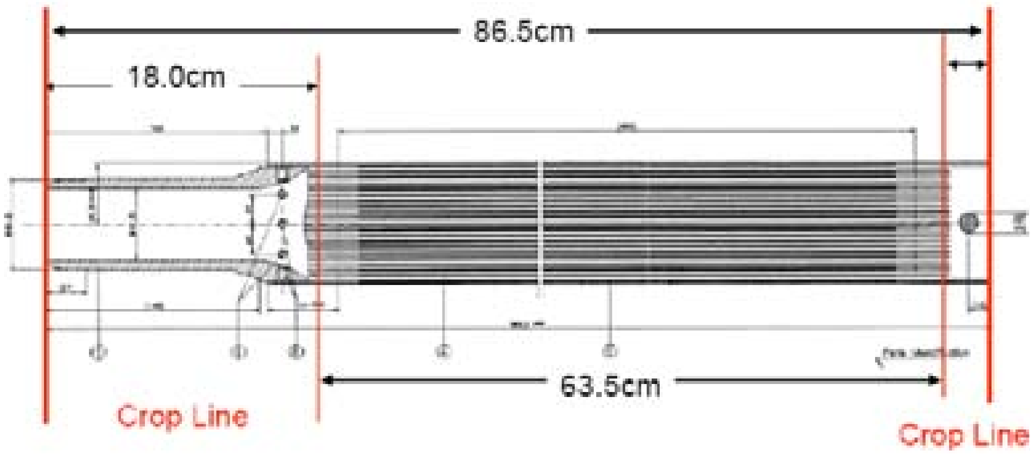

FIG. 2. Typical fuel assembly drawing marked to show cropping.

Note (b) When canning of fuel is required, describe can using these entries. The following shows information to be provided in the 'Fuel Assembly' section as it applies to the shipment of 'Loose Elements' in an canister or fastened together in a configuration other than as it was in the reactor. This is typical for ARGONAUT type fuels.

Line 1: Number of elements per assembly. Provide number of plates per can.
Line 2: Nominal ^{235}U weight/element x number of elements/can with uncertainties in grams.
Line 3: Total weight of U. Nominal total U weight/plate x number of elements/can with uncertainties in grams.
Line 4: Enrichment (% +/- % uncertainty). Nominal ^{235}U enrichment % for contents of can with uncertainties in %.
Lines 5 through 12: NA
Line 13: Other structural material in the assembly include material, quantity, dimensions, and weights; i.e., dummy and/or guide plates, screws, etc.c
Line 14: Canning material.
Line 15: Canning dimensions. Length x cross section x thickness or length x outer diameter x thickness, as applicable, with weight of can in grams.
Line 17: Over-all dimensions. Same as the dimension portion of Line 15.
Line 18: Over-all weight. Total weight of can and loose plates within.

Note (c) Non-fuel material connected/attached to the fuel assembly is not allowed unless specifically authorized by the Contracting Officer. The Customer shall provide DOE with a complete description of any non-fuel material (targets, irradiation materials, samples, thermocouples, dummy plates, wires, etc.) that is normally removed by the Customer from the element or assembly prior to shipment, but cannot be removed due to fuel failure, warpage, or other reasons.

2.3 Failed Fuel, Degraded Fuel, or Materially Damaged Fuel

For first time participants, the WSRC fuel assessment visit, described in Section 2.2, WSRC Fuel Assessment Criteria, is a good opportunity to document the condition of the fuel. When fuel condition issues are identified early, it allows time to assess the condition an make arrangements on how to handle the fuel without disrupting the shipping schedule. Appendix A, Section 2.3, has been revised to include a full description of Failed Fuel, Degraded Fuel, or Materially Damaged Fuel.

SECTION D. FUEL IDENTIFICATION

Each separately removable unit of fuel must be identified by a durable tag or by embossing. Provide a list of all assembly/can identification numbers for the assemblies/cans to be shipped to SRS. These numbers must be listed exactly as labelled on the assemblies/cans and must be identically listed on the cask loading diagram. Part of WSRC's review requires checking that fuel identified in the Appendix A exactly matches the fuel identification provided in the cask loading diagram. When fuel is unloaded at SRS, the loading diagram is used as positive identification that the fuel received is the same as the fuel authorized for receipt by the Appendix A. At SRS, if the identification on a received fuel assembly does not match the cask loading diagram or Appendix A, then the fuel must be isolated and a Limiting Condition for Operation (LCO) is entered for unidentified fuel. WSRC then must contact the shipping facility to resolve the matter which may cause delays in fuel unloading operations at SRS. Figure 3 shows how a typical fuel identification matches the Appendix A and Fuel Loading Diagram.

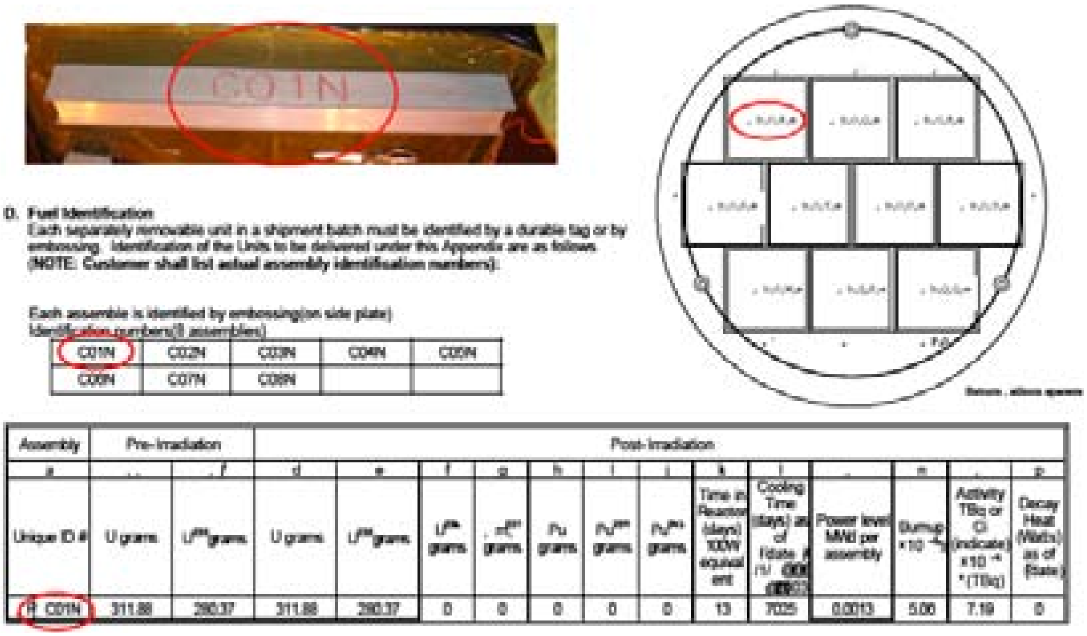

FIG. 3. Fuel Identification matches fuel, Appendix A, and loading diagram.

SECTION E. CASK AND BASKET INFORMATION

Facilities responsible for the selection of the cask are asked to provide the cask and basket types and number of assemblies/cans to be shipped as soon as possible. The Appendix B, Section J, Transport Package Design, identifies the transport packages that SRS is equipped to receive and are listed in Table 1 below. WSRC has developed procedures, designed rigging, and trained operators to handle each of the identified casks. The time it takes WSRC to unload each package varies because each cask requires special handling. Facilities may consult with WSRC if they have questions on how long an unloading sequence takes or if there are special operational considerations for a particular cask. If a facility chooses to use a cask not identified in Table 1, they must obtain prior approval of DOE-SR in accordance with the requirements of Appendix B.

For shipments of multiple packages from multiple facilities, DOE-SR should be advised by the shipping facility if there is a specific need for unloading and returning empty casks. DOE-SR will establish the order in which WSRC will unload casks based on these needs.

TABLE 1. TRANSPORT PACKAGES HANDLED AT SRS

BMI-1	JRC-80Y-20T
F-257	LHRL-120
GE-2000	NAC-LWT
GNS-16	NAC-NLI-1/2
GNS-11	TN-MTR
TN-6/3	TN-7/2
JMS-87Y-18.5T	JRF-90Y-950K
	(up to 2 casks per ISO container are authorized)

SECTION F. Reactor Operation and Fuel Irradiation General Information

Provide a general summary of normal reactor operations and how fuel is cycled through the core. The summary information in this section should reflect the assembly specific irradiation history of the fuel provided in Section G. The computer code used to determine the post irradiation mass inventory of heavy metals should be identified. WSRC will confirm the post irradiation mass inventory of heavy metals and thermal decay heat fall within the predicted ranges [3].

SECTION G. ASSEMBLY SPECIFIC FUEL IRRADIATION DATA

Irradiation history data is used to confirm the nuclear mass inventory of the fuel to be shipped. SRS uses the post-irradiation data to enter accountable quantities into the site inventory. These quantities are also used to ensure inventories remain within the Authorization Basis for accidents involving a radiological release and for maintaining the appropriate level of security for the amount of material stored.

Table G is used to record the unique identification numbers for all assemblies/canisters along with the stated pre- and post-irradiation data. Be sure to provide “as of” dates for cooling times and decay heats as well as which unit (TBq or Ci) is used to indicate activity.

4. Conclusions

Reactor facility personnel have many decisions to make and activities to complete long before making a shipment of spent nuclear fuel. In many ways, the completed Appendix A is the starting point from which these decisions and activities are based. Reactor facility personnel will find the overall shipment process easier by investing time early to complete the Appendix A. A completed Appendix A will clearly aid in a more thorough evaluation of potential shipping casks. This is because it contains all the necessary fuel characteristics required to determine if a cask license authorizes, or can be amended to include, the fuel as contents. Completing the Appendix A early in the overall shipment process will help the reactor facility make decisions, like cask selection, which may result in avoiding unnecessary costs or schedule delays. Completing an Appendix A can appear to be a daunting task, however, Washington Savannah River Company (WSRC) will work with reactor facility technical personnel during the Appendix A development and acceptance process. A complete and accurate Appendix A is essential to WSRC because the data is used as input to analysis for safe handling and interim storage of the fuel at the Savannah River Site. Ultimately, the Appendix A will be used as part of a data package for final disposition of the spent nuclear fuel in the United States.

REFERENCES

[1] UNITED STATES DEPARTMENT OF ENERGY, Appendix A, Spent Nuclear Fuel Acceptance Criteria, Revision 8, September 2000.

[2] UNITED STATES DEPARTMENT OF ENERGY, Appendix B, Transportation Package (Cask) Acceptance Criteria, Revision 9, September 2000.

[3] POND, R. B., MATOS, J.E., Nuclear Mass Inventory, Photo Dose Rate and Thermal Decay Heat of Spent Research Reactor Fuel Assemblies, ANL/RERTR/TM-26, Revision 2, Argonne National Laboratory, May 2000.

ATTACHMENT 1

SPENT NUCLEAR FUEL ACCEPTANCE CRITERIA

WITH:	
UNDER CONTRACT NUMBER:	

THIS APPENDIX to the contract referenced above, completed by the Customer, provides a detailed description of the material to be delivered to DOE in accordance with the requirements of this Appendix and the referenced contract and also enumerates the specifications and requirements which the Customer must meet. Failure of the material delivered hereunder to comply with the specifications and requirements given in this Appendix will void the designation as Authorized Material and will result in DOE's further consideration, based on the updated information, of the decision whether to accept the incorrectly described material as Authorized Material in accordance with Article III of this contract. The Customer shall submit a separate Appendix A for each element or assembly which is different in description including but not limited to a different number of plates (i.e. standard or control assemblies), length, U-235 content, or uranium enrichment.

A. Correspondence

1. Customer Contact information

Laboratory/ Research Center/University	
Reactor Name	
City, State, Country	
Customer Name	
Customer Signature*	
Title	
Phone Number	
Fax Number	
Email (if available)	
Date	

* The signature in this block indicates that the information provided is to the best of your knowledge correct and accurate and that any changes to this information will be provided to DOE as soon as possible in the form of a revision. This block must be signed when each revision of the Appendix A is submitted for review and approval. The DOE will issue a written "Authorization to Ship" prior to each shipment to indicate DOE's final approval of each applicable Appendix A and DOE's readiness to safely receive the Authorized Material.

2. Department of Energy Contact

All correspondence or inquiries regarding this document and the information contained herein shall be directed to:

U. S. Department of Energy
National Nuclear Security Administration
Foreign Research Reactor Spent Nuclear Fuel
Acceptance Program Office
Savannah River Site
Road 1
Aiken, South Carolina 29802

Phone and facsimile inquiries may be made to:
Phone: +803-952-5873 or 5874
Fax: +803-952-6115

B. Definitions

The following definitions are applied to the material described in this Appendix:

Fuel Element - The smallest integral unit of clad fuel (e.g., plate, tube, rod, disc, etc.).

Fuel Assembly - A group of elements that are combined in a structural unit. The assembly is usually the fuel structure which is removed from the reactor as an individual unit.

C. Form and Composition of Specification Material

1. Drawing Identification

The following drawing(s), six (6) copies of which are attached and which are incorporated herein by reference thereto, constitute(s) a comprehensive illustration in sufficient detail and accuracy of the fuel elements and assemblies to be delivered. One (1) copy of the applicable fuel specification report or manufacturer's data is included, if available. The Customer shall submit a separate Appendix A for each element or assembly which is different in description including but not limited to a different number of plates (i.e. standard or control assemblies), length, U-235 content, or uranium enrichment.

Drawing Number	Revision	Title

NOTE: If fuel is to be cropped (cut) for shipment, provide marked-up drawings with dimensions locating the cut(s) on the assembly and element, as applicable.

2. Material Description

The following tables summarize the description of fuel elements and assemblies to be delivered. Where dimensions are required, the nominal dimensions from the fuel element and assembly drawings must be used. If changes in dimensions have occurred due to cropping or other modification, the best estimate of the maximum change in these dimensions must be given. Weights must be dry, unirradiated weights with the expected range of weights also to be included. Where isotopic weights of uranium are required, tolerances shall be specified.

Non-fuel material connected/attached to the fuel assembly is not allowed unless specifically authorized by the Contracting Officer. The Customer shall provide DOE with a complete description in this appendix of any non-fuel material (targets, irradiation materials, samples, thermocouples, non-aluminum dummy plates, wires, etc.) approved for delivery.

2.1 Fuel 'Element' Description *(If more than one type of element per assembly, divide the space to describe each type of element or duplicate this page as necessary.)*

Provide dimensions in centimeters (cm) and weights in grams (g).

1.	Fuel element type (curved or flat plate, disc, rod, tube, etc.)	
2.	Chemical form of fuel meat (e.g., U_3O_8-Al, U-Al_x-alloy, UAl_x-Al, U_3Si_2-Al, etc.)	
3.	Dimensions of fuel meat (cm)	
4.	Weight of U-235 (g ± g uncertainty)	
5.	Weight of total U (g ± g uncertainty)	
6.	Alloy or compound material, weight (g)	
7.	Dispersing material, weight (g)	
8.	Total weight of fuel meat (g)	
9.	Cladding material & method of sealing	
10.	Clad thickness (cm), total clad weight	
11. [(a)]	Bonding material, if any (Na, Al-Si, etc.)	
12.	Bond thickness (cm), weight	
13.	Other items included with the fuel element: (include materials, quantities, dimensions (cm) and weights(g))	
	Will the <u>element</u> be cropped (cut) when it is shipped? Yes or No. (If yes, provide dimensions and weights in terms of the cropped element.)	
14. [(b) (c)]	Dimensions of element (cm) (include clad and bond,)	
15. [(c)]	Total weight of fuel element(g)	

Notes:
(a) The Customer will immediately notify DOE if Sodium is present.
(b) For curved plate elements, provide flat plate dimensions (before forming).
(c) Provide dimensions and weights in terms of cropped element if applicable.

2.2 Fuel 'Assembly' Description. (*Describe the assembly in its "as-shipped" form.)*

Provide dimensions in centimeters (cm) and weights in grams (g).

1.	Number of fuel elements per assembly	
2.	Total weight of U^{235} (g ± g uncertainty)	
3.	Total weight of U (g ± g uncertainty)	
4.	Enrichment (% ± % uncertainty)	
	Will the assembly be cropped (cut) when it is shipped? Yes or No. (If yes, provide dimensions and weights in terms of the cropped assembly.)	
5.	Side plate material	
6.[(a)]	Side plate - dimensions (cm), weight per plate (g)	
7.	Spacer material	
8.[(a)]	Spacer - dimensions (cm), weight per spacer (g)	
9.	End box or fitting material	
10.[(a)]	End box or fitting dimensions (cm), weight (g)	
11.	Braze or weld material	
12.	Braze or weld dimensions (cm), weight (g)	
13.[(c)]	Other items in assembly e.g. dummy plates, thermocouples, etc. (include materials, quantities, dimensions (cm), and weights) (g)	
	Will any fuel be shipped in a can? Yes or No (If yes, provide a can drawing(s) in section C.1. and provide can data below.)	
14.	Can material	
15.	Can dimensions (cm)	
16.	Can weight (g) (completely assembled and empty)	
17. (a)(b)	Over-all assembly dimensions (cm)	
18. (a)(b)	Over-all assembly weight (g)	

Notes:
(a) Provide dimensions and weights in terms of cropped assembly if applicable.
(b) Provide dimensions and weights in terms of can size and can plus assembly weight.
(c) Information is required if non-fuel material will be shipped with the fuel assembly and is approved by the Contracting Officer.

2.3 Failed Fuel, Degraded Fuel or Materially Damaged Fuel

The Customer is required to provide DOE with a complete description of each fuel unit known or suspected to be failed, degraded or physically damaged at least 180 days before delivery to DOE. Failed Fuel, Degraded Fuel or Materially Damaged Fuel includes, but is not limited to:

- Fuel deformed beyond reference drawing dimensional limits specified in 2.1 or 2.2.
- Fuel removed from the reactor due to cladding failure.
- Fuel meat exposed due to physical damage to or corrosion of the cladding.
- Fuel condition does not comply with applicable Transport Package certificate requirements.
- Fuel is not structurally sound and may change shape during transport or handling.
- Fuel is deformed and could cause interference with cask or basket surfaces.
- Any other failure or physical condition that may require special handling or packaging for transportation or storage.

Non-fuel material connected/attached to the fuel assembly is not allowed unless specifically authorized by the Contracting Officer. The Customer shall provide DOE with a complete description of any non-fuel material (targets, irradiation materials, samples, thermocouples, dummy plates, wires, etc.) that is normally removed by the Customer from the element or assembly prior to shipment, but cannot be removed due to fuel failure, warpage, or other reasons.

Any spent nuclear fuel not meeting this criteria will be considered to be Failed Fuel, Degraded Fuel or Materially Damaged Fuel as defined by Article I of the contract and will require special treatment up to Canning prior to loading in the cask for transport to SRS. All special treatment conducted must be authorized in writing by DOE.

2.3.1 Statement of fuel condition:

The Customer will describe the condition of the fuel considered to be Failed Fuel, Degraded Fuel or Materially Damaged Fuel and any special treatment (up to canning) authorized by DOE. OR, if there are no Failed Fuel, Degraded Fuel or Materially Damaged Fuel, make a statement to that effect here:

D. Fuel Identification

Each separately removable unit in a shipment batch (assembly or fuel can) must be identified by a durable tag or by embossing. The fuel **ID provided below must exactly match the ID on the assembly** as it is shipped. Identification of the fuel to be delivered under this Appendix A are as follows:

E. Cask and Basket Identification

The specific cask and basket type being shipped under this Appendix A need not be supplied upon the initial or subsequent submittals of an Appendix A. However, this data should be supplied by the Customer as soon as it is known and must be identified in the final submittal of the Appendix A and prior to DOE's issuance of the "Authorization to Ship". The final submittal of the Appendix A must include a preliminary cask loading diagram showing how each fuel assembly, by ID number, will be loaded in the cask basket(s).

Cask	Number of Baskets/Cask	Number of Assemblies/Cask

F. Reactor Operation and Fuel Irradiation General Information

In the space below, provide a brief summary of the reactor's fuel cycle which supports the assembly specific fuel irradiation history provided in Section G:

United States Department of Energy **APPENDIX A**

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G. Assembly Specific Fuel Irradiation Data
The irradiation history for ~~each~~ assembly is to be provided according to the format specified below. Additional tables may be attached, as necessary. Please submit this information electronically using the format provided in this document. Duplicate this page as needed.

~~Assembly~~	Pre-Irradiation		Post-Irradiation												
a	b	c	d	e	f	g	h	I	j	k	l	m	n	o	p
Unique ID #	U grams	U^{235} grams	U grams	U^{235} grams	U^{236} grams	Np^{237} grams	Pu grams	Pu^{239} grams	Pu^{241} grams	Time in Reactor (days)	* Cooling Time (days) as of: (date)	Power level mwd per assembly	Burnup %	Activity TBq or Ci (indicate)	Decay Heat (Watts) as of: (date)
Totals															

* Cooling time of less than 2 years upon arrival at SRS requires additional evaluation prior to acceptance.

Presentations

Part II

IAEA Assistance on return of research reactor spent fuel to the country of origin

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Abstract. The International Atomic Energy Agency has been involved for more than twenty years in supporting international nuclear non-proliferation efforts associated with reducing the amount of highly enriched uranium (HEU) in international commerce. IAEA projects and activities have directly supported the US "Reduced Enrichment for Research and Test Reactors" (RERTR) programme, the U.S. "Foreign Research Reactor Spent Nuclear Fuel Programme" (FRRSNF), as well as the "Russian Research Reactor Fuel Return Program" (RRRFR), to return research reactor fuel to the country of origin where it was originally enriched. Agency efforts have included the development and maintenance of several data bases with information related to research reactors (RRs) and research reactor spent fuel inventories that have been essential in planning and managing both RERTR and spent fuel return programmes. Other IAEA regular budget and Technical Cooperation activities have supported research reactor conversion and spent fuel return programmes.

After the announcement of the Global Threat Reduction Initiative (GTRI) by United States Secretary of Energy Spencer Abraham on May 2004 at the IAEA headquarters in Vienna and following recommendations of the 2004 RERTR meeting, held in Vienna in November 2004, IAEA support of the programmes of repatriation of research reactor fuel to the country of origin are being strengthened. A comprehensive number of new activities have been initiated and some others are being prepared.

Since 2004 eleven shipments have successfully taken place to remove and return to Russia around 430 kilograms of fresh and 63 kilograms of spent Russian-origin HEU and 37 shipments with more than 1,100 kg of US-origin HEU were return to the United States of America.

This paper briefly describes IAEA involvement since the early 1980's in these areas, including regular budget and Technical Cooperation programme activities, and focuses on efforts in the past five years to continue to support and accelerate U.S. and Russian Federation research reactor spent fuel return programmes [1] [2] [3].

1. Introduction

Research reactors have played an important role in the development of nuclear science and technology. However, of the more than 650 research reactors constructed around the world in the second half of the twentieth century, at the present time only 275 are operating. About 375 research reactors have been closed, of which a bit less than half (168) have been decommissioned. Further, of the 275 operating reactors, a significant number are under-utilized and may be closed in the near future. Spent fuel management is a major consideration for many facilities.

The IAEA has sought to address the changing needs of its member states in the research reactor field, by providing assistance with strategic planning for increased utilization, refurbishment, ageing management and spent fuel management. At the same time, it has also addressed emerging non-proliferation concerns related to research reactors, such as assisting in the reduction of the use of HEU.

The IAEA has been involved for many years in supporting international nuclear non-proliferation efforts associated with reducing the amount of HEU in international commerce. IAEA projects and activities have directly supported the U.S. Foreign Research Reactor Spent Nuclear Fuel (FRRSNF), as well as the "Russian Research Reactor Fuel Return Program" (RRRFR).

Agency efforts have included the development and maintenance of several databases with information related to research reactors and their spent fuel inventories that have been essential in planning spent fuel return programmes. Other IAEA regular budget programs have been highly useful in supporting research reactor fuel conversion from HEU to LEU, and in addressing issues common to many member states in dealing with spent fuel management problems and concerns.

After the announcement of the Global Threat Reduction Initiative (GTRI) by United States Secretary of Energy Spencer Abraham on May 2004 at the IAEA headquarters in Vienna and following recommendations of the 2004 RERTR meeting, held in Vienna in November 2004, IAEA support of repatriation of research reactor fuel to the country of origin has been strengthened.

Since 2004 eleven shipments have successfully taken place to remove and return to Russia around 430 kilograms of fresh and 63 kilograms of spent Russian-origin HEU and 37 shipments with more than 1,100 kg of US-origin HEU were return to the USA.

In association with IAEA support to research reactor fuel return programs (see below), the IAEA organized a series of fact-finding missions to research reactors to assess the fresh and spent fuel situations, under TC project RER/9/058, Safety Review of Research Reactor Facilities. The initial missions took place in 17-23 June 2001 to Ukraine, Uzbekistan, and Yugoslavia. Additional missions took place February 10-21, 2003 to Romania, Czech Republic and Latvia, March 16-22, 2003 to Kazakhstan, December 9-20, 2003 to Poland, Bulgaria, and Hungary and March 3-4, 2004 to Belarus. IAEA, Russian, and other international experts have taken part in the missions, which have established the basis for spent fuel shipments from these countries.

2. How is IAEA technical assistance delivered?

IAEA's assistance is delivered through two main mechanisms that are summarized in the following paragraphs.

2.1. Regular Programme

The main mechanism is through co-ordinated research activities. These co-ordinated research activities are normally implemented through Co-ordinated Research Projects (CRPs) that bring together research institutes in both developing and developed Member States to collaborate on the research topic of interest. The Agency may also respond to proposals from institutes for participation in the research activities by awarding individual contracts not related to a CRP. A small portion of available funds is used to finance individual projects, which deal with topics covered by the Agency's scientific programme. The Agency designates a Project Officer for the CRP, usually from its technical staff, who will liaise with the persons nominated as Chief Scientific Investigators for the participating institutes. Between them, they manage and liaise on the research programme, which has a duration normally of between 3 to 5 years. The Agency's Research Contracts Administration Section (NACA) of the Department of Nuclear Sciences and Applications is responsible for co-ordinating and administering the CRP financial and contractual arrangements.

Other useful mechanisms are participation in IAEA's Technical Meetings and Workshops and getting in touch with IAEA staff members to discuss and provide suggestions and ideas on new activities or required assistance.

2.2. Technical Cooperation Programme

From January 2005, a regional project, RER/4/028, "Repatriation, Management and Disposition of Fresh and/or Spent Nuclear Fuel from Research Reactors", is being used to handle all projects on fuel repatriation.. The objectives of this project are: (i) to assist Member States with research reactors to repatriate, manage or dispose of their fuel, fresh or irradiated; (ii) to support the Russian Research Reactor Fuel Return (RRRFR) programme; and iii) to support the Global Threat Reduction Initiative (GTRI) by facilitating the return of fresh or irradiated HEU (including the possibility of contracting the manufacture of transport casks) or LEU spent fuel to the country of origin.

The Agency offers its management and technical experience in terms of technical advice, training, contract drafting and negotiations, Safeguards inspections, and application of safety standards to ensure that efficient and secure preparatory steps are taken in the country and the transfer is managed safely and securely. In addition, the Agency provides expert advice, organize training or technical workshops in areas relevant to safety, security or transportation, where needed.

For future spent fuel shipments under this programme, the IAEA would also help with pre-transport activities such as environmental impact assessment, contracting the supply of transport casks, assessment of transport routes, and by providing advice in respect of handling deteriorated research reactor fuel.

All mechanisms of IAEA TC are applied to GTRI related TC projects: (i) training (fellowships and scientific visits); (ii) expert missions; (iii) organization of technical meetings or workshops; and (iv) procurement of equipment, fuel and services.

When dealing with relevant purchases (like LEU fuel for conversion or fuel repatriation services), the IAEA offers mechanisms for international bidding, thus ensuring transparency and fairness and the best value for the money. The evaluation of tenders is carried out by international, independent and neutral experts and the contract is finally awarded to the best technical and financial offer.

Application for TC projects is carried out by interested Member States, by sending their project proposals to the TC Department of the IAEA.

3. Relevant IAEA activities in support of U.S. Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Acceptance Programme.

In 1986, to further encourage foreign research reactor operators to convert to LEU fuel, the U.S. Department of Energy DOE "Off-Site Fuels Policy" was extended to include the acceptance of foreign spent nuclear fuel containing uranium enriched in the United States. The U.S. accepted foreign research reactor spent nuclear fuel until the program expired (in 1988 for HEU fuels and 1992 for LEU fuels). A number of urgent “relief” shipments of spent fuel of U.S. origin did continue to take place, however.

During the period following the expiration of the U.S. Off Site Fuels program (which coincided with the creation of the research reactor fuels program in the IAEA Department of Nuclear Energy), the IAEA was involved as an observer in many of the meetings of the "ad hoc" group of research reactor operators, known as the Edlow/Egan Group. Beginning in January 1992 this Group kept up pressure on the U.S. DOE to accept US-origin spent fuel from foreign research reactors

Toward the same end, the Director General of the IAEA, Hans Blix, wrote letters to Secretary O'Leary of the US DOE (1 July 1993) and Victor Michailov, Minister of Atomic Energy of the Russian Federation, (2 February 1995) suggesting that these major partners in RERTR could facilitate the non-proliferation goal of RERTR by taking back foreign research reactor fuel.

A Record of Decision was published by DOE on May 13, 1996 to re-start the U.S. Foreign Research Reactor Spent Nuclear Fuel (FRR SNF) Acceptance Program with a deadline of May 13, 2006 for eligible fuel to be discharged from reactors and a deadline of May 13, 2009 for fuel to be received in the U.S.

With the re-initiation of the U.S. take-back program, the IAEA began a number of activities to assist member states eligible to ship spent research reactor fuels back to the U.S. The IAEA convened experts to develop guidance for Member States in this regard, which produced a Guidelines Document on Preparatory Work Prior to Return of Spent Fuel of US-Origin from Foreign Research Reactors", Draft IAEA-TECDOC (June 1996). (Note: These documents and lectures from the training courses, below, are available on the ANL/RERTR website at

http://www.td.anl.gov/Programs/RERTR/RERTR.html).

In response to a request from the US Government the IAEA organized two interregional training courses on the "Technical and Administrative Preparations Required for Shipment of Research Reactor Spent Fuel to its Country of Origin", in cooperation with the Government of the United States through Argonne National Laboratory. The first course was held at Argonne in January 1997 and the second in May 1999, also at Argonne. These courses included participants from Russian Federation research reactors.

The purpose of the courses was to provide participants with the technical, organizational and administrative information needed to prepare irradiated research reactor fuel for shipment to its country of origin, in this case, the United States [4].

The Global Threat Reduction Initiative (GTRI) was announced by U.S. Secretary of Energy Spencer Abraham at a speech at the IAEA on May 26, 2004.

The stated goal of the program is to substantially expand existing national and international efforts in order to secure and remove high-risk nuclear and radiological materials that continue to pose a threat to the United States and the international community. GTRI is to be carried out in cooperation with the IAEA and other international partners, building upon existing efforts such as the RERTR programme, and the U.S. and Russian research reactor spent fuel return programs. IAEA Director General ElBaradei has expressed his support and the Agency's willingness to work together to achieve the goals of the GTRI. There have been several discussions between IAEA and U.S. officials to clarify cooperative activities, and a GTRI Partners Conference was held in Vienna on September 18-19, 2004, which adopted conference findings supportive of the goal of accelerating and expanding relevant programs such as RERTR and the spent fuel take back programs.

A Technical Meeting on "National Experiences on Return of Research Reactor Spent Fuel to the Country of Origin" was held in Vienna from 28 to 31 August 2006. The TM was attended by 46 experts from 27 Member States.. The Technical Meeting allowed operators and managers of RRs that have successfully shipped RRSNF back to the country of origin describe their experiences, exchange information and transfer lessons learned to managers and operators of RRs that have not make any shipment yet but are considering the return of their RRSNF in the future.

4. Relevant IAEA activities in support of the "Russian Research Reactor Fuel Return Program" (RRRFR).

At the IAEA General Conference in September 1999, U.S. Energy Secretary Bill Richardson announced that the U.S. was prepared to work with Russia and the IAEA to manage and dispose of Russian-origin HEU research reactor fuel remaining in a number of countries.

On 14-15 December 1999 the IAEA convened the first Ad Hoc Tripartite Meeting on the possible management and disposition of Russian origin fuel currently at foreign research reactors. The meeting reviewed the situation regarding fresh and spent Russian origin research reactor fuel in various locations around the world, Russian experience in regard to spent fuel transport, legal, policy and safeguards issues, criteria for prioritising sites; scenarios for a demonstration shipment and action plan, as well as financial issues.

The Second Tripartite Meeting was held 27–29 March 2000 in Vienna, which included a presentation of the data and information collected by the IAEA, discussions of the IAEA role in the program as well as applicable Russian laws, regulations, and policies. It was decided that the IAEA should send a letter to targeted member states to assess their interesting in participating in a fuel return program. It was also decided that the site for a first demonstration shipment would be decided based on the responses to the letter, and the U.S. would provide funding for the shipment.

IAEA Director General Mohamed ElBaradei sent a letter on 29 September 2000 to sixteen countries with inventories of Russian research reactor fuel (Belarus, Bulgaria, China, Czech Republic, Egypt, Germany, Hungary, Kazakhstan, Latvia, Libya, Poland, Romania, Ukraine, Uzbekistan, Vietnam, and Yugoslavia). There were thirteen responses, all positive (one with reservations) and three did not reply (one of these, Libya, later shipped fresh fuel to Russia in 2004, see below).

The third and fourth Tripartite Meeting were held in April and September 2001, which requested and reviewed fact-finding missions to Ukraine, Uzbekistan, and Yugoslavia to begin detailed planning for eventual spent fuel shipments. Additional Tripartite meetings were held in November 2001, July 2002, and January 2003, the last of which included a report on the Vinca fresh fuel shipment which had taken place the previous August (though not a Tripartite shipment), progress on a possible fuel shipment from Uzbekistan, as well as for additional fact-finding missions to Latvia, Czech Republic, Romania, and Kazakhstan.

The first shipment of the Tripartite Initiative took place on September 21, 2003 (see http://www.iaea.org/NewsCenter/News/2003/weapons20030922.html), fresh HEU fuel was returned from the Magurele research reactor in Romania to Russia (14 kg uranium total, 10 kg U-235). The U.S. provided the funding for the shipment, which was carried out by the IAEA, under IAEA TC project RER/9/058. In association with the shipment, the U.S. provide approximately $4 million to IAEA Technical Cooperation project ROM/4/024 for the full-core conversion of the Triga research reactor at Pitesti, and committed to pay for the eventual repatriation of the Russian-origin spent fuel at Magurele (the U.S. origin spent fuel at the Triga reactor in Pitesti is eligible for repatriation under the U.S. return program.).

On November 7, 2003, U.S. Energy Secretary Abraham and Russian Minister of Atomic Energy Rumyantsev issued a joint statement concerning the Russian return program, stating that a government-to-government agreement to provide the legal framework for the implementation of the Tripartite Initiative would be ready for signature. The joint statement also committed to the development by the end of 2003 of a schedule of shipments of fuel.

The Eighth Tripartite Meeting was held 3–4 December 2003 in Vienna. The fresh fuel shipment from Romania in September was noted, as was an upcoming shipment from Bulgaria. The U.S. announced that it had already contacted Ukraine, Kazakhstan, and Vietnam regarding future shipments including incentive packages. Agreement was reached on a suggested schedule of both fresh and spent fuel shipments. Discussions also took place on the subject of a programmatic ecological expertise for the spent fuel shipments, as well as the potential spent fuel shipment from Vinca. A feasibility study using large scale transport and/or other casks was requested.

The second tripartite shipment took place in mid-December 2003 (see http://www.iaea.org/NewsCenter/News/2003/bulgaria20031224.html), with approximately 17 kg of 36% HEU removed from the IRT research reactor in Sofia. Once again, the fuel removal was funded by the U.S. under TC Project RER/9/058, and the U.S. committed to assist Bulgaria with an LEU fuel core for a planned research reactor as well as to eventually ship the existing spent research reactor fuel to Russia.

The third Tripartite shipment took place in early March 2004 (see http://www.iaea.org/NewsCenter/News/2004/libya_uranium0803.html) from the Tajoura Nuclear Research Centre near Tripoli. Libya, consisting of 80% HEU in the form of fresh fuel, in fuel assemblies containing about 13 kg of fissile U-235 as well as about 3 kg of uranium. The $700,000 fuel removal project was funded by the U.S. under TC Project RER/9/058, following Libya's announcement in December 2003 that it was ceasing all activities related to development of nuclear, chemical, and biological weapons. This fuel removal was accompanied by a U.S. commitment to fund, under an IAEA TC project, the full core conversion for the Tajoura reactor, as well as a U.S. pledge to pay for the eventual return of the Russian-origin spent fuel at Tajoura.

The fourth tripartite shipment took place on September 2004 (see http://www.iaea.org/NewsCenter/News/2004/uzbekistan.html), with about 10 kg of fresh reactor fuel transported by truck and air from the Institute of Nuclear Physics of the Academy of Sciences of Uzbekistan, near the country's capital, Tashkent to the Russian Federation. Once again, the fuel removal was funded by the U.S. under TC Project RER/9/058. and the U.S. committed to assist Uzbekistan to eventually ship the existing spent research reactor fuel to Russia.

The U.S. and Russia signed the bilateral agreement concerning the repatriation of Russian-origin HEU research reactor fuel to Russia on May 27, 2004, under which more than a dozen countries are eligible to receive financial and technical assistance from the U.S. under the Tripartite Initiative.

Since end of 2004 to present, the following main activities have been performed under this Agreement:

(a) On 22 December 2004, the IAEA helped Czech authorities and the US National Nuclear Security Administration (NNSA) remove HEU from the Czech Republic. Six kilograms of fresh HEU were safely returned to the Russian Federation. The nuclear fuel was originally supplied to the Czech Republic by the Soviet Union for use in the Soviet-designed 10 megawatt LVR-15 multi-purpose research reactor, located in Rez near the Czech capital, Prague. The IAEA carried out all the procurement and contract activities and safeguards inspectors were present in Rez to monitor the process of loading the fuel.

(b) On 25 May 2005, about three kilograms of fresh HEU were safely airlifted from Latvia back to Russia. The mission was a joint effort between Latvia, the Russian Federation, the United States, and the IAEA. The IAEA facilitated the contracts for the shipment to take place and IAEA safeguards inspectors were present to monitor the mission.

(c) By request of the RRRFR Programme, the IAEA initiated on July 25, 2005 a bidding process for the "Supply of a Research Reactor Spent Fuel Transport and Storage Cask System to be used for the Repatriation of RR Spent Nuclear Fuel to the Russian Federation". The Agency prepared the Scope of Work for this procurement, in consultation with the RRRFR Programme. An evaluation group consisting of five (5) international expert consultants was convened October 24-28, 2005 at IAEA Headquarters, Vienna, for the purpose of review and technical evaluation of the bids submitted in response to this IAEA procurement. A 4 million Euro contract was finalized before end of 2005 and a purchased order timely issued. The casks, fabricated by Skoda (Czech Republic) and inspected by IAEA experts, were delivered to the RRRFR programme.

(d) On 26-27 September 2005, fourteen kilograms of fresh HEU have been safely returned to the Russian Federation from the Czech Republic. The mission was a joint effort between the IAEA, the United States, the Czech Republic and Russia. IAEA safeguards inspectors monitored and verified the packing of the HEU for from a research reactor at the Czech Technical University, Prague. The shipment contract was arranged by the IAEA, as part of its technical cooperation activities. The nuclear fuel was originally supplied to the Czech Republic by the former Soviet Union for use in a Russian designed multi-purpose research reactor operated at the Czech Technical University for education and training of physics and engineering students.

(e) The IAEA has been provided funds by the Nuclear Threat Initiative (NTI) to proceed with the planning and implementation of the safe removal of the spent research reactor fuel from the Vinča Institute in Serbia. In addition, supplementary funds have been committed from the US-DOE. With such financial support, several studies of the spent fuel stored at Vinča have been carried out, as well as other activities to improve the spent fuel storage conditions. In 2005 the IAEA solicited bids from qualified firms to remove the spent nuclear fuel from the Vinča Institute for reprocessing at the PO Mayak facility in the Russian Federation. A group consisting of eight international expert Technical Evaluators, invited observers, and International Atomic Energy Agency (IAEA) observers was convened December 5–9, 2005 at IAEA Headquarters, Vienna, for the purpose of review and technical evaluation of the bids submitted in response to IAEA procurement and a preferred tender was selected. In October 2006, through its technical cooperation programme, the IAEA concluded a US $4.3 million contract with a Russian consortium and Serbia to start the work, which initially involves repackaging about 8000 TVR-S fuel elements for transportation. Another contract of nearly $5.5 million is being negotiated to cover transport and related tasks. The contract is one of the biggest involving the IAEA technical cooperation programme.

(f) In January-April 2006 the International Atomic Energy Agency assisted NNSA to complete the first HEU spent fuel shipment to the Russian Federation. During four shipments 252 fuel assemblies (63 kg of HEU) were removed from the Institute of Nuclear Physics in Tashkent, Uzbekistan. Two Russian TUK-5 railroad cars were used for the transport, 16 Russian small capacity TK-19 casks were used in Uzbekistan

(g) On 25 July 2006, the IAEA helped Libyan authorities and the US National Nuclear Security Administration (NNSA) remove fresh HEU from the Libyan Arab Jamahiriya. The nuclear fuel was originally supplied to the Libyan Arab Jamahiriya by the Soviet Union for use in the Soviet-designed facilities in the Tajoura Nuclear Research Centre. The IAEA carried out all the procurement and contract activities and safeguards inspectors were present in Tajoura to monitor the process of loading the fuel. The HEU fuel shipped consisted of 17 IRT-2M fuel elements of 80% enrichment containing 2.72 kg of U-235. The fuel removal was funded by the US-DOE, through an IAEA technical cooperation project.

(h) On 9 August 2006, the IAEA helped Polish authorities to remove close to 40 kg of fresh HEU from a nuclear research reactor facility at Otwock-Swierk near the capital of Warsaw. The HEU was safely airlifted back to Russia, which had originally supplied it to fuel Poland's RRs. The two-day mission was a joint effort between the United States, Poland, Russia, and the IAEA. Both IAEA safeguards inspectors and technical experts from the US National Nuclear Security Administration (NNSA) were present to monitor loading the fuel into canisters. The fuel removal was funded by the US-DOE, through an IAEA technical cooperation project.

(i) In December 2006 the removal and shipment of 268 kg of fresh highly enriched uranium from the Rossendorf facility took place. This is the largest shipment of Soviet-origin HEU ever conducted under GTRI. The five-day operation was completed in close cooperation with Germany, the Russian Federation, the International Atomic Energy Agency and Euratom. The HEU was loaded into 18 Russian TK-S16 specialized transportation containers at Rossendorf site. The canisters were transported under heavy guard and then airlifted from Dresden Airport to a secure facility in Russia.

The IAEA is also supporting the RRRFR through the organization of meetings, workshops, training and publication of guidelines to facilitate operator's/institutions participation in future spent fuel shipments. A meting was held in Vienna in July 2006 with representatives of the Government of Ukraine and countries in Central and Eastern Europe which seek to ship spent research reactor fuel in order to discuss transit requirements and arrangements for such shipments. The meeting participants succeeded in:

- Identifying the domestic regulatory, international and IAEA requirements, and other information needs, as well as other issues involved in transporting irradiated RR fuel across national boundaries to the Russian Federation (Mayak reprocessing plant, Ozersk region), especially through the territory of Ukraine;
- Exploring the possibility of standardized transit arrangements/agreements for the transportation of irradiated research reactor fuel through the territory of Ukraine to the Russian Federation; and

- Agreeing on next steps to be taken to simplify transit approvals in order to expedite future shipments of irradiated research reactor fuel to the Russian Federation.

A workshop was held in Belgrade, Serbia October 4-7 2006, on the basis of preparatory meetings held in Vienna in February and July 2006, to provide information to Member States in order to assist in their preparations for shipping irradiated Russian origin RR fuel to the Russian Federation, including the sharing of lessons learned from the planning, preparation, and conduct of shipments in early 2006 of irradiated RR fuel from the Institute of Nuclear Physics in Tashkent, Uzbekistan.

Since this initiative was launched in 1999, this workshop was the first occasion on which representatives of all countries shipping Russian-origin spent nuclear fuel to the Russian Federation had an opportunity to meet to share experience and review preparations.

The workshop participants succeeded in:

- Providing essential information related to technical and administrative preparations for shipping irradiated Russian research reactor fuel; including sharing experience gained and lessons learned from the successful Uzbek spent fuel shipments;

- Determining actions to be taken by IAEA and others to facilitate future shipments.

- Compiling a complete Guideline "Technical and Administrative Preparations for Shipment of Russian-Origin Research Reactor Spent Fuel to the Russian Federation", which provides key information for the planning and return of Russian origin spent nuclear fuel or materials containing highly enriched uranium (HEU) to the Russian Federation. It is intended for use by all parties involved in the planning, preparations, coordination and operations associated with returning SNF to the Russian Federation and it is available from the IAEA upon request.

The IAEA has considered organizing training courses in Russian for countries with Russian-origin spent research reactor fuel in order to assist countries to prepare for spent fuel shipments.

5. Conclusions

The IAEA continues contributing to international non-proliferation efforts in connection with HEU minimization by supporting RERTR and the programmes of return of RR fuel to the country of origin. Important progress has been achieved until today. These efforts will be maintained, strengthened and expanded in the coming years.

To assist Member States and the international initiatives in their efforts to reduce and eventually eliminate the use of HEU, the Agency offers all mechanisms available through its Regular Agency Programme and Technical Cooperation Programme.

Agency's involvement is not limited to supporting programmes of return of RR fuel to the country of origin.. Due consideration is also being given to support HEU minimization activities and RR sustainability in the post conversion and post return of spent fuel phases.

The International Atomic Energy Agency stands ready to assist Member States in these issues upon request.

REFERENCES

[1] GOLDMAN, I.N., ADELFANG, P., RITCHIE, I.G., "IAEA Activities Related to Research Reactor Fuel Conversion and Spent Fuel Return Programmes". Proceedings of the XXVI International Meeting on Reduced Enrichment for Research and Test Reactors, Vienna, Austria, November 7–12, 2004.

[2] ADELFANG, P., GOLDMAN, I.N., "Latest IAEA Activities Related To Research Reactor Conversion and Fuel Return Programmes". Proceedings of the XXVII International Meeting on Reduced Enrichment for Research and Test Reactors, Boston, Massachusetts, USA, November 6–11, 2005.

[3] ADELFANG, P., GOLDMAN, I.N., A. Soares, E. Bradley "Status and Progress of IAEA Activities on Research Reactor Conversion and Spent Fuel Return Programmes in the years 2005-2006". Proceedings of the XXVIII International Meeting on Reduced Enrichment for Research and Test Reactors, Cape Town, S. Africa, October 29 – Nov. 2, 2006.

[4] www.RERTR.ANL.GOV/COURSE.HTML

Argentine experience in shipping irradiated fuel to the United States of America

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Abstract. The National Commission of Atomic Energy (CNEA) owns the research reactors in Argentina. In the late ´90s, Argentina entered negotiations to adhere to the Foreign Research Reactors Spent Nuclear Fuel Receipt Program ran by the US-DOE. As a result, two shipments of MTR-type irradiated fuel manufactured by CNEA with HEU of American origin were successfully completed. The December 2000 shipment involved 207 spent nuclear fuel assemblies burnt in the RA-3 reactor, located at Ezeiza Atomic Centre (CAE). It was dispatched from the Central Storage Facility, a wet interim storage facility also located at CAE but away from the reactor building, where the fuel had been stored for 13 to 23 years. The July 2006 shipment involved 436 fuel plates that had been in the core of the RA-2 critical facility, and due to their very low activity were stored in a dry storage facility at Constituyentes Atomic Centre. Both campaigns had an initial stage of fuel characterization based on visual inspection to determine if the fuel was acceptable for transport and for storage in the receiving facility, or if canning was required. In the case of the RA-3 fuel, the inspection was performed through the analysis of remotely recorded video images of in-air fuel assemblies. The subsequent stage was the conditioning of the fuel for transportation, basically by cropping or disassembling the structural parts and re-identifying the assemblies/plates when the original ID number was removed with the separated parts. Finally, the conditioned fuel was loaded in the transport baskets that were placed in the transport casks. Intermediate transfer systems were used to load the RA-3 fuel. Specific topics as national legislation, institutional and legal aspects, safeguards, transport operations, etc., associated to the shipment, as well as the progress made in the RA-6 reactor fuel return scheduled for 2007, are outlined in the present work.

1. Introduction

The National Commission of Atomic Energy (CNEA) is the governmental institution that advises the executive power on nuclear policy in Argentina and, among many other incumbencies in the nuclear activity of the country, it is responsible for the spent fuel and radioactive waste management. CNEA is the owner and operates all the research reactors in the country; at present there are six.

In the early 1960s, USA began supplying Highly Enriched Uranium (HEU) of about 90% in isotope U-235 to Argentina, in the frame of the Program “Atoms for Peace”. Thus, Argentine reactors locally developed like RA-2 and RA-3 could start up with fuel manufactured in CNEA with the mentioned uranium of American origin.

Argentina adhered to the RERTR Program since it started in 1978, and a result was the conversion of the RA-3 reactor. RA-3 resumed operations in 1990 with a Low Enriched Uranium (LEU) core that allowed an increase of its nominal power to 5 MW

In the late 1990s, Argentina entered negotiations to adhere to the Foreign Research Reactor / Domestic Research Reactor (FRR/DRR) Spent Nuclear Fuel (SNF) Acceptance Program of the US Department of Energy (DOE) in order to get contractual agreements for the take-back of the HEU spent fuel irradiated in the RA-3, and some years later, agreements were extended to ship the HEU material from RA-2 and RA-6 reactors. As a result, two shipments of MTR-type irradiated fuel with uranium of American origin were successfully completed, and there is an ongoing plan to convert the RA-6, the only reactor that is currently working with a HEU core, and to ship back this core immediately after discharged.

The RA-3 experience consisted of the shipment of 207 MTR assemblies with a U-235 remnant quantity of 24.68 kg. It took about two years of technical preparations, including a full visual inspection of the fuel inventory to determine if it was eligible for shipment and storage in the receiving basin. The fuel was dispatched from the Central Storage Facility at Ezeiza Atomic Centre (CAE) on 13 December 2000 to the L-Basin in Savannah River Site (SRS).

The RA-2 experience involved 436 MTR fuel plates with negligible burnup that represented a mass of 3.34 kg de U-235. The material, which was stored in a dry storage facility at Constituyentes Atomic Centre (CAC), left the site on 14 July 2006 and was transported by air to the USA. Four days later, the material arrived at the receiving storage facility in the Y-12 complex, in Oak Ridge.

Since Argentina has been classified by the World Bank as an “other-than-high-income-economy country” (OTHIEC), all the shipment activities of both campaigns were funded by the DOE.

2. Fuel description

Since the time of the reactors’ start-up, the fuel has been manufactured in Argentina, primarily in CNEA, and then in the associated local companies. In the early 1960s the fabrication of MTR fuel with American origin HEU of about 90% in U-235 started at CAC. This fuel was intended for the RA-2 and RA-3 reactors that were under construction at that time. About 20 years later, when the RA-6 started up, fuel of the same design was used.

Figure 1 is a view of the Argentine HEU MTR standard assembly (SA) and Fig. 2 shows the arrangement of the exterior and interior fuel plates in the assembly. Main features of the fuel are: (i) U-Al alloy meat covered by Al-1100 cladding; (ii) The U-235 content was initially 7.8 g per fuel plate. In the early 1980s, that mass was increased to 10.5 g; (iii) Curved fuel plates; (iv) Nominal dimensions of fuel plate (cm): 75.5 (exterior) / 65.5 (interior) x 7.1 x 0.13-0.14; (v) Nominal dimensions of fuel meat (cm): 61.5 x 6.0 x 0.052-0.056; vi) 19 fuel plates in standard assemblies and 15 plates in control assemblies. Control assembly (CA) has fewer plates because there are two canals where the neutron absorber plates slide along; (v) Most of the structural parts of the assembly, as the flow nozzle, side walls, reinforcing lugs, lifting bail and spacer, were made of Al-1100. Instead, the screws, the internal guides for the absorber plates and the rivets in the CA were made of stainless steel 304; (vi) Fuel plates were fixed to the side walls by swaging. Table 1 provides complementary data of fuel assemblies.

TABLE 1. OVERALL WEIGHTS AND DIMENSIONS OF HEU FUEL ASSEMBLIES

	SA (7.8 g)	**SA (10.5 g)**	**CA (7.8 g)**	**CA (10.5 g)**
Number of plates	19 (2 ext. + 17 int.)		15 (2 ext. + 13 int.)	
Overall dimensions (cm)	88.0 x 8.4 x 7.62		161.1 x 8.4 x 7.6	
Overall weight (g)	4 974.4	5 252.7	5 998.7	6 219.0
Total weight of U (g)	164.7	222.3	130.0	175.5
Total weight of U^{235} (g)	148.2	200.1	117.0	158.0

FIG. 1. HEU MTR fuel assembly.

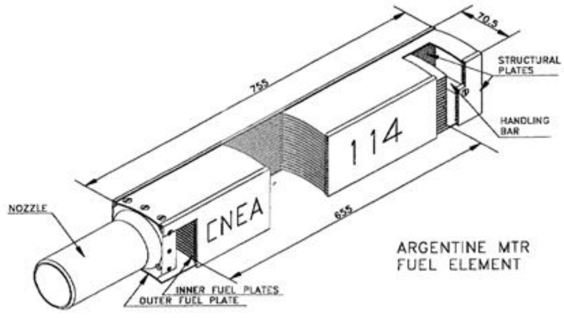

FIG. 2. View of fuel plates and structural parts.

3. RA-3 Experience

Along the last quarter of 2000, activities associated to the shipment to SRS of 207 MTR spent nuclear fuel (SNF) assemblies consisting of 166 SA and 41 CA took place in CAE. Basically, the activities performed for the shipment were the fuel conditioning operations inside of the storage facility, i.e. remote transference of the assemblies to the operation pool, fuel cropping, fuel re-identification, loading in transport baskets, etc., all them conducted by CNEA [1].

After a cooling period in the reactor decay pool, the spent fuel was transferred for interim storage to the Central Storage Facility for Irradiated Fissile Material (DCMFEI), located in the radioactive waste management area of the CAE, where the shipment campaign took place

The cask vendor chosen by DOE was NAC and the transport cask was the NAC-LWT. The contractor also supplied the MTR42 fuel basket. Each LWT holds six stacked NAC-MTR42 baskets containing a maximum of 42 assemblies. Five NAC-LWT casks (of about 5 m long and 21 TN) were necessary to transport the 207 assemblies. The loading of the filled baskets into the LWT was carried out by means of an intermediate transfer system (ITS) and a Dry Transfer System (DTS), both operated by NAC personnel with the support of CNEA personnel. The underwater cropping device to shorten the length of the assemblies was also supplied by the cask vendor.

3.1. RA-3 Reactor and history of HEU fuel core

The RA-3 is an open tank research reactor located in CAE, about 33 km from Buenos Aires city. The reactor is refrigerated and moderated using demineralized light water, and the reactor core is composed of 23 SA and 4 CA. Until 1987 the reactor worked with the fuel that was finally shipped to USA and the nominal power was 2.8 MW. In 1988, the core conversion to LEU was initiated and in 1990 the converted reactor returned to normal operation with a thermal power of 5 MW and 120 h/week of operation. In 2002, while maintaining the LEU core and the same operation regime, the reactor power was increased to 10 MW, which is the current authorized power for the reactor.

RA-3 reactor used HEU fuel since it went critical for the first time in Aug 1968 until Dec 1987. At the beginning, the reactor was fed with fuel with U-235 nominal content of 7.8 g per fuel plate. In the early 1980s, that mass was increased to 10.5 g. The overall burnup of the fuel inventory was 17.5% on average with a maximum value of 36.6% for the former, and 35.1% on average with a maximum value of 46.2% for the latter, which indicates that the reactor operation improved the fuel core management in the last years of the HEU cycle.

The RA-3, built in a joint program between CNEA and private industry in Argentina, is primarily used for production of radioisotopes but also provides facilities for research applications like material testing, activation analysis, neutron radiograph, etc. In the last years, the reactor has also been used for the irradiation of prototype fuel assemblies that subsequently are subjected to pos-irradiation analysis for CNEA qualification as fuel supplier.

3.2. Description of the DCMFEI storage facility

The DCMFEI is a wet storage facility for research reactor spent nuclear fuel located at the waste management area of CAE, about two kilometres from the RA-3 reactor. The building of the storage facility is 35.2 m long, 11.5 m wide and 4 m high. The structure is primarily concrete block walls with an iron plate roof and sliding bay doors at each end.

The facility contains two sectors of in-ground storage tubes for SNF (Fig. 3). The front sector has 6 rows with 16 storage tubes per row and the back sector has 6 rows with 17 tubes per row. Therefore, its total capacity is 198 storage tubes.

The storage rows, which are 0.5 m apart, have raised curbs at ground level with a lead-filled plug at the top of each storage tube. Each storage hole is lined with a 316 series stainless steel tube that is 2.1 m deep and 15 cm in diameter. Each tube can hold two MTR standard assemblies or one control assembly. There is about 15 cm of water shielding above two stacked SA or 30 cm of water shielding above a CA.

The storage tubes and rows are interconnected via a water recirculation system that has not been in operation for a long time. An inspection pit between the storage sectors contains valves and piping for the mentioned system.

The DCMFEI was not designed to accommodate commercial SNF shipping casks and did not have water–filled basin. CNEA modified a number of systems inside of the facility to support the planned conditioning of the SNF for loading into shipping casks (Fig. 4). The original 1 TN overhead crane used to handle the storage tube shield plugs was replaced by a new 2 TN overhead crane with a dual beam bridge, capable of handling a shielded transfer cask for SNF baskets, and the bridge rail supports were strengthened. Besides, a small auxiliary hoist attached to the bridge was used to handle individual fuel assemblies. Both the crane and the auxiliary hoist could be operated either remotely or via a tethered pendant controller. The valve pit was converted to a basin for underwater SNF conditioning operations by installing a stainless steel pool. The basin is 0.85 m wide, 5 m long and 2.5 m deep, holding about 10 cubic meters of water, with an associatied water filtering system. Two shielded stations were located inside of the storage facility. The three-sided walls were built of 10-centimeter thick lead bricks inside of a steel frame. The front walls were approximately 2 m wide by 2.75 m high with two lead-glass view-ports.

FIG. 3. View of DCMFEI facility.

FIG. 4. SNF conditioning in the facility.

As in a previous full inspection for SNF characterization, a monitoring station was set in the adjoining building about 10 m from the DCMFEI to follow the in-air transfer operations with the aid of cameras conveniently distributed inside of the facility. The LCD displays that received the camera signals and the pan-tilt-zoom control boxes of the cameras were installed in the station, which was sheltered by walls built with high-density concrete bricks.

3.3. Characterization of the spent nuclear fuel

In 1998, experiments based on the installation of aluminium coupon racks were launched to monitor the corrosion of the SNF, in the frame of a IAEA CRP on Corrosion of Research Reactor Aluminium-Clad Spent Fuel in Water. The results indicated that the conditions of storage in the DCMFEI generated aluminium corrosion. A preliminary visual inspection in June 1999 led DOE to decide to inspect all 207 HEU assemblies prior to shipping them back to the US. In October 1999, an inspection team from Westinghouse Savannah River Company (WSRC), with the cooperation of CNEA personnel, performed a detailed inspection of the whole SNF inventory [2]. The purpose of this full inspection was to characterize the fuel condition (structural damage, corrosion degree, etc.) in order to meet acceptance criteria for transportation compliance (i.e. containment) and provide assurance that the SNF could be safely handled and stored in SRS basins. The shape and small diameter of the storage tubes did not allow a procedure for underwater viewing by means of underwater video probes or underwater cameras. Therefore, an in-air fuel inspection remotely performed via video viewing and camera control system was an appropriate solution.

Three video units were positioned on the raised curbs approximately 120 degrees apart around the location of the selected tube. Each unit was equipped with a black and white video camera, a zoom lens, a pan & tilt, and components to control the systems from a remote location over a single coaxial cable. The coaxial cables were suspended between the DCMFEI and the adjoining building located approximately 10 m away, where the video units were controlled from a monitoring station. The LCD displays that received the camera signals and the pan-tilt-zoom control boxes of the cameras were installed in the station, which was sheltered by walls built with high-density concrete bricks (Fig. 5). After video cameras positioning, the following step was to remove the shielded plug from the selected storage tube with the aid of the bridge crane hook. Then, the assembly was manually grasped (Fig. 6) with special handling tools. Different grappling tools were available to engage standard assemblies and control assemblies. When removing a standard assembly from the bottom of the tube, a longer grappling tool with an underwater camera attached to the shaft was used. The camera was connected to a goggle-type video display worn by the tool operator. Once the assembly was grappled and the tool was hung on to the auxiliary hoist, the assembly was remotely raised for inspection and either lowered back into the tube afterwards or relocated to another storage tube after inspection to allow the lower assembly to be raised for viewing. The three video cameras were typically positioned to provide clear views of both outer fuel plates and at least one side plate on each assembly after removal from the storage tube. Operators zoomed and focused each camera on the plate in view and panned up and down the full length of the assembly while recording the inspection on digital videotape. The inspection record for each assembly consisted of 3 to 5 minutes of videotape from the three cameras. Special emphasis was placed on recording the assembly identification numbers and any abnormal characteristics such as damage or excessive corrosion. The cameras were manually repositioned after inspecting assemblies in two or three adjoining storage tubes.

Digital videotapes were extensively reviewed upon completion of the inspection effort. Assemblies were categorized by the extent of penetrating corrosion visible on the two outer fuel plates and/or structural damage, as shown in Table 2 [3].

TABLE 2. SNF DAMAGE CATEGORIZATION BY SEVERITY INDEX

Number of Fuel Assemblies	Severity Index	Exposed Fuel Area
67	0	No corrosion product nodules or general corrosion
24	1	< 0.1 cm^2
48	2	≥ 0.1 cm^2 to < 0.5 cm^2
33	3	≥ 0.5 cm^2 to 1.0 cm^2
16	4	≥ 1.0 cm^2 to 1.5 cm^2
19	5	≥ 1.5 cm^2

About 45% of the fuel assemblies exhibited minimal or no corrosion damage, and only 10% of the inventory showed significant corrosion indications. The reviews concluded that the SNF inventory condition was generally satisfactory for handling, shipment, and storage in the SRS basins.

FIG. 5. Monitoring station with shielding wall.

FIG. 6. Grasping SF from the bottom of a tube.

3.4. Spent fuel conditioning and loading activities

The fuel conditioning operations were performed by CNEA personnel inside of the storage facility. These activities were: remote transference of the assemblies to the operation pool, fuel cropping, fuel re-identification (when necessary), loading of the cropped assemblies in transport baskets and loading of the filled baskets into the ITS. The assemblies were processed in lots of seven (capacity of the NAC-MTR42 basket). CNEA had already established a list of assemblies to be loaded into each transport basket. The loading sequence was designed to: (i) Gather the assemblies classified with severity index 4 and 5 in only two transport casks. (ii) Ensure an elevated radiation rate in each sixth basket (the one closest to the cask lid) to comply with the physical protection requirement that the loaded LWT be Class II (self-protected; > 1 Sv/h measured on top with the cask lid open). (iii) Store a CA in the central slot of every basket, since the shorter length of those cropped assemblies helped to minimize the possibility of incorrect grasping of the DTS grapple during the basket transfer.

3.4.1. Remote transfer of fuel assemblies to the operation basin

Like in the full inspection, the first step was to remove the shielded plug from the selected storage tube with the aid of the bridge crane hook. Then, the assembly was manually grasped with the appropriate special tool. The same grappling tools were available to engage SA and CA. Once the assembly was grappled and the tool was hung on to the auxiliary hoist, the assembly was remotely lifted until it was entirely out of the tube and then transferred to the operation basin (Fig. 7), where it was lowered into the water.

3.4.2. Fuel cropping

The first underwater operation was fuel cropping. All the SNF inventory was cropped to reduce the length of the assemblies (to 0.69 m for CA and to 0.72 m for SA) and thereby reduce the required number of LWT casks. The nozzle was cropped from every fuel assembly. Besides, the top control plate guide box was cropped from every CA.

The cropping device consisted of an underwater band saw actuated by compressed air flux on a stainless steel plate that served as cropping table (Fig. 8). A pneumatic actuator drove the blade downwards along the cutting section at low constant speed producing straight cuts. The table had adjustable stops to control cropping length and three clamps pneumatically actuated to hold the assembly in position against a guide rail during cutting.

3.4.3. Re-identification of control assemblies

Since control assemblies lost their ID number when their top extremes were cut, it was necessary to re-identify them. Holding the cropped assembly upright, an aluminium flat bar was inserted between two fuel plates and slid along the assembly. One end of this bar was bent at 90º and the ID # engraved on it. The other end finished in a bending tab that protruded and was bent to fasten the ID bar to the assembly. The CA was loaded in the basket with ID # facing the top.

3.4.4. Fuel loading in the intermediate transfer system (ITS)

The empty NAC-MTR42 transport basket lay underwater inside of the inner shield of the ITS (ITS-IS) (Fig. 9). After the lot of seven assemblies was cropped, the personnel loaded each assembly into the basket with the help of the appropriate handling tool (since the assemblies lost their lifting bails, a parallel grip tool that clamped the side walls was used). The IS lifting lid was subsequently placed on the shielding with the overhead crane and the bolts secured. The IS was remotely raised from the basin and suspended over it for a few minutes to let water drain. The IS was then remotely transferred across the facility to the outer shield of the ITS (ITS-OS) that lay close to the facility rear door and lowered into it. The loaded ITS was removed from the storage facility with a forklift and placed outdoors about 15 meters from the LWT cask.

3.4.5. Loose plates canning

Three items consisted of loose fuel plates with very low irradiation. These plates were inserted in transport cans provided by NAC that fitted the size of the transport basket slots (Fig. 10).

3.4.6. Check-out and testing of loaded LWT

After the LWT was fully loaded, the cask lid was placed on top and bolted. The five loaded LWT casks were subjected to a radionuclide sampling test, consisting of the measurement of the Cs-137 activity increase in samples from the water that occupied the free volume of the loaded cask. The casks were filled with deionized water and water samples were taken initially, at 4 hours and at 12 hours after the commencement of the test. In three casks, the activity values detected in the 12-hour samples were less than 1% of the acceptable increase limit (1325 dpm/ml) and in the other two it reached 7% and 15% of that limit. After this test, a controlled water removal and vacuum were performed in each cask. The next step was to ensure confinement by filling the casks with helium and monitor its leakage through the cask seals. Then the casks were placed inside of their respective ISO containers. To verify proper transport conditions, superficial contamination (α and β/γ emitters) and the dose rate in contact and at 1 m were exhaustively measured on each cask. The ISO containers were also subjected to an equivalent radiological control.

FIG. 7. Remote view of in-air SNF transfer.

FIG. 8. SNF placed for cropping.

FIG. 9. *Basket inside IS and SNF under water.*

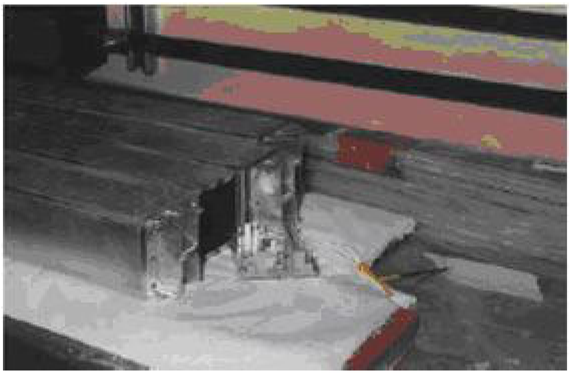

FIG. 10. *Transport cans for loose fuel plates.*

3.5. Radiological protection measures

From the radiological protection point of view, the most sensitive stage was the SNF in-air transfer from its storage position to the operation basin and, to a lesser degree, the transference of the loaded ITS-IS from the basin to the point where the ITS -OS was placed. In order to minimize the dose cost on the workers, these operations were carried out remotely.

After the selected assembly was engaged for removal from the storage tube, all the personnel evacuated the storage facility and stayed in the shielded monitoring station (adjoining building) during subsequent SNF removal and transfer, except for the crane operator and radiological protection officer. These workers sheltered behind the lead wall with shielded view-ports. The in-air movements were monitored via video displays from the monitoring station and two- way radios were used to communicate with the crane operator. Personnel left the station and the crane operator left the shield wall only after the assembly was safely in the operation basin. None of the staff involved (NAC personnel, CNEA personnel, mobile crane operators, safeguards inspectors, etc) worked in the outdoor loading area while in-air transfers took place. Light and sound signals alerted the staff not to enter the waste management area during in-air transfers. Contingency procedures had been foreseen for incidental cases (i.e., assembly disengaged in transfer, sudden crane detention). Seventeen people participated in the campaign and the typical daily staff was about thirteen people. The collective dose was approximately 10 mSv.man and represented 50% of the estimated dose.

3.6. Transport to the exit port

The transport convoy departed from CAE on 13 December 2000 at 3:30 AM to the selected harbour that was about 750 km away, close to Bahía Blanca city. The convoy was formed by 7 trucks each one transporting an ISO container, 5 for the LWT casks and two for the transfer systems and associated tools and hardware. Besides, there were four escort vehicles of security forces and two more vehicles with CNEA personnel responsible for the transport. On the same day, the ISO containers were loaded in an exclusive-use transport vessel that departed for Charleston at 19:02 PM.

3.7. Overall timeline for shipment campaign

By the end of 1998, the tracking down of the power history (in many cases lack of records was an issue) and design documentation of every fuel started, in order to get the necessary information to complete the DOE Appendix “A”. The preliminary visual inspection was in June 1999 and lasted approx. one week. In October 1999, the full inspection to characterize the fuel condition took place, during two and a half weeks of intensive work, working 12 hours per day and 6 days per week. The SNF conditioning and loading activities, which at that time corresponded to the largest shipment dispatched from one country in one stage, began in November 2000 and were carried out within a tight schedule in 23 days, with the same regime as in the full inspection, with an average rate of 1.3 loaded baskets per day. The whole loading campaign and land transport demanded five weeks of intense technical work and administrative preparations. The vessel departed for Charleston on 13 December

2000; made a stopover in Santiago to load a sixth LWT loaded with SNF from CCHEN, the Chilean nuclear institution, and arrived in SRS on 11 January 2001.

4. RA-2 Experience

In July 2006, the loading campaign to ship 436 MTR fuel plates that formed the RA-2 core took place in Constituyentes Atomic Centre (CAC), close to Buenos Aires city. For the transport, 18 stacks of fuel plates with a maximum of 25 plates each had been prepared. The material was sent to the Y-12 National Security Complex in Oak Ridge.

The facilities involved in the shipment campaign were: (i) the Storage Facility (DCMFE) where RA-2 fuel was stored and where the tasks for fuel characterization and preparation for transport took place, and, (ii) the Building # 37, the loading sector, with capabilities for the inlet, storage and handling of the transport containers.

Since the exported material had very low level of irradiation, the involved parts, i.e. consignor, receiving institution and regulatory bodies, concluded that the material could be classified as "non irradiated" for shipping purposes. Therefore, type B(U)F shipping containers for transport of non-irradiated uranium in form of bearing parts was used. DOE selected the French origin TN-BGC1 shipping container plus the TN-90 interior container. Each TN-90 accommodated two fuel stacks piled one on top of the other.

According to the contractual agreements with DOE, in August 2006 Y-12 sent to CNEA an amount of U-235 equivalent to the RA-2 fuel inventory, in the form of fresh metallic LEU, to support the local fabrication of fuel assemblies for the ongoing core conversion of the RA-6 reactor.

4.1. RA-2 reactor and history of fuel core

RA-2 was a tank type critical facility located in CAC. Its initial objectives were to provide training to the future operators of the RA-3 as well as to analyze potential configurations for that core. The RA-2 core was surrounded with graphite reflector and demineralized light water was used as moderator and refrigerant. Cooling was carried out by natural circulation of the water through the reactor core. The reactor started operation on 1 July 1966 and worked until 23 September 1983. RA-2 power was 0,1 W and it worked about 20 hours per week. The reactor was shut down after a criticality accident caused during a core reconfiguration sequence. The facility was dismantled and the fuel transferred to the dry storage facility DCMFE away from the reactor in the same site. Since the irradiation was at very low power, the burnup was reported as negligible.

4.2. Overview of the storage facility and fuel inventory

At the time of evaluation for shipment to the USA, the RA-2 fuel inventory was stored at the CAC storage facility (DCMFE) used to store remnant quantities of HEU in different special forms, as well as fabrication scrap from the time of HEU fuel manufacturing. The RA-2 inventory consisted of 14 standard assemblies, 5 control assemblies and 12 lots of loose fuel plates. Nominal U-235 quantity per fuel plate was 7.8 g and the total amount for the fuel inventory was 3.34 kg of U-235.

The radiological evaluation carried out on the lot of fuel assemblies and fuel plates irradiated in the RA-2 from 1966 to 1983 indicated very low activity, which facilitated their manipulation without radiological risk. The measurements taken on the whole of the material indicated that the maximum activity in contact were 63 µSv/h, measured on a SA. Regarding the surface contamination of the material, the values obtained of the lot of samples taken were not higher than the natural radiation background.

In the DCMFE, the tasks for fuel disassembling, fuel plate characterization and conditioning for transport took place. After this process, the material eligible for shipment consisted of 436 fuel plates separated in 18 stacks with a maximum of 25 plates each.

4.3. *Fuel preparation for shipment*

4.3.1. *Fuel disassembling*

The fuel was wrapped in plastic film and placed in racks in one of the storage rooms. They were taken to a suitable prepared worktable, where the plastic film was removed and the condition and identification of the fuel was checked. Then, they were taken to the disassembling machine to separate the plates from the structural parts. This machine consists of a table where the fuel is placed between two turning wheels. The two side walls of the assembly are attached by their ends to the aforementioned wheels which, when turning outwards, separate the fuel plates from the side walls (Figs 11 and 12). In some cases, the structural components were manually disassembled. The loose plates and the disassembled parts were wrapped in plastic film and returned to their original place in the rack.

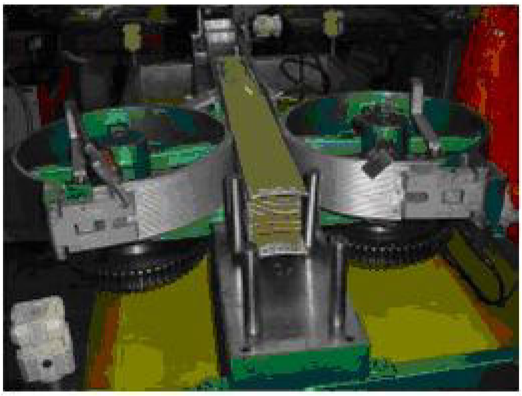

FIG. 11. Disassembling machine in operation.

FIG. 12. View of side walls after separated.

4.3.2. *Fuel plates characterization*

The identification number engraved on one end of each plate was recorded to make possible the identification of the uranium mass against the fabrication records. The dimensions and conditions of the plates were also checked, and conditioning was affected where needed. Since the external plates were longer than the internal ones, both ends of the former, which were non-fuelled parts, were cropped to attain the same length of the latter; as the original identification was lost, it was rewritten in the centre of the plate. A radiological characterization of each plate was then carried out, involving: the measuring of β-γ exposure rate in contact, the measuring of γ dose rate in contact and at 30 cm, and the calculation of β exposure rate in contact. Typical γ dose rate values for fuel plates were such that the average was 6.5 and the maximum was 11.5 (µSv/h in contact).

4.3.3. *Package conformation*

To get ready for transport, the fuel plates were piled in stacks of up to 25 units each. In order to minimize the dose rate in contact, the highest activity plates were placed in the centre of the stacks to be shielded by the external plates. The distribution of fuel plates to conform the stacks was carried out on a work table on which no more than 38 plates were placed at a time. Each of these stacks was held together by four metallic bands. Each stack was weighed before and after fastening, and both weights were recorded. A radiological assessment of each of the stacks was then carried out, involving: the measuring of β-γ exposure rate in contact, the measuring of γ dose rate in contact and at 30 cm, and the calculation of β exposure rate in contact. Gamma dose rate values for fuel stacks were such that the average was 58.3 and the maximum was 74 (µSv/h in contact).

Each stack was marked with the ID#, net and gross weights, uranium mass, etc. Once assembled and characterized, the stack was placed on the racks with a predetermined distribution, which ensured the subcriticality of the lot. Finally, the safeguards inspectors sealed the entrance to the storage room until the time of the loading campaign.

4.3.4. Loading of fuel packages in the transport containers

At the time of the shipping campaign the fuel stacks were transferred to the Building # 37, a maintenance shop facility adapted for the loading activities, one block away from the storage facility. Building # 37, where the empty TN-BGC1 were temporarily stored (Fig. 13), has a bay door that permits the entrance of forklifts and is equipped with a 1 TN manual hoist that facilitated the shipping container handling. Fuel stacks were transferred two at a time and immediately inserted into the TN-90, the inner container of the TN-BGC1 (Fig. 14). For safety and security reasons, each pair of fuel stacks was brought to the building only after the previous loaded container was closed and sealed. In order to transport the 436 plates (plus two fresh mini plates also fabricated with HEU), nine TN-BGC1 and their corresponding internal containers TN-90 were used. After fuel loading, a set of check-out measures were performed: i) leak test performed by Y-12 inspectors, ii) installation of the shock absorber, iii) sealing with tamper-indicating devices (TID) applied by safeguards inspectors and Y-12 inspectors, iv) radiological evaluation that consisted of dose rate measurements in contact and at 30 cm, neutron flux and surface contamination, to comply with transport and receiving facility requirements, and v) transport package labelling. Having completed the check-out stage, the TN-BGC1 plus the tool box and other devices were loaded on to the truck and conveniently secured to the trailer by means of belts. Security escorts distributed in two cars accompanied the convoy in the journey to Ezeiza airport, where the material was dispatched in an air cargo to the USA. Fuel transport was performed following the ARN regulation AR 10.16.1 “Transport of Radioactive Materials” that closely follows the IAEA “Rules for the Safe Transport of Radioactive Materials”, N° TS-R 1, 1996 Edition (Revised).

FIG. 13. View of TN-BGC1 in Building # 37.

FIG. 14. Fuel plates being inserted into TN-90.

4.4. Overall timeline for shipment campaign

During October and November 2004 several visits to the DCMFE took place, one of which included Y-12 personnel, to assess the status of the RA-2 inventory, make a preliminary list of the eligible material and proceed to a preliminary radiological characterization. These visits aimed to generate a draft version of the DOE Appendix “A”. During the following months, the main task was to design the procedure for fuel conditioning for transport and the loading campaign, taking into account the selected shipping container. This stage included the presentation of documentation to the regulatory body for the authorization of the practice and for the validation of the TN-BGC1 license in the country. At the end of July 2005, the tasks of fuel disassembling and fuel plate characterization (explained in sections 4.3.1. and 4.3.2.) were completed. Immediately afterwards, followed a stage of identification of uranium mass per plate based on the study of the fabrication records, most of them of the early 60´s. During the following September, the activities related to the conformation of the fuel plate stacks (section 4.3.3.) were completed. This stage included the presence of Y-12 and safeguard inspectors who checked the conformed material. After a period of several months, needed to obtain governmental clearances in both countries, the loading activity (as described in section 4.3.4.) began on 7 July 2006 and demanded 3 working days. On 14 July at 9:30 AM, the transport truck left CAC for Ezeiza airport, where it arrived two hours later. The air cargo with the shipping containers took off early in the morning of 17 July. Finally, after a land transport from the arriving airport, the RA-2 fuel entered Y-12 on 18 July at 7:30 AM.

5. RA-6 Status

RA-6 reactor is located in Bariloche Atomic Centre (CAB), in the city of San Carlos de Bariloche, in the south of Argentina. At present, RA-6 is the only Argentine reactor using HEU fuel. The fuel inventory consists of 42 MTR-type assemblies, all of them eligible for taking back to USA. They have been in the reactor since the start up in 1982.

In the frame of the RERTR Program, studies have been conducted to reduce the enrichment of the RA-6 reactor core. The engineering of the new LEU fuel for the reactor has been completed, and in 2005, CNEA and DOE signed a contract to consolidate the conversion of the RA-6 reactor to LEU and for shipping back all 42 spent fuel assemblies in a single shipment under the DOE FRR SNF Acceptance Program. As in the case of the RA-3, the receiving facility for the RA-6 HEU fuel inventory will be SRS. The shipment is planned to occur in 2007.

5.1. RA-6 reactor and its HEU fuel core

RA-6 is an open tank type reactor rated at 500 kW (nominal) that has been operating since 26 October 1982. It was constructed primarily to support nuclear research and education, and provide developments for medical treatment, i.e., BNCT. The reactor works about 40 hours per week on average.

The original core started with an arrangement of 25 fuel assemblies, but currently it has 27 standard assemblies plus 5 control assemblies, arranged into an 8X10 grid plate. The RA-6 fuel inventory is completed with 10 standard assemblies stored in the decay pool at the reactor building. Most of the fuel inventory was firstly irradiated at RA-3 when it worked at 2.8 MW. In 1982, 37 irradiated standard assemblies were transported from Ezeiza to Bariloche, at the time when the RA-6 reactor was going to start up, and the reactor core was completed with 5 fresh control assemblies.

As of July 2004, fuel burnup was 18.6% on average with a maximum value of 23.3% for SA, while burnup for CA was about 6.3%; the remaining mass of U-235 in the fuel was 5.19 kg. In March 2005, a team from SRS, which will be the receiving site, performed a cursory inspection of 10 of the 42 fuel assemblies with the logistical support of the reactor personnel. No significant issues were identified relating to the fuel condition.

5.2. Reactor building capabilities for fuel handling

The RA-6 reactor is an open stainless steel lined tank with overall dimensions of 10.4 m height and 2.4 m in diameter. The reactor core is kept about 6.6 m below the water level. The reactor building is approx. 20 m height and three levels can be identified: (i) the ground level (level 0) where the experimentation facilities surrounding the reactor are placed, as well as the access bay door, (ii) the reactor tank and the control room, at approx. 8 m above level 0 (Fig. 15) and, (iii) the underground level at approx. 5 m below level 0, where the SNF decay / storage pool (Fig. 16) and the water purification system are.

A shielded container of about 800 kg is used for transferring irradiated fuel between the reactor tank and the decay pool. The container is submerged in the water with the aid of the bridge crane and the selected fuel, handled with a grasping tool, is loaded into the recipient. The access from level 0 to the decay pool is by means of a hatch in the floor with dismountable lids that allows the transfer of the container to and from the reactor tank. Decay pool dimensions are 1.5 x 1.1 m and 4 m depth. There are two storage racks that are fixed to the pool bottom but may be easily dismounted.

A 5 TN double-beam bridge crane spans all the building at a height of approx. 6 m above the reactor tank. Attached to it, there is an auxiliary hoist of 1 TN with stainless steel hook used to move the container. The reactor building has an access bay door for large components at level 0; its approximate dimensions are 4 m wide and 3 m height. Out of the bay door there is a descending ramp that leads to the rear street.

FIG. 15. View of the reactor from control room.

FIG. 16. RA-6 decay pool.

5.3. Perspectives for the shipment campaign

DOE already announced that the NAC – LWT shipping cask has been selected to ship back the RA-6 fuel inventory. Only one LWT unit is necessary to place the 42 fuel assemblies, provided the shipping basket model MTR-42 is used, producing a great economy in cask vendor services. Therefore, cropping of the nozzle of every fuel assembly, as well as the upper structural part of the control assemblies, must be performed to accommodate the fuel in the mentioned shipping basket. As a consequence, a facility for the disposal of those cropped parts will be constructed into the reactor building, because the CAB does not have appropriate installations for temporary storage of medium-level radioactive waste. Except for the mentioned storage facility, RA-6 building has the basic capabilities to handle and prepare the material for shipment, as well as to enable the LWT loading by means of the cask vendor standard hardware: the ITS and the DTS.

The decay pool size has been identified as the critical path in the design of the fuel conditioning procedure. Basin dimensions indicate that there is very limited capacity to submerge the necessary hardware to process the irradiated fuel. It was concluded that an underwater cropping machine in which the fuel is vertically positioned would leave room for the other components. The cask vendor committed to supply an electrically powered machine like this. Most likely, the fuel racks and other components that are in the decay pool will be temporarily withdrawn to facilitate underwater work. In this case, the MTR-42 in the ITS inner shield would serve as a stand for both untreated and treated fuel. The third main component that requires room in the pool is the shielded container for fuel transfer from the reactor tank. Besides, the use of some other objects under water, such as a basket to store cropped parts and several handling tools, should be considered in the procedure design.

Once the MTR-42 is loaded, the ITS-IS should be raised from the basin with the aid of the 5 TN crane and lifted through the hatch to level 0, where the ITS-OS would be placed. In a single movement, the IS should be positioned on the OS and then nested inside it. The whole ITS should be moved and placed outdoors, close to the LWT cask, by means of a 6 TN forklift. The forklift should get in or out through the bay door at level 0, which opens on to the street behind the reactor building. Therefore, this street is an appropriate place to install the cask loading area, where the large components like the LWT cask, the DTS, the ISO containers, the mobile crane, etc., would be conveniently placed. It is estimated that the fuel conditioning and loading into the transport cask will last approximately 15 working days

Transport to the exit port is another important aspect to analyze. There is only one paved access to the CAB that borders the Nahuel Huapi lake about 10 km on the way. The features of the landscape and the shape of the road are not the best for the transport of heavy loads. These conditions get worse during the winter due to the snow; therefore, transport should be constrained between October and March. Out of the urban zone, potential truck routes are approximately 900 to 1 000 km to the potential ports and may require over-night stop to complete the journey. CNEA has initiated a deep study of the regulations regarding the transit of nuclear material in the provinces that the convoy may pass. The physical protection during transport from CAB to the port should be similar to the security provided during the RA-3 reactor shipment in 2000.

6. Legal, institutional and safeguards aspects

The National Commission of Atomic Energy is an autarchic organism created on 31 May 1950 that depends of the Ministry of Federal Planning and Public Investment. CNEA develops its wide scope of activities within an extensive legal frame. The two main laws in this frame are the Decree–Law N° 22.498/56, ratified by Law N° 14.467, and the Law N° 24.804 "National Law for the Nuclear Activity". Besides, CNEA is responsible for the application of the National Law Nº 25.018 "Management of Radioactive Waste" and of the "Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management", ratified by National Law Nº 25.279 in June 2001.

CNEA created two projects to carry out the SNF shipments and settled the local counterparts of the representatives from the DOE Acceptance Program, the receiving facilities and the cask vendor. The first project was the PRECI, for the restitution of the RA-3 HEU SNF inventory. The second is the ongoing UBERA-6 project, for the core conversion of the RA-6 reactor and shipment of the corresponding HEU fuel, which also involved the recently completed RA-2 shipment.

Article 41 of Argentina's Constitution prohibits the entry to the National territory of waste that is either actually or potentially hazardous, and the radioactive waste. The American and Argentine parties for the shipment contracts were advised on potential legal problems for joint shipments that imply the entry in Argentina of spent nuclear fuel (SNF) from another country, even temporarily. In this sense, CNEA identified that authorization just for the transit and mooring of an ocean-going vessel with SNF from another country would not be possible. Therefore, the alternative of joint shipment as in the aforementioned cases was disregarded by the parties.

Since the exportation of nuclear material such as SNF is considered sensitive by the Argentine Government, the National Commission for Control of Sensitive Exportations and Belligerent Material (created in 1992 by Decree N° 603) has to issue a previous license for exportation. For this, the 603 Commission, which includes representatives from the Ministries of Defence, Economy and Foreign Affairs, requires the material receptor to provide assurances that the material will not be used for non-peaceful purposes by delivering an End User Certificate. The issue of this Certificate by the US Government and subsequently the issue of the 603 License have been critical paths in the shipment timelines having produced significant delays in the shipment projects completion.

The Argentine regulatory body (ARN), has a major role in shipment campaigns. ARN revises the transport cask design and issues a local revalidation of the package license, subjected to the validity of the original one, which in the case of the US, is issued by the DOT. ARN also authorizes the practice of fuel conditioning, loading and transport, in order to ensure that the operator meets safety, security and safeguards requirements. Besides, ARN issues an Authorization of Exportation required by the Argentine Customs to permit the exportation of the nuclear material. Finally, ARN is the advisor organism for the 603 Commission in nuclear matters as well as the official liaison with safeguards organisms.

Safeguards in Argentina involve the IAEA and the ABACC. The last one is an organism that came up from a bilateral agreement between Argentina and Brazil for common accountancy and control of nuclear material. Subsequently, a quadripartite agreement was signed by Argentina, Brazil, the

ABACC and the IAEA for the application of international safeguards (INFCIRC 435). In this frame, when nuclear material leaves Argentina for another country it must be put under IAEA safeguards. On the other hand, DOE facilities that are receptors for the FRR SNF Acceptance Program are not in the list of facilities eligible for the application of safeguards by the IAEA. A facility capable of carrying out that role is the DOE storage facility for fresh nuclear material at Oak Ridge. Therefore, although the RA-2 and RA-3 fuel inventories were not physically placed there, DOE proposed that non-irradiated HEU of the same quantity substitutes for the HEU content of said inventories. The situation turns more complicated if that fresh uranium is withdrawn for fuel fabrication, specially when the transferred fresh material carries a third country obligation to Argentina.

REFERENCES

[1] GOLDSCHMIDT, A., NOVARA, O., LAFUENTE, J., "Preparations for the Shipment of RA-3 Reactor Irradiated Fuel" (Proc. 24th RERTR Meeting, Bariloche – Argentina, 2002). Available at: http://www.rertr.anl.gov/Web2002/2003Web/FullPapers-PDF/Goldschmidt.pdf

[2] NOVARA, O., LAFUENTE, J., LARGE, S., ANDES, T., MESSICK C., "Inspection experience with RA-3 spent nuclear fuel assemblies at CNEA´s Central Storage Facility" (Proc. 23th RERTR Meeting, Las Vegas, USA, 2000).
Available at: http://www.rertr.anl.gov/Web2000/PDF/Ande00.pdf

[3] ANDES, T., CASTLES, R., LARGE, S., ROBINSON, C., SINDELAR, R., "Inspection of RA-3 spent fuel assemblies at CNEA´s Central Storage Facility at Ezeiza", Report WSRC-TR-2000-0001, Jan 2000.

Spent fuel situation at the ASTRA Seibersdort and the TRIGA Vienna Research Reactors

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Abstract. In the past decades Austria operated three research reactors, the 10 MW ASTRA reactor at Seibersdorf, the 250 kW TRIGA reactor at the Atomic Institut Vienna and the 1 kW Argonaut reactor at the Technical University in Graz. Since the shut down on July 31st, 1999 and decommissioning of the ASTRA reactor and the shut down of the ARGONAUT reactor Graz on July 31, 2004 only the TRIGA reactor remains operational. The MTR fuel elements of the ASTRA reactor have been shipped in spring 2001 to Savannah River and the fuel plates from the ARGONAUT reactor Graz in December 2005 under the DOE fuel return programme.

1. Introduction

The TRIGA reactor Vienna is used intensively for education and training, all reactor systems are in excellent condition, spare fuel elements are available to operate this reactor for another 10 to 15 years and at present there is no indication whatsoever that this reactor should be closed down in the coming years.

This paper discusses the experience with the completed fuel shipment of the ASTRA reactor, while the fuel shipment from the ARGONAUT reactor is presented in a separate paper at this conference [1] further an outlook of possible options for the TRIGA reactor is given.

2. Historical survey

In the period between 1959 to 1965 three research reactors were built and operated until 1999. The first reactor was the MTR type ASTRA reactor at the Austrian Research Centre Seibersdorf (ARCS www.arcs.ac.at) which for a long period was the main research facility for nuclear research in Austria as well as the planning centre for a nuclear power plant to be installed at the site of Zwentendorf (730 MWe BWR), and which as it is well known was never put into operation due to a public negative referendum in 1978. This also effected the further development of nuclear research and in particular the programs at the ARCS. For several non-technical reasons the 10 MW ASTRA reactor was finally shut down on 30 July 1999 and immediately decommissioning started.

The second reactor was planned as a typical university training and education reactor in Vienna. A TRIGA Mark II reactor was selected, built and reached first criticallity on 7 March 1962. This reactor is well maintained and utilized and it is in operation without any specific deadline for shut down.

3. Present situation

3.1. The ASTRA Reactor in Seibersdorf

After 39 years (1960 to 1999) of successful operation, the 10 MW multipurpose MTR research reactor ASTRA at the Austrian Research Centers Seibersdorf (ARC) has now been decommissioned [2] and the final release from the nuclear legislation has been issued by 19 October 2006.

The 54 MTR-fuel elements (310.5 kg of HLW), described in Table 1 were shipped to US-DOE Savannah River Plant for ultimate disposal in May 2001.

TABLE 1. ASTRA FUEL

Fuel	Fuel-Element Identification	No. of Elements	Mass/Element [kg]	Total-Mass [kg]	Costs [US $]
HEU	AR-S-16 – AR-S-20	5	5.28	26.40	118 800
HEU	C-38 – C-42	5	4.52	22.60	101 700
HEU	AR-P-08 – AR-P-09	2	3.46	6.92	31 140
LEU	S-31 – S-33	3	6.38	19.14	71 775
LEU	AR-S-34 – AR-S-44	11	6.10	67.10	251 625
LEU	AR-S-45 – AR-S-60	16	6.22	99.52	373 200
LEU	G-497 – G-499, G-501 – G-503	6	6.29	37.74	141 525
LEU	AR-C-43 – AR-C-47	5	5.37	26.85	100 687
LEU	AR-P-10	1	4.21	4.21	15 787
Total		54		310.48	1 206 240

Before the shipment the following steps had to be performed:

- June 1997: First inspection of fuel element conditions by DOE and by the ASTRA managers
- November 1998: Visit by DOE, official statement by ARCS about permanent shut-down in 1999
- December 1998: ARCS formally applies to DOE about intentions to ship spent fuel
- May 1999: Contract raised by DOE received in ARCS
- November 1999: Return of the contract signed by ARCS, Austrian Federal Government, and Euratom
- November 1999 to February 2000: leak-proving of the fuel elements carried out in the reactor pool
- May 2000: completed papers were returned to DOE
- International tender, three suitable offers were received favouring Transnucléaire
- Transport to Rotterdam was carried out by Sommer and Grottke/Germany using two NAC-LWT-6 casks
- From Rotterdam to Savannah River transport responsibility was with NAC
- May 31st 2001: 54 spent MTR-fuel elements (310.5 kg of HLW) left Seibersdorf (six months later than scheduled)
- July 1st 2001: fuel received at US-DOE Savannah River Plant
- February 2003: Ten new elements still remaining were sold to GKSS-Forschungszentrum Geesthacht, Germany

In immediate succession and still under the operating license, all experimental facilities and components of the reactor within the vicinity of the core, or in intermediate storage within the building e.g. old beam-tube installations, 492 kg of ILW and 5 212 kg of LLW, were removed and treated. In the course of this procedure custom-designed, remote-controlled equipment was built and three GNS-Mosaik containers were filled, partly under water, with the remaining material. Also the task of clearing the reactor building from remaining experimental equipment, obsolete storage facilities and the transfer of the structures of the industrial source services including a 21-ton-lead-cell to NES Hot Cell Laboratories were accomplished to 90% by May 2003.

A preliminary evaluation of the expected amount of material to be decommissioned was performed which showed that it would amount to approximately 1 600 t of total material of which 10% with 160 t considered radioactive waste.

Now that the decommissioning procedure is finished the actual values are as follows:

- Total material involved: 1 800 t
- Material free for recycling: 1 600 t or 89.5%
- Material free for storage: 120 t or 6.5%
- Radioactive waste: 80 t or 4.5%

3.2. The TRIGA reactor at the Atomic Institute Vienna

The reactor operated since its first criticality with an average of 220 days per year without any long outages. The TRIGA-reactor is purely a research reactor of the pool type that is used for training, research and isotope production (Training, Research, Isotope Production, General Atomic = TRIGA) [3]. Throughout the world there are around 50 TRIGA-reactors in operation, Europe alone accounting for 8 of them. The reactor core consists presently of 80 fuel elements (3.75 cm in diameter and 72.24 cm in length), which are arranged in an annular lattice. Two fuel elements have thermocouples implemented in the fuel meat which allow measuring fuel temperature during reactor operation. At nominal power (250 kW) the centre fuel temperature is about 200 °C. Because of the low reactor power level, the burn-up of the fuel is very small and most of the fuel elements loaded into the core in 1962 are still there. A summary of the fuel situation is shown in Table 2.

TABLE 2. FUEL ELEMENT SITUATION AT THE TRIGA VIENNA AS OF 1. 1. 2006

Number of FE	Location	Cladding		Enrichment	Remarks
		Al	SST		
80 plus 2 in storage	core	57	25	70 FE 20% 9 FE 70%	2 instr. FE
13	fresh fuel storage	-	13	20%	2 instr. FE
8	spent fuel storage	8	-	20%	1 instr. FE
1	hot storage facility	1	-	20%	cut into 3 pieces
Total: 104		66	38		

The TRIGA reactor is heavily used for training and education of students in the nuclear field but also used for national and international training courses with the IAEA and with neighbour countries (Germany, Czech Republic, Slovak Republic, European Nuclear Engineering Network – ENEN). Many cooperation projects exist with the IAEA as the TRIGA reactor Vienna is the closest nuclear facility to the IAEA and the irradiation services has increased since the shut down of the ASTRA reactor although for many requested services the TRIGA cannot offer the requested power and neutron flux. At present there is no indication from the government that an imminent shut down of this facility is taken into consideration.

REFERENCES

[1] SCHACHINGER, E., NINAUS, W., “Repatriation of Spent Fuel of the Siemens Argonaut Graz Research Reactor to the USA”, on this publication.
[2] ARCS: www.arcs.ac.at
[3] TRIGA Wien: www.ati.ac.at

Repatriation of spent nuclear fuel from the Siemens-Argonaut-Graz research reactor to the United States of America

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Abstract. We report on the repatriation of spent nuclear fuel of the Siemens-Argonaut zero power research reactor Graz (WAGL). This reactor had two cores, one LEU core consisting of 234 aluminum coated fuel plates and one HEU core consisting of 125 aluminum coated fuel plates. The maximum burn-up was 10^{-4}% according to the FIFA definition (Fission per Initial Fissile Atom). This low burn-up resulted in simplified loading operations because of the very low radiation level. A detailed description of the reactor site will be given in its relevance to the loading operations. We will discuss further the planning phase, problems in selecting a feasible transport route, national regulations and we will also give details of the timeline.

1. The Siemens-Argonaut reactor Graz

1.1. General information

The Siemens-Argonaut Reactor Graz (WAGL) [1] at Reaktorinstitut Graz was operated from 1965 to 2004 by a small group of scientists who were organized in a private club first called “Verein zur Förderung der Anwendung der Kernernergie” and later “Verein zur Förderung der Strahlenforschung”. This reactor had a ring-cylindrical core containing the fuel elements which were covered by water and additional graphite blocks. These graphite blocks served together with the water as moderator. The inner container was filled with a graphite cylinder and the outer container was surrounded by a graphite wall. Both graphite parts acted as reflectors. Around the reactor core the main part of the biological shield was formed by concrete blocks as shown in Fig. 1 in a pictorial view of the WAGL.

The reactor operated as a zero power research reactor with the reactor power limited to a maximum of 100 W thermal. It served as a research reactor in the field of power reactor development and its variable core configuration allowed the simulation of particular neutron flux profiles expected to be observed in the core of various reactor types, in particular pebble-bed reactors [2],[3]. The restriction to zero power ensured very low radiation background and made possible high accuracy measurements. The reactor also served for educational purposes to teach the basics of reactor physics and as a source of thermal neutrons to students of the Graz University of Technology for other basic experiments.

Most important was a big experimental volume in the reactor’s centre and the attached thermal column which allowed activation experiments in a controlled environment of thermal neutrons. The mobile water tank provided one more experimental volume.

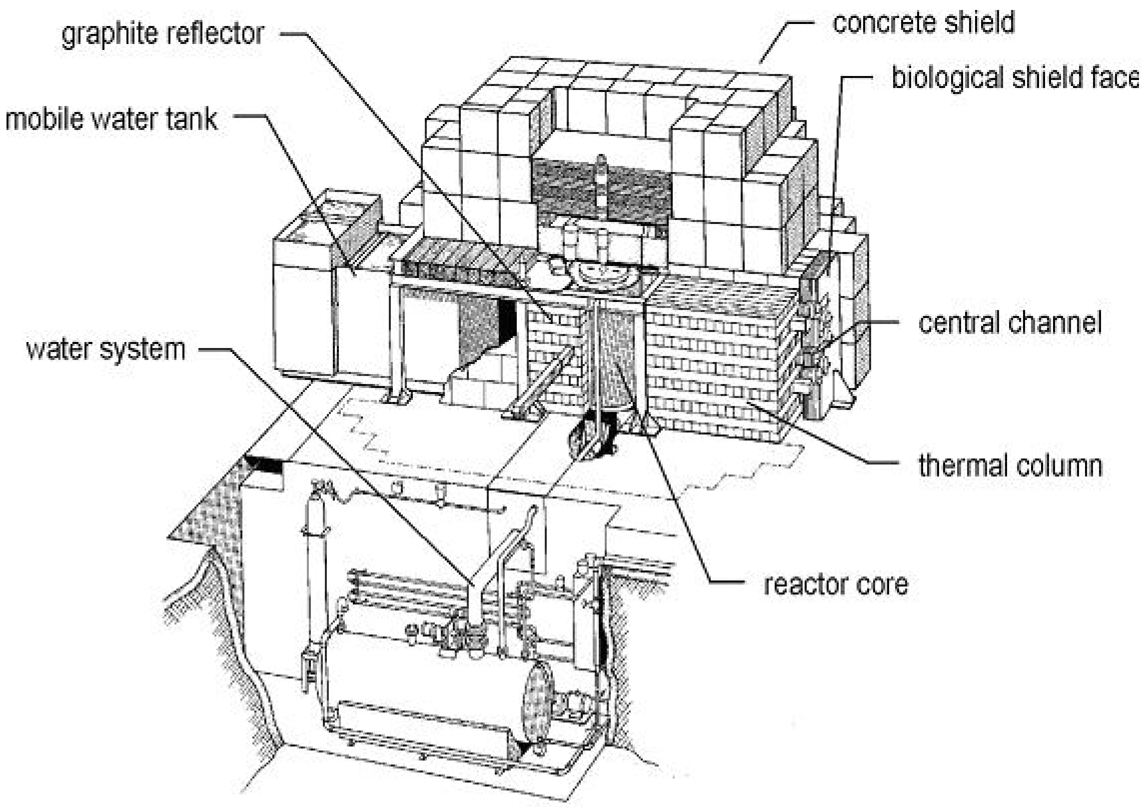

FIG. 1. Pictorial view of the Siemens-Argonaut Reactor Graz.

Finally, by the end of year 1999 the club's management decided to close down the reactor operation and to repatriate the LEU and HEU fuel plates. Operation was closed by the end of July 2004.

1.2. The fuel

The fuel elements (see Fig. 2a) were built from a set of fuel plates. The reactor had two sets of fuel plates, one set of 234 LEU plates and a second set of 125 HEU fuel plates. The fuel matrix of each LEU plate consisted of uranium oxide and contained about 105 g U_3O_8 with 20.8 g U-235 in form of U_3O_8-Al powder. The mean dimension of the fuel matrix was 600 x 70 x2 mm³ and its weight was 286.5 g. We had two types of HEU fuel plates: (i) the 'standard' plate had a fuel matrix of UAl alloy of dimension 600 x63 x0.58 mm² and a weight of 69.7 g. It contained about 22.3 g U with 20 g U-235; (ii) the 'end' plate had a fuel matrix of UAl alloy of dimension 600 x 63 x 0.74 mm² and a weight of 86.9 g with the same content of U and U-235 as the standard plate.

The fuel matrix of both types of fuel plates was aluminum coated. The outer dimensions of each fuel plate extended to 650 x75 x3 mm³. A maximum of 17 such fuel plates have been combined to build one fuel element. In case of a HEU fuel element 15 standard plates and two end plates have been used. A fuel element could also contain blind plates (non-fuel plates of pure aluminum) to allow for the most homogeneous distribution of fuel along the ring-core.

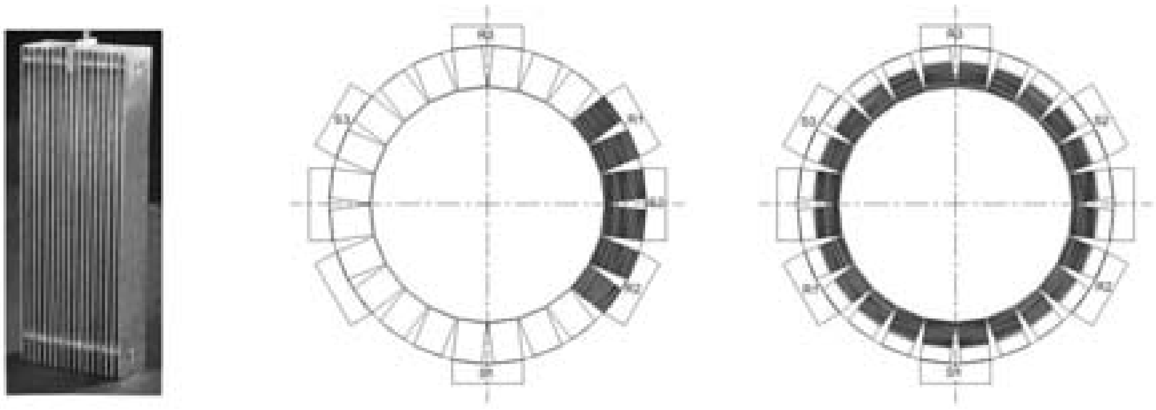

FIG. 2. a) Fuel element; b) One-slab reactor core configuration; c) Ring reactor core configuration.

From May 1965 to April 1985 the SAR-Graz was driven by an annular core with 234 LEU fuel plates. Alternatively, an asymmetrical one-slab loading with 125 HEU fuel plates was used starting in the year 1970. These two configurations are shown in Figs 2c and 2b, respectively. In these figures the positions of the safety plates are labeled S1, S2, and S3 while the control plates are marked by R1, R2, and R3. The hatched areas indicate the positions of fuel elements. The free volume between the fuel elements was filled with graphite segments.

Since 1985 most of the LEU fuel plates have been located in dry storage because the aluminum cladding of about 50% of the fuel plates was damaged by corrosion. Nevertheless, until the year of 1999 a small number of LEU fuel plates have been added to the HEU fuel plates to allow for particular core configurations.

After July 2004 the reactor core served as dry storage for the HEU fuel elements. The configuration of the fuel elements was changed to ensure a non-critical reactor setup.

1.3. Burnup

Because of its particular mode of operation as a zero power reactor the determination of the burn-up was not necessarily easy and we had to develop a particular procedure for this purpose. We decided to use the FIFA definition [4]:

$$\text{FIFA (}\underline{\text{F}}\text{issions per }\underline{\text{I}}\text{nitial }\underline{\text{F}}\text{issile }\underline{\text{A}}\text{tom)} = \frac{N(U235)}{N(U235\,\text{initially})} \cdot 100 \quad [\%],$$

with N(U-235) the fissioned U-235 mass (in grams) per fuel plate and N(U-235 initially) the initially existing U-235 mass per fuel plate. The mass of fissioned U-235 and the burn-up of U-235 of a fuel plate were computed by means of the total Cs 137- activity which was produced during reactor operation from 1965 to 1985 in case of the LEU elements and from 1970 to 2004 in case of the HEU elements. It was our main assumption that there exists a linear correlation between the Cs 137 production and irradiation time. A full account of the procedure applied was given by Pichl and Ninaus in Ref. [4],[5]. The maximum burn-up was estimated to be of the order of 10^{-4}% a value which was later on contested by the experts of the Savannah River Site (SRS) in their evaluation of Appendix A. (Please see also Chapter 2.1.) This required a full documentation of the power load of the reactor according to its log and we reproduce the result in Figs 3a and 3b for the LEU and HEU fuel plates, respectively. It became evident that our estimate was quite reasonable.

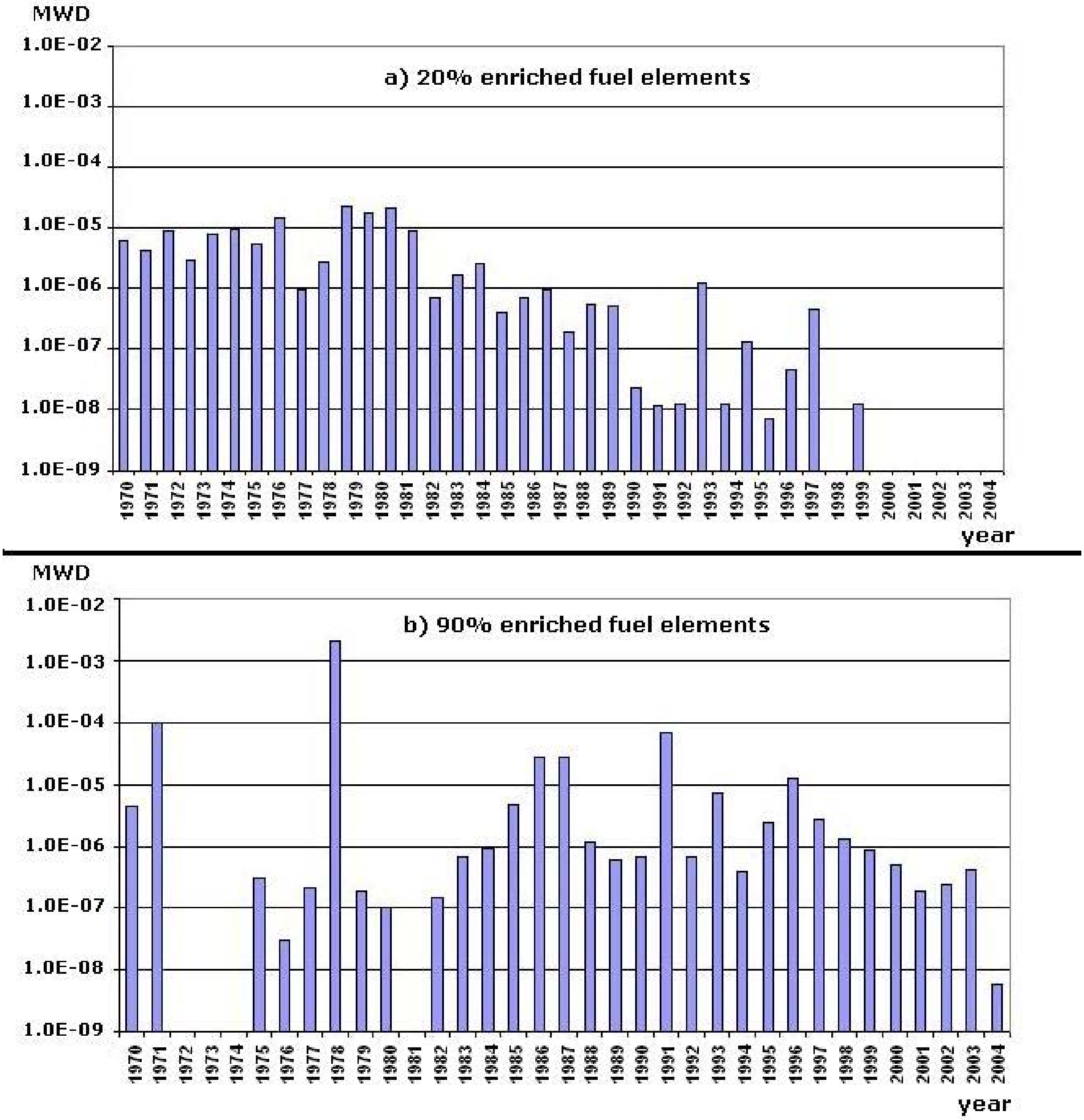

FIG. 3. . Reactor load in Mega Watt Days (MWD) per year a) for LEU fuel plates and b) for HEU fuel plates.

Finally, the average γ-dose rate of the fuel elements in one meter distance was determined to be < 0.5 μSv/h for the LEU fuel elements and < 5.0 μSv/h for the HEU fuel elements. This allowed open air handling of the fuel plates behind a 50 mm lead shield wearing cotton gloves.

1.4. The reactor site

Fig. 4 gives a map of the reactor site. The reactor itself was contained in the reactor hall together with its control panel, monitoring devices, and the dry storage area. Furthermore, a 3 t crane was available within the reactor hall. The reactor hall is connected via a hall way with the main Joanneum Research building with the offices and other small laboratories of Reaktorinstitute Graz on the first floor. This seven floors building also houses a number of offices and laboratories of other research groups not involved in radiation or reactor physics research. The whole complex is situated in a now rather heavily populated area not far from downtown Graz.

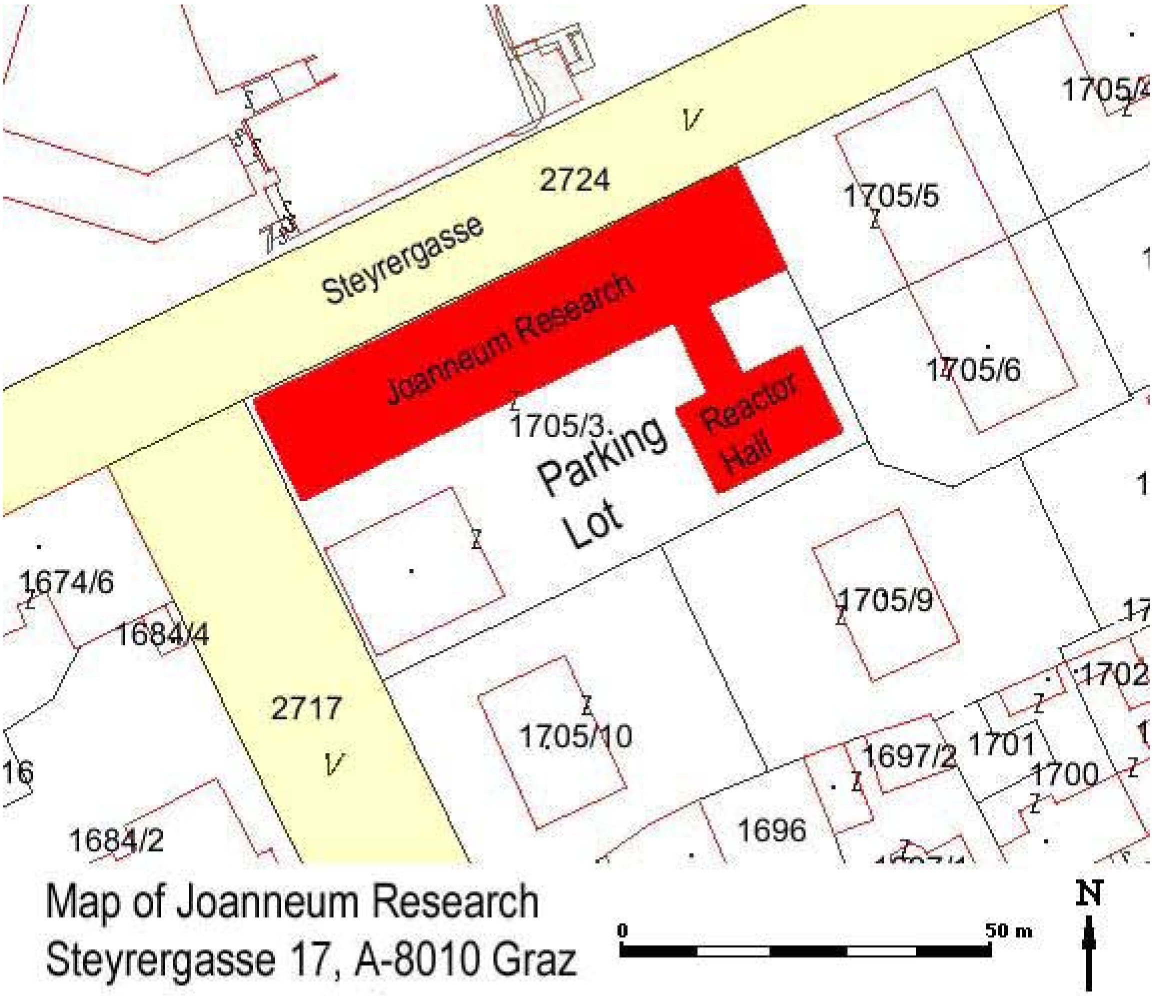

FIG. 4. Map of the reactor site.

The reactor hall itself has a main door which opens to a parking lot usually used by the employees working in the Joanneum Research building. This parking lot was big enough to accommodate easily two trucks with standard ISO-container trailers and a big 90 t mobile crane during the loading operation. There were no interferences from fences, light poles, etc.

2. The transport

2.1. First steps

At the end of year 1999 the Austrian Federal Ministry of Economics and Labour (bm:wa) was informed about the intention of the Reaktorinstitut Graz to repatriate its LEU and HEU fuel plates to the U.S.A. under the US FRRSNF Acceptance Programme. It took almost a year until it was possible to establish direct contact with the programme's manager at the US Department of Energy (DoE). We received a default contract which also contained the forms of Appendix A. This appendix to the contract is to contain a meticulous account of the physical, chemical, and geometrical properties of the spent fuel. As we decided rather early on to transport our fuel canned the appendix had also be organized according to the anticipated content of each transport canister.

A first version of Appendix A was submitted to SRS in the beginning of year 2001. Several iterations were necessary until Appendix A fitted the regulations established by SRS. On 5/6 September 2002 our fuel was inspected by SRS personnel and we received the preliminary acceptance of Appendix A on 9 December 2002 and, thus, our fuel was accepted for repatriation under the FRRSNF programme.

As more and more details about the transport became clear Appendix A underwent a number of additional modifications.

2.2. Organizing the transport

A first window of opportunity for a possible transport opened for the middle of year 2003 by joining a transport originating from Japan at some European Atlantic port of either France or Germany. We learned pretty soon that a land transport of our fuel to such a port was almost impossible because of either bureaucratic problems or exorbitant security costs. It turned out later on that joining this transport would have been impossible anyway because by adding our fuel to the transport would have changed its ranking to Category I causing additional problems and costs.

Nevertheless, the only solution for reaching a port by land was to transport our fuel via Slovenia to the Slovenian port of Koper at the Mediterranean sea. Fortunately, the Slovenian authorities have been very cooperative and agreed in principle to such a transport. The drawback was that we now had to wait for a FRRSNF transport out of the Mediterranean originating either from Greece and/or Turkey. Such a transport was announced by the DoE for the middle of the year 2005.

This gave ample time to organize the transport in detail. Two major restrictions for the loading procedure were the rather low operating height of the 3 t crane (~ 5.4 m) and the need for a 'quiet' loading, i.e.: if possible, the transport should take place without catching public attention. On the other hand, there was the advantage that the use of a dry transfer system seemed to be unnecessary because of the very low radiation level of our fuel. Thus, we placed a limited call for tenders and received two quotes offering two quite different loading procedures. One anticipated the use of a rather short transport cask (~ 3.5 m tall). Using appropriate tools it was planned to place this cask upright into the reactor hall and load it vertically. This would allow for a loading procedure entirely behind closed doors. The second quote suggested the use of a rather tall transport cask (~ 5 m) which was placed horizontally in a standard ISO container. The idea was to move this container partially into the reactor hall and to load the cask horizontally (not the standard loading procedure for such a cask) because of our height limitations. Nevertheless, the transport company deemed a loading procedure lasting a maximum of ten hours feasible given appropriate preparations in the days before the transport. This would allow for a same day in-and-out transport. Certainly, the first solution was technically more elegant and it would have given us more time for the loading procedure itself. Nevertheless, both solutions were deemed feasible and, finally, only financial aspects dipped the scale in favour of the second solution. The contract was signed on 9 September 2004.

We still had no valid contract with US DoE and it lasted until 26 July 2005 until a contract was signed.

Finally, we were informed that the loading operation would take place most likely on 15/16 October 2005 and that our fuel was planned to join a FRRSNF transport from Greece. This date was called off on 8 October 2006 because of security problems. A new date was set for 5/6 December 2005.

2.3. Legal issues - bureaucracy

The chosen transport cask needed an extension of its licence to encompass the transport of Siemens-Argonaut fuel plates. This extension was granted by the US Nuclear Regulatory Commission on 2 February 2005. This revised version was submitted to the Austrian Federal Ministry of Transport, Innovation, and Technology (bm:vit) which validated this licence for Austria on 4 March 2005. The validated licence was then passed on to the Slovenian authorities who followed suit.

The next step was to apply for an export licence at the bm:wa which required and End Use Undertaking to be signed by DoE. It took a while to get this document signed but then the export licence was issued immediately by the Austrian authorities.

The transport itself needed a licence on the basis of the "Austrian Nuclear Non-Proliferation Act" ("Sicherheitskontrollgesetz") which is to be issued by the Austrian Federal Ministry of the Interior (bm.i). Such a licence is issued to the transport company and defines in detail the transport route to be taken within Austria from the point of origin to the nearest border crossing and the security measures to be taken to ensure a safe transport. This part involved Austrian federal and provincial border and security departments which also established a close cooperation with the Slovenian border and security authorities. Our licence was issued at the beginning of October 2005 and then extended to the end of the year 2005. After this license had been issued the Slovenian authorities issued an equivalent license for the Slovenian part of the transport after an additional environmental impact study of the transport had been submitted by the transport company. We also had to provide a copy of the insurance policy proving that our transport was insured according to the "Act on Liability for Damage Caused by Radioactivity 1999" (Austria and Slovenia) and under the "Paris Convention" for the transport of our fuel on open seas.

Finally, we needed a permit by the US Department of Transportation to allow the transport of goods owned by Reaktorinstitut Graz (the spent fuel) to be transported inside the US inside the cask utilized for the transport. This permit was granted on 16 September 2005.

The contract with DoE under the FRRSNF programme required the concurrence of the EURATOM Supply Agency. Thus, three copies of the contract were sent to EURATOM via bm:wa. (Communication with EURATOM was only possible via this Federal Ministry.) Surprisingly, EURATOM required the "EURATOM Safeguard Clause" to be added to the contract as an addendum. The reaction of DoE was a rather surprised one. After quite some negotiations EURATOM finally agreed to a 'side letter' in which Reaktorinstitut Graz acknowledged the EURATOM Safeguard Clause for its transport as long as the transport was inside EU territory.

In a final step SRS was informed about the precise loading scheme of the transport baskets. These baskets had 7 slots and it was requested to determine which transport canister was to be placed in each slot of each basket (see Fig. 5).

2.4. The loading procedure

In preparation of the loading procedure the LEU fuel elements were removed from dry storage and the HEU fuel elements from the reactor core and both were disassembled. Distance holders, bolts and nuts were removed and stored as activated material for later disposal. Finally, the fuel plates were placed into the transport canisters according to a specific loading plan which had previously been transmitted to SRS.

FIG. 5. A transport basket filled with canisters containing fuel plates.

The loading procedure started on 5 December 2005 with the unloading of the auxiliary ISO container in which the transport baskets and a number of tools necessary for the loading procedure like He leak test utility, etc. had been transported. (For this work a 60 t mobile crane was hired.) In preparation of the loading procedure the various canisters were placed into their respective slots of the transport baskets (see Fig. 5). Particular care was necessary in handling the transport baskets because they were still activated/contaminated from previous use. To prevent contamination of the reactor hall's floor it was covered by a special foil provided by the transport company because our contract contained a particular clause which required the transport company to decontaminate the reactor hall in case of contamination caused by the handling of the transport baskets.

On 6 December 2005 the truck with the ISO container which contained the transport cask arrived in early morning because the whole loading procedure was timed to take a maximum of 10 hours for a same day in-and-out transport.

First the 90 t mobile crane lifted the ISO container and moved it to the very end of the trailer. Then the roof was lifted off from the ISO container and the shock absorber was removed. The transport cask itself had been decontaminated by the transport company prior to shipping to Graz. Nevertheless, the radiation protection group of the Government of the Province of Styria made a radiological survey of the transport cask's surface. After this turned out to the satisfaction of this group the trailer was backed partly into the reactor hall and the lid of the transport cask was removed together with the seals. The four transport baskets were then loaded horizontally into the transport cask with the help of our 3 t crane and quite some manual interference, as shown in Fig. 6.

FIG. 6. Horizontal loading of a transport basket into the transport cask.

After loading was completed the seals were re-installed and the lid of the transport cask closed. The helium leak test was started and turned out to give satisfactory results. (It was not necessary to perform the mandatory SIP test because it was waived by DoE prior to transport.) The ISO container was moved out of the reactor hall and the radiation protection group performed the radiological survey of the transport cask following the instructions put forward by Appendix B of our FRRSNF programme contract. As everything was within allowed limits the radiation protection group authorized the transport. The transport cask was now sealed by EURATOM and the transport company. Finally, the shock absorber was remounted, the roof of the ISO container closed, and the ISO container was moved into its original position on the truck's trailer by the mobile crane. The loading procedure was finished after about 9 hours. The completed Appendix B was faxed to SRS immediately after the trucks left the parking lot.

The whole loading procedure was monitored by IAEA, EURATOM, the bm:wa and by the radiation protection group of the Government of the Province of Styria. Public safety requirements had been taken care of by plain cloth police around the clock starting 5 December 2005. As 6 December 2005 was a bank holiday in Slovenia, Slovenian authorities asked for a delayed border-crossing on 7 Dec. 2005 at 22:00 hours. Thus the truck was parked at police headquarters in Graz and closely guarded throughout. The transport followed the predetermined route from Graz to the Austrian-Slovenian border under unmarked police escort and was then transferred to the Slovenian authorities in a procedure both countries agreed upon earlier. The transport to the port of Koper was reportedly uneventful and arrived on time for immediate transfer of the two ISO containers to the ship.

The fuel arrived on 29 December 2005 at the US port of Charleston Naval Base and was immediately transferred to SRS.

3. Summary

The whole process from the decision to close down operation of the reactor by the end of year 1999 to the arrival of the spent fuel at the Savannah River Site by the end of year 2005 took six years. This is certainly quite a time span despite the fact that everything went smoothly and we enjoyed the support of all authorities involved. The main reason was the particular nature of our fuel which very often was regarded to be 'unirradiated' and, thus, not 'self protecting'. This excluded a number of possible transport routes across Europe to a port in Germany or France which would have allowed us to join one of the more frequent FRRSNF transports passing through this area. Being restricted to the Slovenian port of Koper as the only port of exit meant that we had to wait for a transport under the FRRSNF programme originating from other Mediterranean countries. (The transport of our fuel as a single shipment had to be ruled out for financial reasons.) Otherwise, we had no real problems and we certainly benefited from previous experiences made by the Austrian Research Centre Seibersdorf which also repatriated its spent fuel under the FFRSNR programme a couple of years earlier.

At the moment WAGL is being decommissioned and this process will be finished by the first half of October 2006.

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REFERENCES

[1] SIEMENS-SCHUCKERTWERKE A.G., “Der Siemens-ARGONAUT-Reaktor im Rahmen der Arbeiten zur Reaktorentwicklung der Siemens-Schuckertwerke AG”, Siemens-Zeitschrift **12** (1959) 743.

[2] SGOURIDIS, S., et al., “Neutron Physical Investigation on the Shutdown Effect of Small Boronated Absorbing Spheres for Pebble-Bed High-Temperature Gas-Cooled Reactors”, Nucl. Science and Eng. **97** (1987) 20.

[3] SCHÜRRER, F., et al., “Steady-State Neutronic Investigations to the Accident of Water Ingress in Systems with Pebble-Bed High-Temperature Gas-Cooled Reactor Fuel”, Nucl. Science and Eng. **97** (1987) 72.

[4] BÖCK, H., HAMMER, J., “Gamma spectrometry of TRIGA fuel elements”, ATKE **39** (1981) 195.

[5] PICHL, E., NINAUS, W., “A method to determine the burn-up of fuel plates under low neutron fluence operation”, The 4th International Decommissioning Symposium, Knoxville, Tennessee (2000).

The operational and logistic experience on transportation of Brazilian spent nuclear fuel to the United States of America

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Abstract. In 1999 a shipment of 127 spent MTR fuel assemblies was made from IEA-R1 Research Reactor located at the Instituto de Pesquisas Energéticas e Nucleares (IPEN-CNEN/SP), São Paulo, Brazil to Savannah River Site Laboratory (SRS) in the United States. This paper describes the operational and logistic experience on this transportation made by IPEN staff, EDLOW International Company and the Consortium NCS/GNS.

1. Introduction

IEA-R1 is a pool type research reactor, moderated and cooled by light water, and utilizing graphite and beryllium as reflector. The construction is a Babcock & Wilcox design and the first criticality was achieved on 16 September 1957. After initial tests, the reactor started operating at 2 MW. Due to the growth in radioisotope demand, in 1997 after necessary modifications and upgrading process, the power was increased to 5 MW [1].

Along 40 years of the reactor operation, 127 SFA's had been stored at the facility, 40 in a dry storage and the others 87 were stored inside the reactor pool. As reported in the 21th RERTR [2], in 1996 the Brazilian Nuclear Energy Commission (CNEN) started negotiations with DOE to return the SFA's of IEA-R1 Research Reactor to USA. Finally, in 1998, an agreement was achieved between CNEN and DOE and in November 1999, the shipment was realized with success.

2. Fuel information

The SFA´s transported to USA were used in IEA-R1 RR as follows: the first load corresponds to the first core of the reactor. It was composed of U-Al alloy fuel with 20wt% enrichment, having 19 curved fuel plates produced by B&W. These fuel assemblies failed at the earlier stages of the reactor operation, due to pitting corrosion caused by brazing flux used to fix the fuel plates to the support plates. As the burn up and the dose rate was very low, the assemblies were placed in a dry storage composed of horizontal silos in a concrete wall located at the first floor of the reactor building.

These fuels were replaced in 1958 by new ones, also produced by B&W. They were identical to the earlier ones (U-Al alloy, 20wt% enrichment, 19 curved fuel plates) but brazing was not used for assembling. The fuel plates were fixed mechanically to the support plates.

The third load corresponds to a complete substitution of the core. Fuel assemblies made with U-Al alloy, 93 wt% enrichment, having 18 flat fuel plates were bought from UNC (USA). To comply with the new flat plate type fuel, the control rod mechanical concept was also changed from rod type to fork type (plate type), and the control fuel assemblies were fabricated by CERCA (France), using U-Al alloy, HEU and flat plates.

The fourth load was characterized by the restriction of buying HEU fuel in the international market. IPEN bought, from NUKEM (Germany), 5 fuel assemblies of UAl_x-Al dispersion type, with 20wt% enrichment and having 18 flat fuel plates per fuel assembly.

All the four loads, summarized in Table 1, were returned to the United States, after being inspected by Brazilian [3] and DOE teams [4].

It is important to mention that after the four initial loads, all the fuel assemblies used in IEA-R1 were constructed in IPEN. Many of them with uranium from USA origin, and not yet returned to the US.

TABLE 1. SUMMARY OF THE FUEL ASSEMBLIES USED IN IEA-R1 REACTOR CORE

LOADING	FIRST	SECOND	THIRD	FOURTH
Year	1957	1958	1968	1981
Country	USA	USA	USA/France	German
Enrichment	20	20	93	20
Standard assembly	34	33	33	5
Control assembly	5	4	10	-
Partial assembly	1	2	-	-
Total assembly	**40**	**39**	**43**	**5**

3. Companies contracted for the transport operation

The contract between the CNEN and the Department of Energy (DOE/USA) was signed in 1998. Edlow International Co. and the Germany Consortium formed by Nuclear Cargo + Services (NCS) and Gesellschaft fur Nuklear-Service (GNS) were hired to performe the transport. Tec Radion Comercial Ltda (TRION) was subcontracted by Edlow to provide the necessary local infrastructure for the loading, transport and customs documents.

The German Consortium provided 4 transport casks (two GNS-11 and two GNS-16), a transfer cask, equipment and experts to handling their equipment. IPEN/CNEN-SP performed the necessary work to accomplish the Brazilian legislation as the export license, a detailed transport and security plan, safeguards documents, and the Appendix A. It also provided the operational and radiological protection support to the entire operation.

4. Transport equipment description

The transport cask GNS 11 and GNS 16 are designed in a sandwich construction. The cylindrical cask basically consists of the following components: inner liner with inner liner bottom, lead filling, wall with bottom plate, side wall cover sheet with spacer wire, head ring, primary lid and protective plate. The maximum weight of the cask is 13 230 kg [5,6].

The components of the cask body and the primary lid are manufactured in stainless steel. In the terms of the transport regulations, the “leak-tight containment” consists of the inner liner, the inner bottom plate, head ring, primary lid, with the bolt joint, and the internal seal of the two concentrical Viton seals.

Cap screws are used in order to fasten the primary lid. The closure lid is also fastened to the primary lid using cap screws. In order to achieve the shielding effect, the space between the inner liner and the shell is filled with lead casting. A pair of trunnions are bolted on to the head ring in order to attach handling devices. During transport, the cask is provided with a protective plate. In order to reduce the shock loads arising during the eventual drop of the cask, as stipulated for type B packaging, impact limiters made of wood with a steel-plate shell are attached to the ends of the cask body on the lid and bases sides. Because of the different geometry of the fuel assemblies (FA’s) to be transported, the inner cavity of the cask can accommodate any one of three different baskets, as follows: FR 2/33 to accommodate 33 box-shaped FA´s, FR 2/28 for 28 tubular FA´s and FR 2/15 for 90 rod-shaped

TRIGA FA´s. The two casks, GNS 11 and GNS 16, are similar. A summary of the characteristic data for fissile material and burn-up or box-shaped MTR fuel assemblies that can be transported on the two casks is shown in Table 2.

TABLE 2. CHARACTERISTIC DATA FOR FISSILE MATERIAL AND BURN-UP – BOX SHAPED MTR-FA

	GNS-11	**GNS-16**
Max. Number of FE per cask	33	33
Max. FE lenght, mm	630	915
Max. FE cross section, mm^2	81 x 76.1	84 x 77
Max. FE mass, kg	2.65	7.0
Min. Cooling time	180 days	> 1.5 years
Max. Initial enrichment, weight % of ^{235}U	94	95.1/HEU 45.7/MEU 20.3/LEU
Max. Initial weight of ^{235}U,g	268	459/HEU 328/MEU 420/LEU
Max. Burn-up, MWd/FE		184/HEU 181/MEU 222/LEU
Max. decay heat, W	48.5	40
Max. FE length, mm	610	-
Max. activity (x E14 Bq)	3.3E02	6.3E2

5. Fuel cutting equipment

Before the beginning of the loading operation, the external part of the 19 control fuel assemblies were cut leaving 1.27 cm away from the interior fuel plates. This cut was necessary due to the cask length limitation and a SRS request. The cutting operation of five control fuel assemblies stored in the dry-storage was performed in the first floor of the reactor building. A conventional saw normally used for aluminum profile cut was utilized. The assemblies were manually removed, one by one, from the carbon steel piping of the dry storage and placed in a lead shielding. A second technician cut the plastic that was involving the assembly and took it to the saw for the cutting. The cutting pieces were put in a special place as waste and the assembly was stored again. This operation was possible due to the low burn up and dose rate of these SFA´s.

For the cutting of the 14 SFA´s stored in the reactor pool, it was used a underwater saw machine specially designed and constructed in Brazil under supervising of Edlow/Trion. This saw machine was positioned 2.5 meters below the surface of the water inside an aluminum box covered with an acrylic plate. The saw was of stainless steel construction with an electrical motor that remained above the surface of the water and was controlled from poolside. The fuel assembly was fixed pneumatically inside of the aluminum box.

6. Loading and transport description

On September 16, four containers with two GNS-11 German casks and equipment arrived at IPEN. The two GNS-16 casks had already arrived in IPEN on 7 October. German experts supported by IPEN technicians and the transportation company staff hired by Edlow/Trion removed the equipment from the containers and placed them on a truck which transported the equipment to the reactor building. With the help of a crane with 25 tonnes capacity, equipment were placed on a iron platform located in front of the access hall of the building. With the help of a mobile lift, the platform was moved inside of the reactor building and positioned under an access previously open to the 3rd level (pool surface level). Part of the equipment, as the transfer cask, rotary lid and water tank were lifted by the reactor crane with 10 tonnes capability until the 3rd level. In the same way , the cask was moved inside of the reactor building and positioned in the first level, under the access to the 3rd level.

On September 21, the primary lid was removed from the GNS-11 cask and lifted to the 3rd level. The rotary lid was positioned in the upper part of the cask and a cold test, using a dummy assembly, was made by the Brazilian and German teams. A transfer cask, 4 tonnes weight was used to transfer the assemblies from the wet storage to the transport cask. After the succes of the cold test, t hen the SFA's, one by one, were lifted from the storage racks inside of the reactor pool using a special handling tool and placed inside of a plastic tube located on a metallic platform located about 2 meters below the water surface. In the sequence, the transfer cask was sunk inside of the reactor pool, over the assembly to be removed, and a special tool took the assembly and put it inside of the transfer cask which was lifted by the reactor crane and positioned on top of the rotary lid and transport cask located in the first level, as shown in Fig. 1. Finally, the assembly was guided to one of the 33 positions of the cask. In order to remove the rotary lid and put the primary lid, after cask loading, a water tank was positioned on top of the cask and filled with 4 000 liters of water. Finally, the cask was closed and the water was drained to a water tank positioned close to the transport cask.

This operation was repeated for the 87 assemblies stored in the wet storage. For the others 40 SFA's stored in the dry storage, the transfer cask was not used. Instead of it, a water tank was positioned in the upper part of the cask and the operation was performed as follow: the SFA's were taken from the storage, one by one, by hand, and put in a cylindrical lead shield positioned close to the cask. An operator located in the 3rd level, using a nylon rope with a hook in its extremity lifted the SFA and put it inside of the water tank. Finally, the operator guided the SFA visually to the final position in the transport cask.

On October 15 the four GNS casks had been loaded with the 127 Brazilian spent fuel assemblies. Then, the decontamination procedures were performed.

On October 20 and 21 all the equipment and cask were removed from the reactor building to the containers. The casks were stamped and controlled by safeguards inspectors from ABACC (Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials) and supervised by IAEA.

On November 3, the transport operation was initiated after approval from the Brazilian regulatory bodies (Nuclear and Environmental). The transport licenses were issued by CNEN and IBAMA (Environmental Brazilian Agency) which required detailed documents related to the transport, radiation and physical protection as well as an evaluation of the environmental impact. Also the GNS 11 and GNS 16 certificates issued by American and German authorities had to be revalidated in Brazil. It is worthwhile to mention that to obtain these licenses, an enormous effort was done by IPEN staff. Also opposition from Greenpeace, local politicians and harbor union demanded a political work to overcome this opposition and avoid legal prosecute against the operation, including debates and press clearance.

On November 4 at down a huge convoy consisting of 4 trucks escorted by Federal, State and County Police arrived in the port of Santos. It is also worthwhile to mention that the Highway and the main avenues and streets in São Paulo and Santos had been closed for circulation during the operation. At 2h10min am, the shipment of the containers started, and it was concluded after 42 min. Before and during all shipment the workers had been monitored by IPEN radiation protection staff. At 4h50min am, the ship left the port escorted by boats of the federal police. In the exit of the port, these boats were substituted by a frigate of the Brazilian Navy which followed the ship until a distance of 200 miles away from the Brazilian coast.

FIG. 1. View of the transfer cask used to transfer the assemblies to the transport cask.

7. ^{137}Cs Leaking rate and sealing system of the casks /5//6/

After the cask loadding and before the transportation, two tests were performed. The first one was to evaluate the ^{137}Cs leaking rate inside of the cask. In order to perform this test, three water samples were taken from each cask after 0, 4 and 12 hours. The water sample was collected in a small plastic bottle (1 500 ml) and submitted to gamma-ray spectrometry analysis. All bottles used for sampling were identical. The results of this test are shown in Table 3.

TABLE 3. RESULTS OF THE 137-CS LEAKING RATE TESTS IN THE CASKS

Cask	sample 1 dpm/ml	Sample 2 dpm/ml	sample 3 dpm/ml
GNS11-1	1.24	1.86	3.96
GNS11-2	27.6	12.8	24.0
GNS16-1*	1.48	14.0	24.6
GNS16-2*	233.8	875.1	743.3

Obs. limit value for the GNS after 12 hours is below 992 dpm/ml
*this casks was loaded with the SFA´s stored in the dry-storage

A second test was performed in order to verify the sealing system. The primary lid and the protective caps with their screws and testable O-rings are decisive for guaranteeing the retention of the inventory. Grooves are turned into the primary lid in order to accommodate two O-ring seals on the bottom sides of the lid flanges. The O-ring which is part of the containment boundary is insert into the inner groove on the primary lid. The O-ring inserted in the outer groove is not component of the containment boundary. No account is taken of its sealing effect. It forms a testing volume for the leak test. Each protective cap has a groove to accommodate the respective O-ring of the containment boundary.

The leak tightness of the primary lid is proven with a pressure-drop test via the testing connection "B" of the O-rings of the primary lid. For the protective caps the pressure drop tests is performed on a test volume built by a test adapter covering each cap. The measured leakage rate thus is the combination of leakage through both O-rings or other seals forming the boundary of each test volume. The leakage rate for the O-rings, which are part of the containment boundary, thus in reality is lower than the measured value. However, this is ignored and the measured value of the sealing assigned to the O-rings, which are part of the containment boundary.

As a consequence of Type B tests no systematic deterioration of the sealing characteristics can be assumed. This was proven by tests on transport casks with a comparable sealing system. For the transport cask GNS 11 this test is the guarantee that during and after the Type B tests, the leakage rate specified for normal conditions of transport will not be exceeded. For the GNS 16, the test is similar. Results are shown in Table 4.

TABLE 4. SEALING SYSTEM TEST

Cask	Test nº 1 $hPa.l.s^{-1}$	Test nº 2 $hpa.l.s^{-1}$	Test nº 3 $hPa.l.s^{-1}$
GNS11-1	9.1E-06	9.6E-06	1.9E-05
GNS11-2	2.1E-06	9.9E-06	3.2E-06
GNS16-1	8.4E-06	1.2E-07	-
GNS16-2	7.0E-07	1.9E-06	-

Limit value for the GNS11: below 1.0 E-04 $HPa.l.s^{-}$
Limit value for the GNS16: below 2.0 E-05 $HPa.l.s^{-1}$

8. Conclusions

IPEN-CNEN/SP considered that the loading operation, shipment and transport were performed with success once it occurred without any incident and the 127 spent fuel assemblies burned in the reactor IEA-R1 in the last 40 years were loaded and transported to Santos as planned in the Transport and Security Plans. Also, all the loading operation was successfully achieved due to the professional and friend relationship between the Brazilian and German teams.

REFERENCES

[1] MAIORINO, J.R., et al., The Conversion and Power Upgrading of IEA-R1: Experience and Perspectives (Proc. 21st RERTR, October 18-23, 1998), São Paulo, Brazil. Available at: http://www.rertr.anl.gov/NatProgram98/Maiorino.pdf

[2] FRAJNDLICH, R., et al, Experience of IEA-R1 Reactor Spent Fuel Transportation back to USA (Proc. 21st RERTR, October 18-23, 1998), São Paulo, Brazil.Available at: http://www.rertr.anl.gov/Fuels98/SpentFuel/RFrajndlich.pdf

[3] PERROTTA, J.A., et al, Experience on Wet Storage Spent Fuel Sipping at IEA-R1 Brazilian Research Reactor – IPEN/CNEN-SP, Annals of Nuclear Energy, **25** - 4, (1998) 237-258

[4] A Team of personnel from WSRC, DOE-SR, SAIC and INEL., “Characterization of IEA-R1 Spent Nuclear Fuels at the IPEN, São Paulo, Brazil, July 22 – July 31, 1996” – Internal report.

[5] Application Documents for the Transport Licensing of the Type B(U)F Package Transport Cask GNS11, Gesellschaft fur Nuklear-Behalter mbH (GNB).

[6] Application Documents for the Transport Licensing of the Type B(U)F Package Transport Cask GNS16, Gesellschaft fur Nuklear-Behalter mbH (GNB).

Chilean experience with shipment of research reactors spent fuel

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Abstract. MTR type fuel assemblies used in the RECH-1 research reactor were returned to the United States in the frame of the spent nuclear fuel acceptance program. Two shipments were programmed to return the total of fifty eight spent fuel assemblies to Savannah River Site, South Carolina. The first shipment containing 28 spent fuel assemblies was carried out in August 1996, while the second containing 30 spent fuel assemblies was carried out in December 2000.

Prior to the first shipment several and most diverse activities related to negotiations and management of this operation were carried out. The main activities were the negotiation of contractual matters, safeguard issues and managerial activities related to transportation, security, budget, cost scheduling, and public relation with the media. Above all these activities was the coordination and collaborative efforts performed between the staff of the Chilean Nuclear Energy Commission (CCHEN) and the Department of Energy (DOE), including Washington DC and Savannah River Site personnel. The second shipment made use of the experience gained and the lessons learnt during the first shipment.

NAC International Inc. provided the necessary equipment for both shipments of the spent nuclear fuel to Savannah River Site. These were the NAC LWT shipping cask and the MTR fuel dry transfer system. To move the heavy materials and facilitate the operations involved in the loading and transport of the fuel assemblies, a crane of 50 metric ton of capacity, trucks, forklift and pneumatics scaffoldings were rented locally. Detailed descriptions of the diverse activities are presented in the paper.

1. Introduction

CCHEN operates two nuclear research reactors, RECH-1 and RECH-2, both are pool type. The RECH-1 research reactor is located at La Reina Nuclear Center in Santiago and the RECH-2 reactor is located at Lo Aguirre Nuclear Center near Santiago.

The first criticality of RECH-1 was achieved on 13 October 1974 using highly enriched uranium (HEU) fuel assemblies. The reactor uses light water as moderator and coolant and beryllium as reflector. For most of the time the reactor has been operated at the nominal power of 5 MW in a continuous shift of 24 hours a week, 48 weeks a year. The reactor has an annual shutdown of 3 to 4 weeks for maintenance usually during the early summer. The emphasis of the utilization of the RECH-1 reactor is placed on radioisotopes production, neutron activation analysis, beam experiments, in core experiments, neutron irradiation, and neutron radiography.

RECH-2 is moderated and cooled by light water and it utilizes graphite as reflector. The first criticality was achieved in February 1977. The reactor has a license to operate at the power level of 2 MW using HEU fuel assemblies; however, due to lack of a utilization program the reactor operation has been stopped.

A total of 58 spent fuel assemblies from the RECH-1 reactor were returned to Savannah River Site, South Carolina, in two shipments; both within the frame of the U. S. Foreign Research Reactor Spent Nuclear Fuel (US FRRSNF) acceptance program. The first shipment containing 28 spent fuel assemblies was carried out in August 1996, while the second containing 30 spent fuel assemblies was carried out in December 2000.

Regarding the first shipment, on 14 May 1996 CCHEN received a letter from the Embassy of the United States in Santiago about the decision of the United States Department of Energy to accept and manage in the United States the spent nuclear fuel, all which contains uranium enriched in the United States, over a thirteen years period. On 31 May 1996, CCHEN received a second letter from the Embassy of the United States in Santiago informing of the Record of Decision (ROD) for an environmental impact statement of a policy to manage foreign research reactor spent nuclear fuel. Additionally, the letter advised on the convenience of receiving an assessment team to familiarize reactor operators with the program, and to gather necessary information and make essential contacts in advance. A tentative shipment of all eligible South America spent fuel was planned for August 1996.

On 21 June 1996, a U.S. delegation visited the RECH-1 research reactor and six days latter the CCHEN's Board of Directors approved to participate in the first shipment of foreign research reactor spent nuclear fuel from South America to the United States. Due to the short period of time to do the necessary negotiations and preparations, the first shipment from South America comprised only spent fuel from Chile and Colombia. The transport was initiated on 27 August 1996 when a container loading 28 spent fuel assemblies left La Reina Nuclear Center for the port of San Antonio. The shipment was successfully completed in September 1996.

The second shipment was formally initiated on 28 February 2000 when a U. S. Department of Energy delegation visited the RECH-1 reactor. The visit was done to formalize specific requirements that were contained within the contract between CCHEN and DOE and to coordinate efforts to conduct a shipment from CCHEN to the United States under the US FRRSNF acceptance program. Several agreements and actions were established as a result of the visit. Finally, on 23 December 2000 the remainder eligible spent fuel assemblies from the RECH-1 reactor were transported to the port of San Antonio where joined the vessel already containing spent fuel from Argentina.

2. Spent fuel type and inventory

Both reactors, RECH-1 and RECH-2, utilize MTR type fuel assemblies. The RECH-1 reactor initiated its operation utilizing HEU (80% of ^{235}U) fuel. The United Kingdom Atomic Energy Authority (UKAEA) in Dounreay, Scotland, fabricated 58 fuel elements of 16 flat fuel plates each using highly enriched uranium provided by the United States. Later, other 40 fuel assemblies were fabricated by the UKAEA but this time using British enriched uranium (45% of ^{235}U). The number of fuel plates per assembly was maintained with minimum changes in the fuel assembly cross-section (74.7 mm x 74.6 mm) and in the overall fuel assembly length (985.9 mm). From 1985 to 1998 the reactor operated with a mixed core configured with HEU fuel assemblies of two different enrichments (80% and 45% of ^{235}U).

Due to the reduced HEU fresh fuel assemblies inventory of the RECH-1 reactor, and after CCHEN developed the capability to produce U_3Si_2-Al fuel, it was decided to fabricate LEU fuel for the RECH-1 reactor. Once the fuel fabrication plant and the manufacturing and quality control procedures were licensed, to permit the production of fuel assemblies based on a dispersed fuel containing LEU fuel as U_3Si_2, a fabrication program started to manufacture 50 MTR LEU (19.75% of ^{235}U) fuel assemblies with Russian origin uranium. The number of plates, geometry and dimensions of the LEU fuel assembly were the same as the HEU (45% of ^{235}U) fuel assembly. A total of 47 LEU fuel assemblies were finally manufactured and delivered to the RECH-1 reactor. From 1998 to 2006 the RECH-1 reactor operated with a mixed core; this time configured with HEU (45% of ^{235}U) and LEU fuel assemblies.

The full core conversion of the RECH-1 reactor was reached on 11 May 2006 when the last 10 HEU fuel assemblies were removed from the reactor core. The full conversion of the RECH-1 research reactor was included in the contract between the U.S. Department of Energy and the Chilean Nuclear Energy Commission for a second shipment of Chilean spent fuel. Therefore, the contract stipulated that "Customer agrees not to use HEU fuel in the reactor other than during any transition period needed to allow the reactor to be converted to use low enriched uranium fuel".

For the RECH-2 research reactor, a total of 31 MTR type fuel assemblies were manufactured by the Joint of Nuclear Energy (JEN), Spain, using French enriched uranium (90% of ^{235}U). Due to scratches that were detected in several outer fuel plates, the fuel assemblies were disassembled by the Chilean Fuel Fabrication Plant and each fuel plate was inspected. The selected plates were enough to reassemble 29 fuel assemblies. Each fuel assembly contains 18 fuel plates, a cross-section of 76.11 mm x 76.02 mm with an overall fuel assembly length of 949 mm.

At present, the inventory of fuel assemblies in Chile is: (a) RECH-1: 40 HEU (45% of ^{235}U), 47 LEU (19.75% of ^{235}U) and 1 experimental LEU fuel assembly, and (b) RECH-2: 29 HEU (90% of ^{235}U) fuel assemblies and 36 HEU (90% of ^{235}U) fuel plates. Thus, the total inventory of the Chilean research reactors is 117 MTR type fuel assemblies plus 36 HEU fuel plates. The total HEU fuel of the RECH-1 reactor is spent fuel, which is decaying in the reactor pool. The 29 HEU fuel assemblies of the RECH-2 reactor are slightly burnt.

3. Spent fuel management

At the RECH-1 reactor, fuel assemblies are considered spent fuel after reaching a burn up greater than 40%. When the fuel assemblies reached the discharged burn up they are removed from the core and stored in racks which lay on the bottom of the reactor pool. In order to measure the fuel burn up of irradiated fuel assemblies, RECH-1 has an in-pool facility which uses the gamma spectrometry technique. This facility has also been used to measure the burn up of fuel assemblies with short decay period [1].

Additional to the burn up measurement, each spent fuel assembly is subjected to sipping test and visual inspection. The sipping test is being performed in order to detect leak of fission products, while the visual inspection provides information of the fuel structural conditions and of the type of eventual existing damage on irradiated fuel assembly. The sipping test and visual inspection capability were not available at the RECH-1 reactor prior the two shipments. They were developed in the frame of the IAEA Technical Cooperation Regional Project RLA/4/018: Management of Spent Fuel from Research Reactors in Latin America.

Since the spent fuel assemblies of both reactors have to remain in wet storage, the water quality must be maintained to prevent corrosion. The determination of pH, conductivity, and concentration of chloride, iron and copper ions is a standard practice in both reactors and it has been carried out using the Norma CCHEN Nº 3.2.1 [2]. Water samples from the reactor pools are regularly sent for radiological analysis using gamma spectrometry of high resolution. One of the most important radionuclide controlled is ^{137}Cs which is extremely soluble in water. The activity of ^{137}Cs is directly related with failed fuel assemblies.

4. Negotiations and management

In order for the CCHEN to carry out the several and most diverse activities related to negotiations and management, it was decided to have a centralized and vertical organization during the shipments. CCHEN has the role of Reactor Operator and Nuclear Regulatory Body providing it with a very fluent communication and relationships; although, each of these roles is well differentiated and independent within our institution.

During the first shipment the negotiation of contractual matters, safeguard issues and managerial activities related to transportation, security, budget, cost scheduling and public relations with the media were particularly complicated. All these diverse activities were easier during the second shipment due to the experience gained and the lessons learnt during the first shipment.

Above all these activities were the coordination and collaborative efforts performed between the staff of CCHEN and DOE, including Washington DC and Savannah River Site personnel. To overcome parallel initiatives and to avoid misinterpretations it was decided at CCHEN to operate in both shipments under a single general manager. The main duties of the general manager were to interact with counterparts in the United States, to coordinate the technical activities, and to report and consult permanently with the Executive Director of the CCHEN on decision making issues.

4.1. Contractual matters

With respect to the legal framework related to both shipments, there were a number of matters which needed a quick response. The way this was resolved was through an intense communication link between the general manager at CCHEN and the United States counterparts. This was also facilitated by the high level of expertise of DOE, Edlow International Co. and NAC International personnel in charge. The issues related to nuclear liability and title transfer location were particularly complex in the first shipment and required some significant efforts of all the parties involved.

4.2. Safeguards

Based on the safeguards agreement between the Republic of Chile and the International Atomic Energy Agency, it was necessary to apply in the United States safeguards to the returning spent nuclear material. This nuclear material was under IAEA's control during its entire utilization and decay period at the RECH-1 reactor. This was strictly a negotiation process that took place between the IAEA and the United States authorities.

4.3. Public relations

One important aspect during the preparation of the two shipments was the CCHEN's handling of the media with respect to public reaction to the transportation of nuclear material. In this respect, it was decided to maintain a low profile during the entire operation and only to inform the public through a formal press conference. This conference was called by special invitation to well recognized press media, including TV, radio and printed press. The main reason to do this was to inform the public about the character and benefits of this initiative; as well as of all the safety and security measurements being taken for the handling and transportation of the spent nuclear fuel.

It was clear to media that the work of all parties involved was thoroughly studied, planned and executed in a serious, detailed and responsible manner. Moreover, it was emphasized that CCHEN was fully complying with the Chilean law and the international regulations related to the transport of nuclear material. The reaction of the Green Peace followers in Chile had no repercussions in the transport of spent fuel from Chile to the United States.

4.4. Others managerial matters

Several activities were accomplished to insure the physical protection, and arrangements with customs and the National Emergency Office.

Among the physical protection tasks the very first one was the notification to the highest authorities of the international airport of Santiago, port of San Antonio and Police authorities.

For the first shipment, the General Director of Customs was informed by the CCHEN's Executive Director through a formal letter about the shipment; thus, it was initiated a process and set up the framework for the follow-up events. All the paperwork and authorizations to do the temporary admission of the shipping cask and equipment, and also the re-exportation of the spent fuel assemblies, were done efficiently and in prompt manner. In particular, the operations at the port of San Antonio in all aspects were really impeccable. The paperwork done during the first shipment was repeated for the second shipment.

The communication at the highest level of the National Emergency Office was facilitated due to the unique position of CCHEN within the Government structure. As a result of this, all the planning and coordination at the various levels was straight forward and expeditious. From this experience, we learnt that a high degree of coordination and accurate information was needed to provide to the

Authorities in charg, and in this way to guarantee a quick response and to maintain the complete operation under control.

5. Technical activities

Together with the negotiation and management activities it was necessary to collect the technical information of the spent fuel assemblies and to initiate several technical activities at the RECH-1 research reactor. Among the most important activities was the gathering of the technical information to fill out the Appendix A (Spent Nuclear Fuel Acceptance Criteria). Other activities were to prepare the reactor to accept the necessary equipment to transfer the spent nuclear fuel assemblies from the reactor pool to the shipping cask situated outside of the reactor building, to write the physical protection plan for loading and transport the spent nuclear fuel assemblies, the risk prevention plan and the operation procedures for preparing the RECH-1 reactor for the transfer operations. The plans and procedures were approved by the Regulatory Body.

During the meeting with the U. S. Delegation at La Reina Nuclear Center on 21 June1996 it was agreed that 28 spent fuel assemblies could be sent in the first shipment from the RECH-1 reactor to Savannah River Site. Another important agreement was that for the first shipment the fuel assemblies would not have the end-fitting cropped. However, for the second shipment it was necessary to crop the end-fitting of the total 30 spent fuel assemblies. The fuel cutting equipment was designed by NAC International Inc. and it consisted mainly in a pipe cutter with an adapted large cutter wheel, an air motor, a collar, a tool splice pin, and a base plate. The cutting equipment was submerged about 2 meters in the reactor pool and it was operated from the poolside. The operation for cropping 30 spent fuel assemblies took 2 days and it generated a few amount of aluminium shaving which was collected on the base plate.

NAC International Inc. provided the necessary equipment for both shipments of the spent nuclear fuel to Savannah River Site. These were the MTR fuel transfer system and the NAC LWT shipping cask. To move the heavy materials and facilitate the operations involved in the loading and transport of the spent nuclear fuel assemblies a crane of 50 metric ton of capacity, trucks, forklift and pneumatic scaffoldings were rented locally.

5.1. Technical information of the spent nuclear fuel

The physical and chemical characteristics, isotopic composition, dimension and weight of the spent fuel was given in Appendix A of the contract as required by the Savannah River Site for the acceptance of the fuel assemblies.

As mentioned before, the HEU MTR type fuel assemblies of the RECH-1 research reactor were manufactured by the UKAEA, at Dounreay, Scotland in 1973. Each fuel assembly comprises of sixteen flat plates, which are composed of highly enriched uranium (80% of ^{235}U) aluminium alloy sandwiched between aluminium. The outer fuel plates load a half of the uranium content of the inner fuel plates. A summary description of the fuel assembly and the fuel plate is shown in Tables 1 and 2 respectively.

TABLE 1. FUEL ASSEMBLY DESCRIPTION

Number of fuel plates	16 (14 inner plates + 2 outer plates)
Over-all dimensions, cm	99.3 × 7.5 × 7.47
Over-all weight, g	4 788
Total weight of U, g ± g uncertainty	206.3 ± 3.0
Total weight of ^{235}U, g ± g uncertainty	165.0 ± 2.8
Enrichment, % ± % uncertainty	80
Side plate material	Aluminium BS 1474 grade N4
Side plate dimensions, cm; weight per plate, g	65.09 × 7.46 × 0.483; 472
Weld material	Aluminium

TABLE 2. FUEL PLATE DESCRIPTION

Description	Inner plate	Outer plate
Nominal dimensions (include clad & bond), cm	62.55 × 7.163 × 0.153	65.09 × 7.163 × 0.153
Nominal total weight of fuel plate, g	193.7	198.0
Nominal dimensions of fuel meat, cm	59.69 × 6.015 × 0.061	59.69 × 6.015 × 0.061
Nominal total weight of fuel meat, g	55.44	50.45
Weight of total uranium, g ± g uncertainty	13.75 ± 1.00	6.88 ± 0.25
Weight of ^{235}U, g ± g uncertainty	11.0 ± 0.8	5.5 ± 0.2
Chemical form of the fuel meat	U-Al alloy	U-Al alloy
Alloy or compound material	Aluminium	Aluminium
Dispersing material, weight, g	41.69	43.58
Cladding material	Al BS 1470 grade 1B	Al BS 1470 grade 1B
Clad thickness, cm; total clad weight, g	0.046; 138.3	0.046; 147.5

Based on the irradiation history of each spent nuclear fuel assembly, the additional information required in Appendix A was determined. Particularly, the following parameters were evaluated: burn up; the content of Special Nuclear Material (SNM) after irradiation; period of time that the fuel assembly stays in core; irradiation time; cooling time; energy obtained per fuel assembly; dose rate at 1 meter in air, and decay heat. A summary of this information is shown in Table 3.

TABLE 3. SUMMARY OF FUEL IRRADIATION HISTORY

Description	First shipment	Second shipment
Number of fuel assemblies	28	30
Average burn up, %	48.8	49.7
Cooling time[a], years	Min.: 4.65; Max.: 8.74	Min.: 2.33; Max.: 8.20
Decay heat[a], W[a]	As of 25-Jul-1996 Min.: 3.11; Max.: 5.46	As of 31-Dec-2000 Min.: 2.90; Max.: 10.27
Total decay heat, W	96.8	162.9
Dose rate at 1 meter in air[a], rem/h	Min.: 53.9; Max.: 72.3	Min.: 56.3; Max.: 112.8
Activity[a], TBq	Min.: 7.1; Max.: 9.5	Min.: 7.4; Max.: 15.0

[a] Determined value per fuel assembly.

5.2. Reactor preparation

Another important activity was to prepare the RECH-1 reactor to load the spent fuel assemblies into the shipping cask using NAC's fuel transfer system. This is a dry transfer system consisting of a transfer cask with basket grapple, a transfer cask carriage, a cask adapter and a pool adapter. Due to physical restrictions at the RECH-1 reactor, the transfer cask was used to move the spent fuel from the reactor pool to the shipping cask located right outside of the reactor building.

Complete technical information of NAC's equipment was received in advance. Using this information, the reactor personnel prepared the platform of the reactor and selected the proper location to install the pool adapter base to support the pool adapter. The platform of the reactor and all working areas were cleared to prevent accidents and to provide enough space for workers.

To prevent any difficulty during the loading operation, a complete inspection of the 20 metric ton overhead crane and the air compressor system was done. The latter one was needed for the pneumatic operation of the basket grapple and tools.

A line of demineralized water was installed for decontamination purposes and to fill out the cask after all fuel assemblies were loaded, with the purpose to take water samples to confirm the acceptable level of ^{137}Cs activity for the shipment. Helium was required for leak testing of the shipping cask closure lid. For this reason, a bottle of helium was provided at the operation area. Helium was also used to fill the cask cavity and maintain the fuel assemblies under an inert atmosphere during the transport.

The contract and the Appendix B (Transport Package (Cask) Acceptance Criteria) required that a water sample of the storage pool should be taken and shipped in accordance with the instructions received from Savannah River Site.

5.3. Loading operation

For the first shipment, the ISO container housing the shipping cask and the necessary equipment loaded in the Unites States arrived in Santiago in a Antonov 124 cargo plane. CCHEN's personnel received the material at the airport and coordinated the transportation to the reactor site. For the second shipment, the containers with the shipping cask and the necessary equipment arrived by ship at the port of San Antonio.

The loading operation was based on the NAC's procedure for the MTR Fuel Dry Transfer System used in conjunction with a NAC LWT shipping cask. This procedure provided the necessary steps to inform the user with the operation features of the system; to assist the user to prepare the requirements for the operation and to operate the system,. In spite of the details contained in this operating procedure, the presence of qualified personnel of NAC International Inc. was essential to expedite the preparation before and during the loading. In this sense, the procedure is meant to be utilized as a guide by experienced personnel.

After an inspection for damage, the equipment was removed from the boxes and set up at the designed location. When the lid of the ISO container was removed, health physics survey of the shipping cask and adjacent surfaces of the container were performed.

Once the top and bottom impact limiters from the shipping cask were removed, the shipping cask was carefully raised to a vertical position on the rear cask support. Then, it was lifted from the ISO container and placed onto the base plate. The pressure in the cask cavity was equalized using a vent valve and then the closure lid was removed. During the first shipment and through visual inspection of the cask cavity, six empty baskets were found into it. These baskets were removed using the transfer cask and decontamination of the cask cavity took place.

A written authorization from the DOE was needed to initiate the loading of the spent fuel assemblies into the shipping cask. During the process for loading each fuel assembly, in accordance with the contract, a description of the observable physical condition was recorded. The results showed no visual evidence of corrosion, pitting or any other physical indication of damage of the authorized fuel assemblies.

Once all fuel assemblies were loaded into the shipping cask, the cask cavity was flooded with demineralized water to do radiological contamination surveys in accordance with the specifications giving in Appendix B. Three samples of water were taken, one at the beginning of the test, an intermediate sample at least four hours after commencement of the wait period but prior to completion of the eight hours, and the last sample after completion of the twelve hours. The samples were analyzed for radioactivity levels. In both shipments, the increase in ^{137}Cs activity levels for the twelve hours test were less than the maximum value specified for the NAC LWT shipping cask of 1 325 dmp/ml.

To remove the water from the shipping cask, pressurized air was blown into the cavity followed by vacuum dried process according to the operation procedure. Then, the cask cavity was filled with helium and the closure lid was leak tested.

After accomplishing the entire tests, the shipping cask was moved back to the container and the impact limiters were re-installed. When the container lid was installed it was sealed by an IAEA's inspector who verified the nuclear material during the loading. Finally, the health physics surveys and the shipping documents were completed. All the associated equipment used in the operation was packed back in the designated boxes in the same original configuration.

5.4. Transport to the port of San Antonio

The last operation was to transport the shipping cask from the RECH-1 reactor to the port of San Antonio located about 120 km from Santiago. The transport was done by using the highway from Santiago to San Antonio. The route was selected by the physical protection and risk prevention group with the advisory assistance of the Police. Alternative routes were considered in case of difficulties before or during the transport. The convoy was always protected by Special Group Police.

After receiving a written authorization from DOE for the shipment, the same night the convoy left the reactor towards San Antonio, reaching the port after a four hours journey. After arrival at the port the container-truck proceeded immediately to the pier where a vessel chartered for the operation was waiting to pick up the container. The transport operations of both shipments were developed in the same way; however, two different vessels were used. Both vessels arrived at the Charleston Naval Weapons Station in South Carolina, the first vessel with two shipping casks, one from Chile and another from Colombia, and the second vessel with one shipping cask from Chile and five from Argentina.

6. Final comments

It is essential for the reactor operator to identify the different local authorities who are responsible for the decisions on the diverse issues related to the operation. In some countries it is conceivable that the decision making process can involve many Government institutions and/or branches, causing compressible complications and delays.

An important issue is for the reactor operator to have a complete understanding of the rights and obligations pertaining to the original contract of the United States supplied enriched uranium.

Due to the diversity of tasks to undertake by the reactor operator it was found essential to work under a centralized and vertical organization scheme, with a general manager of the operation supported closely by the reactor manager. It is highly recommended for both managers to have a capability of communicating with their respective counterparts in the United States.

It is recommended for the reactor operator to have detailed documentation of the fuel assembly characteristics, specifications, drawings, irradiation history, water quality records among others.

Finally, it was found that the presence at the reactor site of an experience person from the cask owner company, before and during the operation, was very important. This was essential to resolve critical technical issues that arose and to facilitate the communication between the parties.

REFERENCES

[1] PEREDA, C., HENRIQUEZ, C., KLEIN, J., MEDEL, J., "Burn up Measurements of LEU Fuel for Short Cooling Times", Transactions 9th International Topical Meeting on Research Reactor Fuel Management, Budapest, Hungary, April 10-13 (2005) 240-244

[2] Norma CCHEN Nº 3.2.1 sobre el Control de Calidad del Agua Refrigerante del Reactor RECH-1, April 1984.

Experience on the return of spent TRIGA fuel from Germany to the United States of America at the example of the Medical University of Hannover

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Abstract. The Medical University of Hannover (MHH) participated in the US Department of Energy´s (DOE) "Research Reactor Spent Nuclear Fuel (RRSNF) Acceptance Program" in order to return its 76 spent TRIGA fuel elements back to the United States in the middle of 1999. The fuel elements have been moved to the Idaho National Engineering and Environmental Laboratory (INEEL) in Idaho.

This paper describes the experience with the procedure, the organizational and technical preparations for handling the fuel elements at the MHH and the various steps from the unloading of the core to the subsequent shipment and finally the storage at INEEL.

1. Introduction

The contract between the United States Department of Energy Idaho Operations Office and the MHH, Germany was discussed in 1997 and 1998 and signed on 1 July 1998 by both parties approximately one year prior to shipment.

In May 1998 all fuel elements of MHH were inspected by the Lockheed Martin Idaho Technologies Company (LMITCO) on behalf of DOE preparation for the return of the fuel elements to the United States. The results of the inspection showed that none of the fuel elements had any considerable damage and thus all of them were approved to be transferred to the INEEL storage facility. There were no defects of the fuel cladding. Special measures were taken for a total of 15 fuel elements to ensure that they could be safely stored at INEEL. For 10 fuel elements this measures were demanded by LMITCO. In addition the MHH decided to apply these measures for further 5 additional fuel elements.

The total number of spent TRIGA fuel elements was 76, 71 with aluminum cladding and 5 with stainless steel cladding. A total of 64 fuel elements were installed in the reactor core and 12 in the racks within the reactor tank. According to appendix A of the contract between MHH and DOE concerning the fuel return, all necessary data of the fuel elements were collected. Using the Origen 2.1 program, the relevant data for the fuel elements like U-content, burn-up etc. were determined by the first of January 1999 as shown in Table 1 below.

2. Organization of the project

Responsible for the return of the fuel elements was the owner of the reactor facility, the State of Lower Saxony, Germany, represented by the Ministry of Science and Culture. This department has delegated its responsibility to the MHH as the operator of the reactor facility, but was still responsible for the financial support. The organization of the whole procedure was done by the manager of the reactor with support in most cases of the company Babcock Noell GmbH. All items regarding any contracts regarding this project were negotiated by the lawyer of MHH. All work for removing the fuel were done by the following groups:

- The unloading of the reactor tank was done by the reactor staff of MHH.
- The subsequent fuel handling procedures in order to load the GNS 16 transport cask were carried out by the staff of the Babcock Noell GmbH.
- Sealing, preparation for shipment and the shipment of the transport cask were done by the consortium of the Nuclear Cargo + Service GmbH and the Gesellschaft für Nuklear Service GmbH (GNS).

The total number of permanent involved people for the fuel handling at MHH was 13.

TABLE 1. MAIN DATA FOR THE TRIGA FUEL ELEMENTS OF MHH CALCULATED BY THE PROGRAM ORIGEN 2.1

Total weight of U-235	2 776 g
Total weight of plutonium	8.85 g
Total activity	3.37×10^{13} Bq
Total decay heat	2.55 W
Average burn-up	6.31 MWd/kg
Maximal U-235 weight per fuel element	37.87 g

3. Consideration of technical and local structural conditions

Due to the location of the reactor on the ground-floor of the building of the radiological department and the clinic of Nuclear Medicine it was not possible to bring the GNS 16 transport cask into or close to the reactor room. The cask had to be loaded outside the reactor facility in a temporary building erected only for this purpose. Therefore it was necessary to remove the fuel elements from the reactor tank and load them first into a special transfer cask, which was then moved to the temporary building. In the temporary building the transfer cask was unloaded using the mobile reloading facility which had been developed for the removal of the fuel elements from German research reactors and first used at the research reactor of the VKTA Rossendorf near Dresden.

4. Fuel handling components

For carrying out all fuel handling procedures existing reactor equipment and special new components were used. Inside the reactor facility the existing components were only used for hoisting the fuel elements out of the reactor core and from the racks. For all the other handling inside and outside of the reactor facility new equipment was necessary. The loading of the GNS 16 transport cask was carried out with the mobile reloading facility, which had been borrowed from the VKTA Rossendorf. The GNS 16 transport cask was rented of the company GNS.

Most of the new components were specially developed and produced for use in the MHH in order to ensure safe fuel handling, e.g.:

- exclusion of load crash
- ensure criticality safety
- reliability of handling procedure
- minimize the personnel dose of the staff
- avoiding of contamination
- avoiding radioactive pollution to the environment

In order to handle the fuel elements loading units specially developed for MHH and made of an aluminum alloy were used. There were two types of loading units, one to accommodate 5 fuel elements (type B2) and one to accommodate 6 fuel elements (type A2). Compared to the type A2 the loading channels of the type B2 loading unit are approx. 2 mm larger in diameter in order to accommodate bent fuel elements and filter plugs at each end of a loading channel. The filter plugs are made of woven stainless steel with a mesh width of 100 µm and serve to hold back particles in accordance with INEEL requirements.

The TRIGA basket in the GNS 16 transport cask can accommodate a maximum of 15 loading units.

Criticality analyses were carried out for both the loading units and for a full basket, not only under normal handling conditions but also in emergency situations. The results showed that the configuration of the fuel elements remained below the critical level under all conditions.

5. Fuel handling procedure

Fuel handling was a completely dry loading procedure. In order to carry out this procedure it was necessary to load the fuel elements into the loading unit in a dry storage pit and to circulate dry air around the fuel elements in the loading unit before it could be loaded into the transport cask.

The whole procedure of the fuel handling was proofed in a dry run one month before unloading the fuel elements from the reactor core.

The following steps were carried out to unload and transfer the fuel elements from the reactor tank into the GNS 16 cask:

- **Unloading the fuel elements from the reactor tank and storing them in the loading unit**
  Each of the fuel elements was pulled out of the reactor core or the storage racks individually and pulled into the MHH transport flask which had been placed on a safety platform over the reactor tank. The flask was located over a dry storage pit next to the reactor tank which contained a loading unit in the bottom part. The fuel element was then lowered out of the MHH transport flask via the shutter device into the loading unit. The loading procedure was repeated until the respective loading unit was full of fuel elements and finally all of the fuel elements had been removed from the reactor tank. The channels of the type B2 loading units were sealed at the top and bottom with filter plugs since here the 15 fuel elements which needed special measures were placed.

- **Pulling the loading unit into the transfer cask**
  In order to load a full loading unit into the special transfer cask, the cask was set on top of the storage pit, as shown in Fig. 1. The loading unit was pulled into the special transfer cask.

- **Drying of the transfer cask**
  In order to remove any residual dampness from the fuel elements the transfer cask was hooked up to a drying apparatus. Any dampness was removed from the special transfer cask by means of dry air being circulated through a mobile filter unit, shown in Fig. 1, left. The air was then pumped into the reactor room spent air removal system.

- **Transfer of the transfer cask under a protection hood inside the MHH**
  After the drying procedure was finished, the transfer cask with a special hoisting mechanism was set in the transfer vehicle in order to move it to the temporary building. Then the transfer cask was covered with a protective hood and the radiological surveys were carried out on the transfer vehicle (Fig. 1, right). The transfer vehicle was moved along the transfer route between the reactor facility and the temporary building.

- **Loading of the GNS 16**
  The protective hood was removed from the transfer vehicle in the temporary building and the transfer cask was set on the mobile reloading facility on top of the GNS 16 transport cask (Fig. 2). The loading unit was then lowered into the appropriate loading position in the basket of the transport cask. After the loading unit was in position the transfer cask was removed from the

mobile reloading facility, set in the transfer vehicle and returned to the reactor facility. The transfer and loading procedure was repeated until all 14 loading units were positioned inside of the transport cask. Altogether there were 10 loading units of type A2 and 4 of type B2 loaded into the transport cask. Each working day one loading unit was loaded with fuel elements in the reactor room, taken to the temporary building and placed in the transport cask.

- **Checking and sealing of the GNS 16 for the shipment**
  The transport cask was then sealed and checked for tightness. After that it was made ready for shipment in accordance with transport requirements. Measurements of the dose rates and surface contaminations were done.

- **Shipment of the GNS 16**
  In the night of the 9^{th} of July 1999 the transport cask was then moved out of the temporary building by means of a mobile air cushion transport system and set in the 20-ft ISO-container of the transport vehicle by means of a 60-ton mobile crane (Fig. 3). After the ISO-container was made ready for shipment the transport cask was released for transport to the port of Esbjerg in Denmark.

Together with other five containers with spent fuel elements from European research reactors the ship arrived at the military port in Charleston / South Carolina. As soon as the container reached the U.S. ground the ownership of the fuel reverted from the MHH to the DOE. Then the fuel elements were transported by rail and truck from Charleston via Savanna River Site to the interim dry storage facility in INEEL. Hence the spent fuel has been ultimately disposed of for the MHH.

6. Milestones of use and return of the fuel elements

After 23 years of operation the MHH TRIGA reactor was finally shut down in December 1996. Since that time the measures for the return of the fuel elements to the United States were prepared. In Table 2 the milestones of use and return of the fuel elements are summarized.

After the removal of the spent nuclear fuel the preparation of the decommissioning of the reactor was started and will be finished next year.

TABLE 2. MILESTONES IN THE PROCESS OF REMOVING THE TRIGA FUEL ELEMENTS FROM MHH TO INEEL

Event	Time
First criticality of the TRIGA Hannover	31 January 1973
Final shut down	18 December 1996
Preparation of fuel handling according to German Atomic Law	August 1997 to July 1998
Dry run of fuel handling at MHH	6 days in April and May 1999
DOE authorization to ship	14 May 1999
Permit of the local authority for fuel handling	7 June 1999
Start unloading fuel elements	9 June 1999
Finish loading of GNS 16 transport cask	28 June 1999
Preparation for shipment and sealing of GNS 16	6 July 1999
Start of shipment at MHH	9 July 1999
Pass of ownership of fuel elements from the MHH to DOE	19 August 1999
Finish unloading of GNS 16 at INEEL	9 September 1999

7. Conclusions

The loading and transfer technology used offers the following advantage of dry spent fuel handling:

- Optimal conditions with respect to radiation safety
- Consideration of the technical and local structural conditions at MHH

- Consideration of strict basic design criteria for the new components, which allowed to satisfy the tight schedule
- Daily routine work at MHH not effected

Altogether the procedure chosen for removing the TRIGA fuel proved to be the optimal solution under the existing conditions at the MHH. The handling equipment could be used also for unloading fuel elements at other TRIGA reactors.

FIG. 1. Transfer cask for TRIGA fuel elements on top of the storage pit with the drying mechanism (left). The transfer cask was covered with a protective hood (right) and carried out of the reactor facility after the radiological surveys.

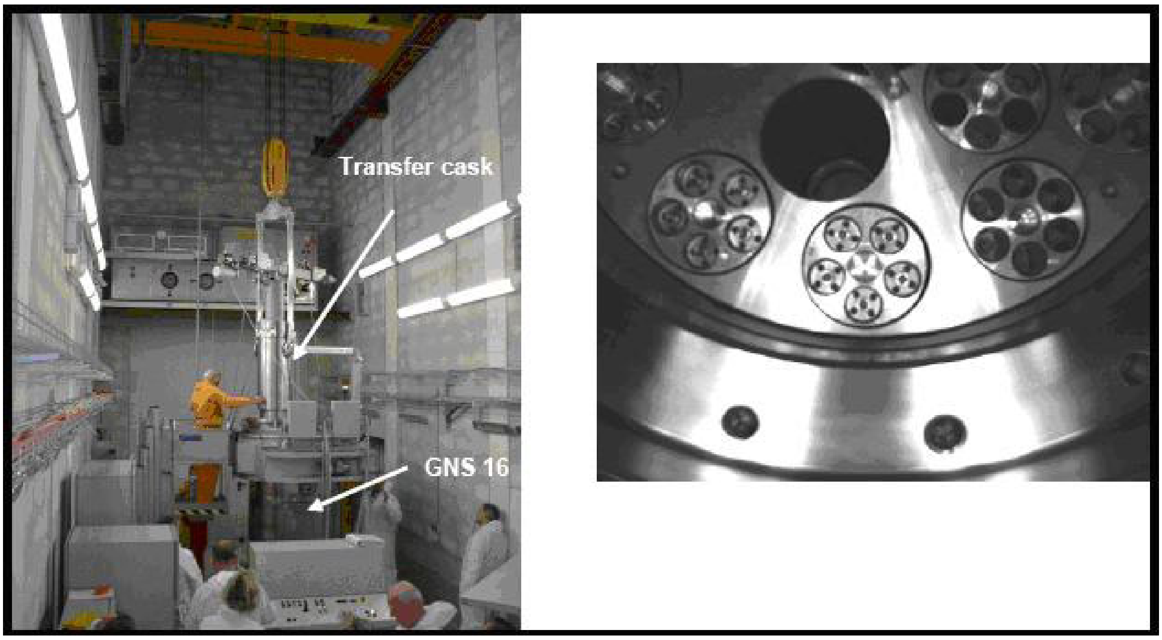

FIG. 2. View in the temporary building with the transfer cask mounted on top of the GNS 16 transport cask to lower the loading unit into the basket of the GNS 16 (left). View on the top of the basket with loading units for 5 fuel elements closed with filter plugs and for 6 fuel elements (right).

FIG. 3. The GNS 16 transport cask was moved out of the building using an air cushion transport system and measurements of the dose rates and surface contaminations were done (left). The GNS 16 was set in the 20-ft ISO-container of the transport vehicle by means of a 60-ton mobile crane (right).

Shipment of spent nuclear fuel from Hahn-Meitner-Institute Berlin to the country of origin: *A successful venture*

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Abstract. In 1996 the Hahn-Meitner-Institute Berlin signed an outline agreement with the Department of Energy for the return of spent nuclear fuel of the research reactor BER II to the USA, which expired in 2006, but could be prolonged in 2005 for another 10 years. Conditional for the first agreement was the conversion from HEU- to LEU-fuel, which was done between 1997 and 2000.

The Hahn-Meitner-Institute has already shipped 145 spent nuclear fuel elements in four campaigns to Savannah River Site (SC, USA). For each shipment many of coordination and paperwork had to be done. But because of the good cooperation with the Department of Energy everything went smoothly and could be achieved in a good time. This report is about main work, which had to be done and problems, which had to be overcome.

1. Introduction

The **H**ahn-**M**eitner-**I**nstitut Berlin (HMI) operates the research reactor BER II (**B**erliner-**E**xperimentier-**R**eaktor) for neutron scattering. BER II, shown in Figs 1 and 2, is a pool type reactor with a thermal power of 10 MW and loaded with MTR type fuel elements. The core is surrounded by beryllium that works as reflector.

The main purpose of the reactor is to be utilized in the field of neutron scattering research. Therefore, there are 9 beam tubes supplying thermal neutrons to the experiments in the experimental hall. Furthermore there is a special beam tube containing a cold neutron source. Six neutron guides are supplied with cold neutrons from this source. These guides extend to experiments in the neutron hall.

Although the main purpose of our reactor is neutron scattering we have three irradiation devices. These are placed in the core and in the beryllium reflector. They are loaded and reloaded from the reactor hall.

The BER II first went into operation in 1973 with a thermal power of 5 MW. From 1985 until 1991 we upgraded our reactor to increase the neutron flux at the beam tubes.

This was done by

- increasing the power from 5 to 10 MW
- reducing the core size
- installing a beryllium reflector

Furthermore we installed a cold neutron source.

In 1991 we restarted the reactor and since then the BER II is routinely in operation.

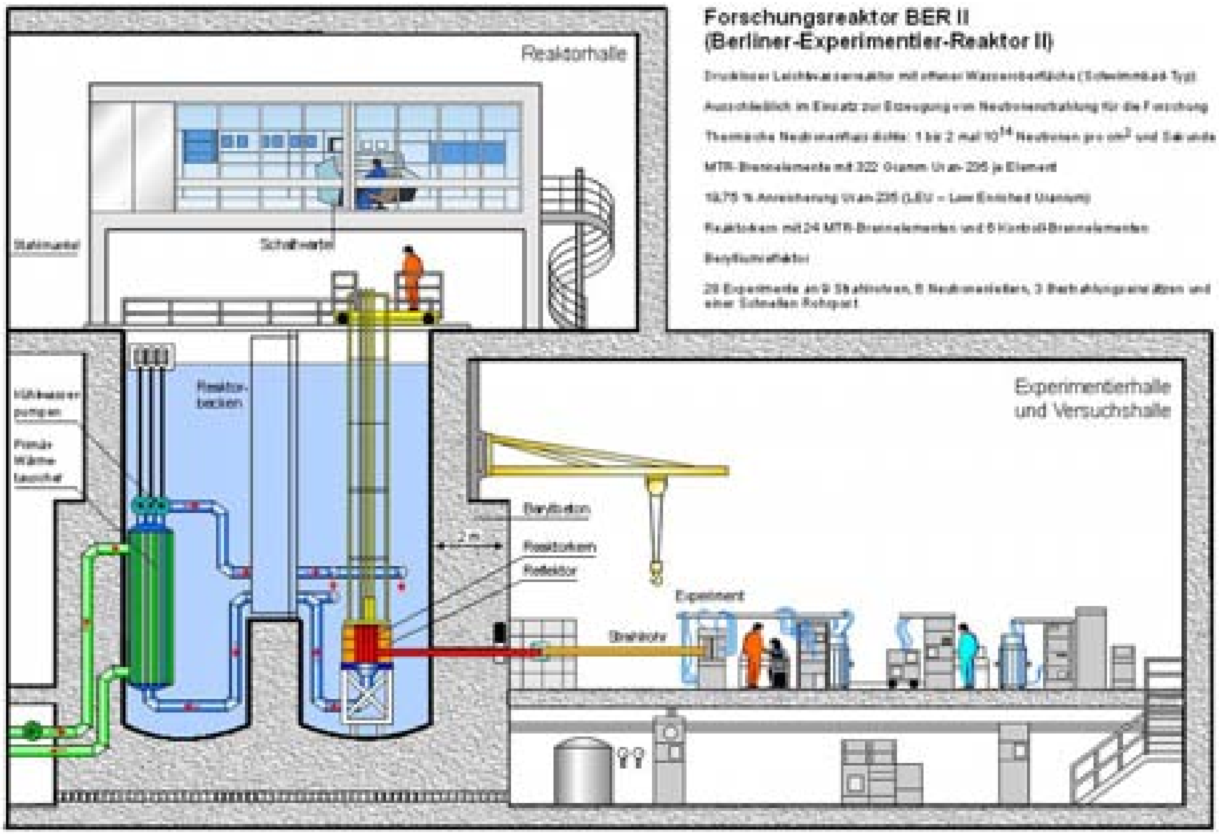

FIG. 1. Research Reactor BER II.

FIG. 2. Overview of the BER II reactor building, showing the experimental hall.

At the end of the 1980s the USA discontinued the original take back programme. This fact caused many problems in getting an operating licence for the just upgraded research reactor in our institute. Because the licence must provide evidence that a valid prevention of spent nuclear fuel disposal exists. This obligation must be verified every year for six years in advance to the atomic authorities by the HMI. In this way, we found a temporary solution for this disposal problem. In the period from 1993 to 1995 the HMI shipped 78 HEU-spent nuclear fuel elements for reprocessing to Dounreay, UK. This could be done according to the "Exchange of Notes" between the Governments of the United Kingdom and Germany in advance. The waste from these campaigns must be returned to Germany in a designated period. This affected other German reactor stations also and is still in progress.

2. Arrangements

After 1995 the USA announced the willingness to take back the spent nuclear fuel in the context of "Non-Proliferation of Weapon Grading Material" for the period from May 1996 to May 2006, with the condition that the uranium must be of US-origin.

An essential condition for opening this disposal route was the obligation for the HMI to convert its research reactor from HEU- to LEU-fuel elements. This was carried out from September 1997 to March 2000 resulting in a build up of eleven mixed cores. Each new single core was loaded with the same amount of fresh LEU-fuel in relation to the reloaded amount of HEU-fuel.

The conversion was done without difficulties and there were only minor changes in neutron flux and operation procedures of the reactor.

In August 1996, the HMI signed an outline agreement which was made by US-Department of Energy (DOE). Herein, the HMI has assured a disposal route for a 10-year period.

3. Shipments

According to the outline agreement the HMI has to meet a special requirement for every single shipment with DOE. In particular, the first shipment with 66 spent nuclear fuel elements took a relatively long time in preparation for the campaign. It started with the handover of copies of the Fuel Element Documentation to DOE. Additionally we had to answer the requests given in the Appendix A "Spent Nuclear Fuel Acceptance Criteria" for standard elements as well as for control elements. This Appendix provides a detailed description of the material to be delivered to DOE in accordance with the contract and also enumerates the specifications and requirements which the customer must meet.

The main topics are:

- Material Description
- Fuel Assembly Description
- Fuel Identification
- Fuel Irradiation Specifications – History and Post Irradiation
- Cask and Basket Identification

After this procedure was completed the DOE had many questions for detailed positions and in a cooperative way all wants of clearness could be clarified in a satisfactory manner. In addition to this task two DOE-specialists came to the HMI and made a visual inspection of all 66 elements. The main focus was to search for corrosion on the elements. As a result of this inspection, all detected elements were free of corrosion.

A further DOE condition is defined in Appendix B "Transport Package (Cask) Acceptance Criteria". When the cask had been loaded, samples had been taken for the radionuclide inventory measurements as well as for the internal/external contamination of the cask, which were than proceeded to DOE.

In March 1999 we made the second shipment with 29 HEU-spent nuclear fuel elements and in September 2000 the third shipment followed with 17 HEU-spent nuclear fuel elements. In preparation of both shipments we had to do the same activities as described for the first shipment. However our experiences in this matter have matured in the meantime and we did not need that much time anymore, although the amount of paperwork did not decline. In appreciation of the good results according to the visual inspection of the elements for the first shipment, DOE waived this procedure for the last two shipments.

With the three shipments a total of 112 HEU-spent nuclear fuel elements were shipped to the USA and with the last shipment from the HMI there is no longer any Highly Enriched Uranium (HEU) fuel left.

The outline agreement guarantees also shipments with low enriched spent nuclear fuel to DOE under the same conditions given for the HEU elements. The HMI has done such a shipment in July 2004 with 33 elements. According to the "Forecast of Shipment" the DOE has scheduled the last HMI-shipment for June of 2008. When this is done, the period of validity of the outline agreement will then expire for our institute.

It is important to mention that beside both partners, HMI and DOE, a third partner was needed, a shipping company. The HMI is working for a long time together with the Nuclear Cargo+Service GmbH (NCS) in good cooperation for all kinds of fuel shipments. Figure 3 shows the links between the three partenrs, HMI, DOE and NCS.

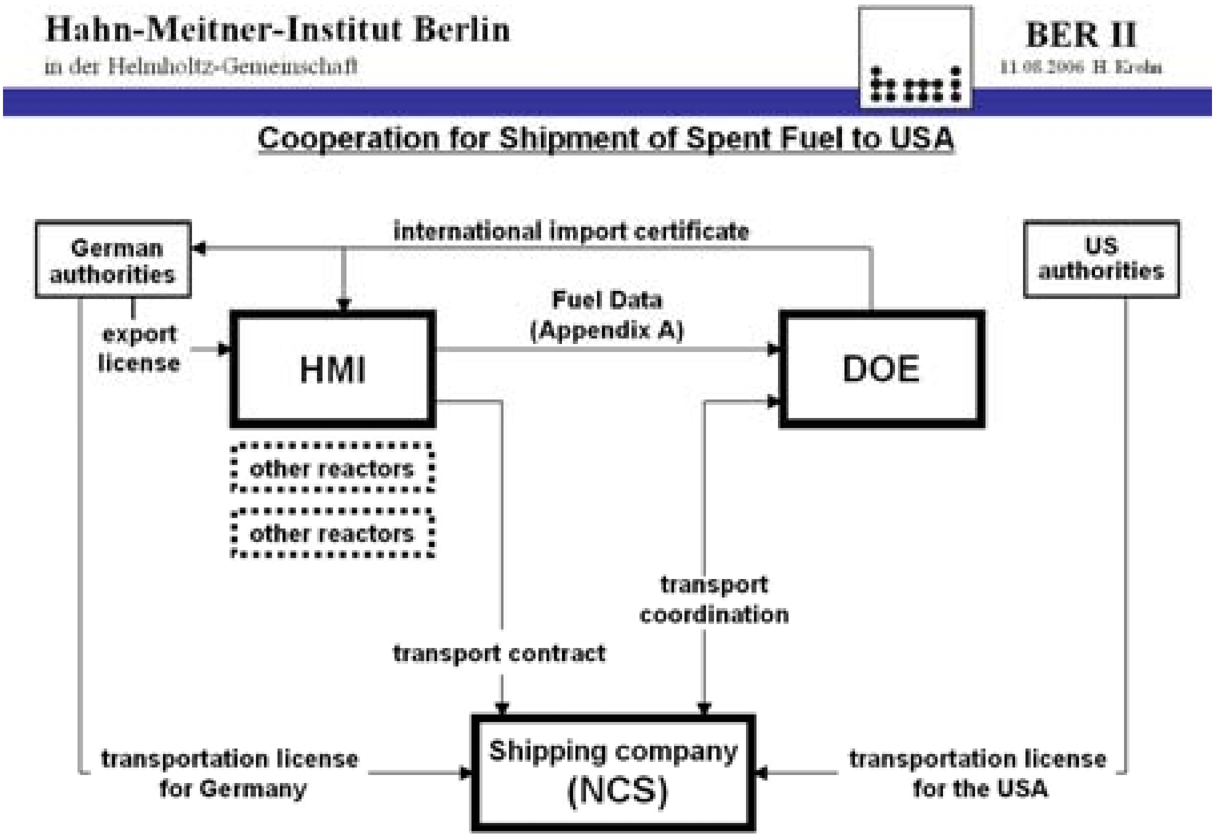

FIG. 3. Links between all three partners HMI, DOE and NCS.

Parallel to the activities between HMI and DOE, the management between HMI and NCS concerning the delivery of the empty cask to the HMI, the loading assistance and the after treatment of the cask has been done between both partners by working hand in hand.

Each of the four shipments was integrated in an European assembly campaign which was managed by NCS. The positive effect of such logistic was a reduction in transport costs for each involved reactor station.

4. Outlook

Already during the running time of this agreement for returning the spent nuclear fuel to the USA the so called “Edlow Group” a lobby with common interests of operating reactor stations took an initiative for a prolongation of this acceptance program for further 10 years. This is essential to operate the research reactors for a longer time. The US-authority has agreed among other reasons to the arguments of the “Edlow Group” and favoured an additional 10-year prolongation at the end of 2004.

In February 2006 the HMI has signed a new outline agreement, provided by DOE. As a result the HMI has secured, that spent nuclear fuel from the BER II, which will be unloaded until May 2016, can be shipped to the USA.

5. Conclusion

This acceptance program gives the guarantee for the HMI for a safe and reliable disposal route for a longer period.

So in summary it can be said, that all four shipments of spent nuclear fuel to the USA run smoothly due to the good cooperation with the experts from DOE. The prolongation of this acceptance program gives us the chance to continue this good teamwork and allows us to dispose our spent nuclear fuel up to 2016. Afterwards we will have to rely on a German solution to dispose our waste because we plan to operate the research reactor BER II for quite a while after this date.

Experience in shipment of spent nuclear fuel from GKSS to the United States of America under the FRRSNF acceptance programme

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Abstract. Already in the 1980' GKSS-Research-Center decided to terminate operation of its two research reactors FRG-1 and FRG-2 on the HEU fuel cycle. This decision was made in acknowledgement of the aims of RERTR program, in which GKSS was a strong supporter from the beginning. The FRG-2 (15 MW) has been used as Germany's largest material testing reactor for power reactor materials and safety test and has played an important rule in the conversion activities at the GKSS research centre. The FRG-2 was shutdown in 1991, therefore the conversion to LEU was not considered reasonable. The FRG-1 (5MW) has been converted in February 1991 from HEU (93%) to LEU (20%) in one step. Due to the unforeseeable termination of the take back policy of US-Doe in 1989 for spent HEU fuel element, there were from the former operation of the two reactors a large number of spent fuel elements to be stored at the reactor site. In 1996 the US initiated the Foreign Research Reactor Spent Nuclear Fuel acceptance program. Under this program GKSS has done 6 shipments with 141 spent HEU fuel elements and 150 spent LEU fuel elements to the US DOE Savannah River.

In December 2005 the US-DOE and GKSS signed the contract "TERMS AND CONDITIONS FOR THE ACCEPTANCE OF FOREIGN RESEARCH REACTOR SPENT NUCLEAR FUEL AT THE SAVANNAH RIVER SITE" for the extension up to 2016.

1. Introduction

Two research reactors have been operated very successfully by the GKSS-Research-Centre over decades in a large connected pool system:

The FRG-2 (15 MW), criticality March 1963, was scheduled to shut down in 1991 for lack of scientific and technical interest for future use. The reactor has been used as Germany's largest material testing reactor for power reactor fuel and power reactor materials development and safety tests. The FRG-2 has played an important role in the conversion activities at the GKSS-Research-Centre.

The research reactor FRG-1 has been originally designed and constructed in 1957/1958 (criticality on 23 October 1958) to serve general scientific research needs in different aspects of fundamental research and some applied research like cracking phenomena of organic coolants and isotope production.

It is clear that during the lifetime of the research reactor the research areas have been changed more than once. The outcome of such changes results on the one side in new experimental facilities at the beam tubes and on the other side in design changes at the reactor.

The following design changes have been made: increase of fuel loading, increase of burn up, reduction of enrichment, reduction of core size, new control rods, installation of a cold neutron source. At present the FRG-1 is being used with high availability for beam tube experiments for fundamental and applied research in biology, materials research, neutron radiography, neutron activation analyses etc. The FRG-1 has been converted in February 1991 from HEU (93%) to LEU (20%) in one step and at that time the core size was reduced from 49 to 26 fuel elements. Consequently the thermal neutron flux in beam tube positions could be increased by more than a factor of two [1][2]. It is the strong intention of GKSS to continue the operation of the FRG-1 research reactor with high availability and utilization. The reactor has been operated during the last years for approximately 250 full power days per year. To prepare the FRG-1 for an efficient future use, the core size has been reduced in March 2000 in a second step from 26 fuel elements to 12 fuel elements. For this purpose the U-235 density has to be increased from 3.7 g U/cc to 4.8 g U/cc. So that finally the size of the reactor is being reduced from 49 fuel elements to 12 fuel elements over the last 10 years [3][4]. The model of the FRG-1 compact core with the beryllium reflector and the beam tubes is shown in Fig. 1.

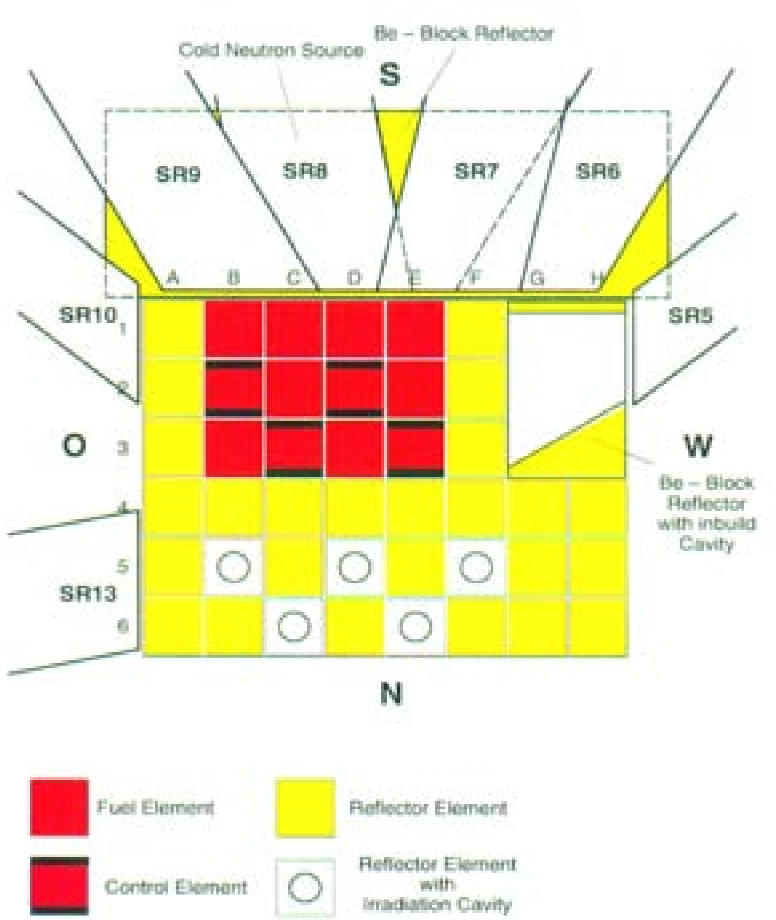

FIG. 1. Model of the 3x4 core with beryllium reflector and beam tubes.

The fuel elements for the operation of the FRG-1 are manufactured by CERCA using U.S. origin enriched uranium. Figures 2a and 2b show the cross section of the fuel element.

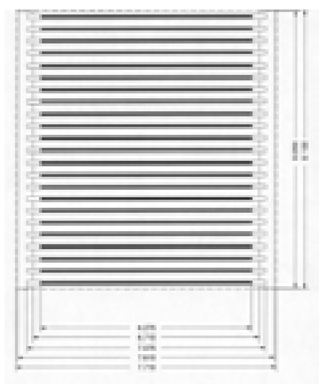

FIG. 2a. Standard fuel element.

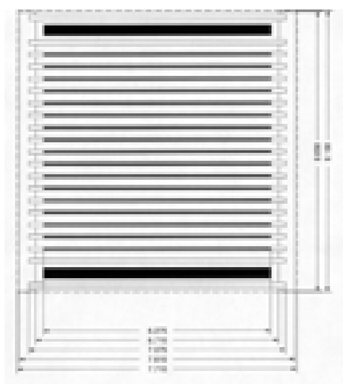

FIG. 2b. Control fuel element.

2. Problems at the back end of the fuel cycle

Due to the unforeseeable termination of the take back policy of US-DOE in 1989 for spent HEU fuel element there were from the former operation of the two reactors a large number of spent fuel elements to be stored at the reactor site. The large numbers of spent fuel is a severe problem as long the storage capacity is limited and an increase of storage capacity causes in Germany many licensing problems.

3. Shipments under the FRRSNF Acceptance Programme

On 13 May 1996, the U.S. Department of Energy DOE announced the Record of Decision for the Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Police Concerning Foreign Research Reactor Spent Nuclear Fuel. Based on this policy, the U.S. DOE will accept and manage in the U.S. foreign research reactor spent nuclear fuel that was originally enriched in the U.S. Under the terms of the policy, aluminum based spent fuel will be accepted at the Savannah River Site of DOE.

A contract for the receipt of spent nuclear fuel elements was signed with the U.S. DOE as well as a contract with the Consortium NCS/GNS for the transport organization. The GNS-11 and the GNS-16 casks were used for the shipment. They have a maximum transport capacity of 33 MTR fuel elements. The maximum cask weight without shock absorber is 11.5 t for the GNS-11 and 13.2 t for the GNS-16. Due to the relatively small weight, this cask can be used in our facility. Both casks are transported in 20’ Open hard-Top Container. The transport casks GNS-11 and GNS-16 have been approved internationally for use in the shipment of spent nuclear fuel from foreign research reactors to Savannah River Site.

For each shipment and each type of fuel GKSS prepared a spent nuclear fuel acceptance criteria document. The Appendix A Agreement identify and categorize the spent nuclear fuel with the following information:

- Physical Dimensions
- Material Description
- Summary of Irradiation History
- Fissile Content
- Other Residual Activities
- Decay Heat
- Activity and Dose Rate
- Additional Information

For the preparation and shipment of spent nuclear fuel GKSS has prepared a detailed step by step plan. This plan has been approved by the authorities and make sure that all steps in the process are performed safely and properly. This step by step plan includes

- Preparation of the facility
- Receipt of the transport cask
- Movement of the cask into reactor hall
- Preparation of the fuel elements
- Loading of the fuel into the cask
- Closure, sealing and testing of the cask
- Preparation for shipment
- Movement of the cask out of the reactor hall
- Loading of the cask onto the transport vehicle
- Radiation protection
- Shipment

For the receipt of the transport container and the movement of the cask into the reactor hall we needed a mobile crane outside of the reactor facility (see Fig. 3). The preparation of the cask and the fuel elements in the storage pool is shown in Figs. 4 and Fig. 5 shows the loading of the GNS-16 cask with the fuel elements.

FIG. 3. Transport container and mobile crane outside of the reactor facility.

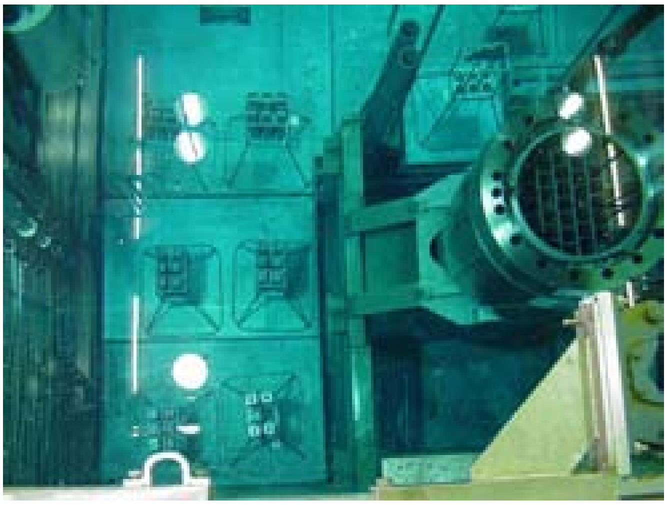

FIG. 4. Preparation of the cask and the fuel elements in the storage pool.

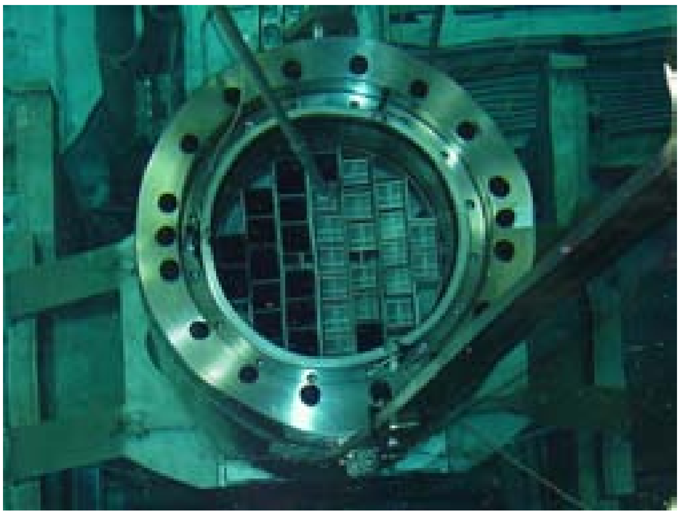

FIG. 5. Loading of the GNS-16 cask with fuel elements .

The shipment of irradiated fuel is a highly regulated activity requiring extensive coordination between the reactor facility, the transport company, international, national and local governments, port authorities and the receiving facility. As a result of the good cooperation with the DOE and with the Consortium NCS/GNS GKSS have done the following 6 shipments to Savannah River Site:

- August 1996, 33 HEU fuel elements in one GNS-11 cask
- March 1997, 66 HEU fuel elements in two GNS-11 casks
- July 1997, 33 HEU fuel elements and 26 LEU fuel elements in two GNS-11 casks
- September 2000, 9 HEU fuel elements and 24 LEU fuel elements in one GNS-16 cask
- September 2001, 66 LEU fuel elements in two GNS-16 casks
- July 2004, 33 LEU fuel elements in one GNS-16 cask

4. Conclusion

Since 1996 GKSS has shipped 141 HEU fuel elements and 150 LEU fuel elements to the Savannah River Site without any problems. After the extension of the FRRSNF Acceptance Program up to 2016 the US-DOE and GKSS signed the contract “TERMS AND CONDITIONS FOR THE ACCEPTANCE OF FOREIGN RESEARCH REACTOR SPENT NUCLEAR FUEL AT THE SAVANNAH RIVER SITE” in December 2005.

REFERENCES

[1] KRULL, W., Enrichment reduction of the FRG-1 research reactor, Proc. Int. Mtg on Reduced Enrichment for Research and Test Reactors (RERTR), Jakarta, November 4-7, 1991.

[2] KRULL W., JAGER, W., Status and perspectives of FRG-1 after conversion to LEU, Proc. Int. Mtg. on Reduced Enrichment for Research and Test Reactors (RERTR), Seoul, October 6-10, 1996, Published by the Korea Atomic Energy Research Institute (1996) 381-388.

[3] SCHREINER, P., KRULL, W., HANAN, N.A., DEEN, J.R., Increasing the neutron flux after reduction of the core size of the FRG-1, Proc. Int. Mtg on Reduced Enrichment for Research and Test Reactors (RERTR), Jackson Hole, Wyoming USA, October 5 - 10, 1997.

[4] SCHREINER, P., KRULL, W., FELTES, W., Increasing the neutron flux at the beam tube position of the FRG-1, Mtg. Int. Group on Research Reactor (IGORR 6), Taejon, The Republic of Korea, April 29 - May 1, 1998, KAERI/GP-128/98, Korea Atomic Energy Research Institute (1998) 31-37.

RRSNF Shipment Operation of Indonesia Research Reactors

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Abstract. In the beginning of the year 2004, reexport of spent nuclear fuel (SNF) of three Indonesian reactors to the origin country under "US FRRSNF acceptance programme" was successfully completed. The TRIGA and MTR type of SNF were sent back to INEEL, Idaho and SRS, Savanah River, USA respectively. The activities took about 6 months of coordination works from starting until loading the SNF onto the ship in the harbor. Two harbors were chosen to upload the SNF i.e. Cigading Port, nearby Jakarta for SNF from RSG-GAS and TRIGA-2000 reactors and Cilacap Port in southern part of Central Java for the SNF for Kartini reactor. A National Team was established to coordinate the whole operation. The report covers aspects of management, preparation works, loading works and transport operation.

1. Introduction

The objectives of the report is to illustrate the entire activities in implementing return, or re-export, of spent nuclear fuel (SNF) from the three Indonesian research reactors to United States of America as the country of origin in 2004.

As known, since May 1996 United State Department of Energy (US DOE) through Record of Decision (ROD) for the Final Environmental Impact Statement (FEIS) on a Proposed Nuclear Weapon Nonproliferation Policy opened opportunity to all research reactor owners to send back their SNF and uranium target of US Origin to America for period of time 10 years. BATAN took this opportunity for the first time by shiping 47 fuel elements and 1 plate of MTR type SNF in March 1999 from RSG-GAS reactor, under Contract No. DE-G109-99-SR18920. The second opportunity was proposed by BATAN for re-export SNF from three reactors to US-DOE in the beginning of 2003. The green light was then shown in November 2003 by visits of US-DOE Team to Jakarta, Indonesia. The Team was coming from US-DOE, SRS-USDOE, INEEL-USDOE, US Embassy in Jakarta, NAC and a Security Consultant.

Negotiation between Indonesian and United States Team in Jakarta during their visit were essential to discuss and coordinate all activities needed to implement the SNF take back program of the three reactors, such as contract preparation, fuel identification, licensing, security, transport route and Ports as well as schedule of activity. Visit to reactor site, Ports and roads were also conducted.

Considering the politic situation in Indonesia at that time, the parties agreed to accomplish the shipment before start the campaign phase for presidential election on 11 March 2004, although the remaining time was very short. This challenge was compensated by the parties by working hard and with good coordination and communication. The first experience of SNF re-export contributed much in emerging this spirit.

The previous experience in all activities of SNF Shipment Operation comprising preparation of technical and administrative documents to fulfill the contract; move the SNF from storage pool to interim storage; cropping SNF; loading to transfer cask; transport to port, arranging permit and or license; and organization of radiation monitoring activities, was a determinant factor to assure that the operation could be terminated in time.

2. Reactor core and fuel

National Nuclear Energy Agency Indonesia, BATAN owns three research reactors located in three separate regions as main tools for conducting the R&D in nuclear science and technology. TRIGA-2000 reactor with the power of 2 MW is operated since 1964 by the Center for Technology of Nuclear Material and Radiometry (ex. P3TkN- BATAN) in Bandung, West Jaw. The reactor is originally TRIGA MARK II type, 250 kW and then upgraded to 1MW by replacing the core and lately it was upgraded again to 2 MW in the year of 2000. The second reactor is KARTINI reactor, 100 KW which is operated since 1979 by Center for Technology of Accelerator and Material Process (ex. P3TM-BATAN) located at Yogyakarta. The reactor core was built using the core of the ex- TRIGA MARK-II Bandung. The last is G.A. Siwabessy (RSG-GAS) multipurpose reactor, 30 MW operated by Center for Multipurpose Reactor (ex. P2TRR-BATAN) since the year 1987, at Serpong Center, Banten Province.

2.1. RSG-GAS reactor

The reactor is an open-pool type, cooled and moderated with light water, using the LEU-MTR type fuel element in the form of U_3O_8-Al dispersion. Since 1998 the oxide fuel element was gradually changed to the silicide (U_3Si_2-Al) type using the same physical dimension as well as uranium density. The reactor core is 60-cm height with rectangular cross section, located in the reactor pool 12.45 m under pool surface. The core configuration is arranged in a 10 x 10 array grid and it consists of 40 standard fuel elements, 8 control elements, 8 AgInCd control absorbers; 1 CIP (Central Irradiation Position), 4 IP (irradiation positions), 5 rabbit systems and other irradiation holes in the reflector region. The reactor core is surrounded by beryllium elements and a beryllium block reflector, as shown in Fig. 1. The figure shows also out-core irradiation facilities comprising Power Ramp Test Facility, Neutron Radiography and Silicon Doping Facility as well as Beam tubes. The reactor is operated at power level of 15 MW for 4 cycles a year. Each cycle consumes 6 fuel elements for about 600 MWD, resulting a consumption of 24 spent nuclear fuel elements per year. The discharge burn up is 56% in average. The main characteristics of the fuel elements are shown in Table 1.

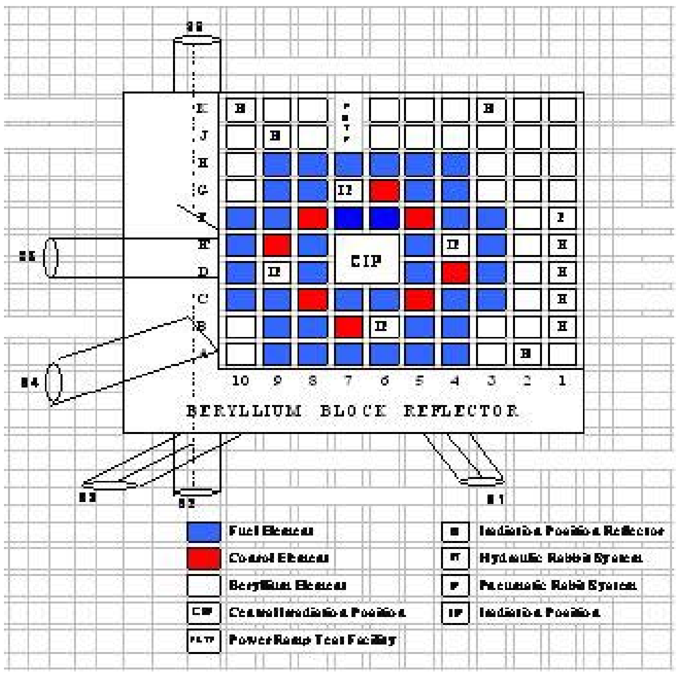

FIG. 1. RSG-GAS core configuration.

The facility has a interim storage for spent fuel facility (ISFSF), outside of the reactor building, which is used interim wet storage of the spent fuel. The spent nuclear fuel elements are transferred to the spent fuel pool through a transfer channel, as shown in Fig. 2.

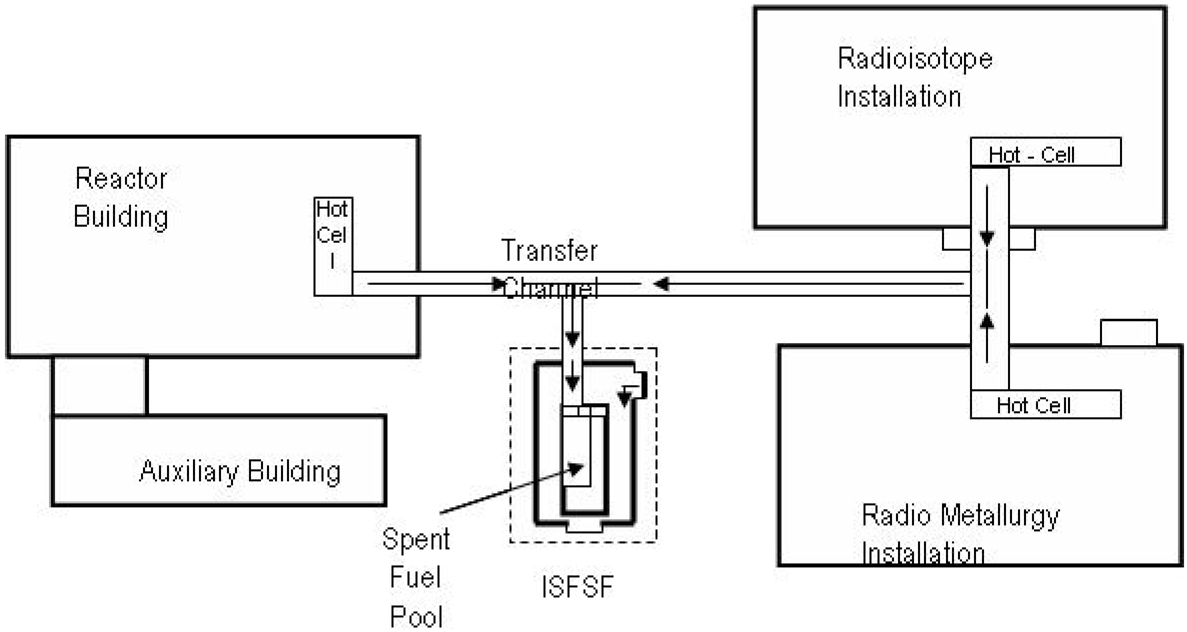

FIG. 2. Interim storage for spent fuel (ISFSF).

TABLE 1. FUEL ELEMENT SPECIFICATION OF THE RSG-GAS REACTOR

Dimension (mm)	77.1x81x 600
No. Plate of Fuel Element (FE)	21
No. Plate of Control Element (CE)	15
Clad material	$AlMg_2$
Clad thickness, (mm)	0.38
Meat dimension, (mm)	0.54 x 62.75 x 600
Meat material	U_2Si_3Al
U-235 Enrichment, (w/o)	19.75
Uranium density, (g/cm^3)	2.96
U-235 weight per FE, (g)	250
U-235 weight per CE, (g)	178.6

2.2. TRIGA Reactor

Both TRIGA reactors, KARTINI and TRIGA-2000 have similar building and core as illustrated in Fig. 3. The spent fuels are stored under water in the bulk shielding pool. The water quality is maintained to be around 6.8 of pH and 0.5 μ mho. KARTINI reactor core is placed under about six meters of light water. The core consists of five rings (B to F) which can accommodate 90 holes of fuel elements. Actually the reactor core contains 69 fuel elements divided in 104 SS type fuels (67) and IFE 204 SS type fuels (2). The core has also 15 dummy (graphite) fuel element, 2 pneumatic tubes, AmBe as neutron source and 3 control rods. The reactor core is cylindrical in shape with a 45 cm diameter and active fuel length of 38 cm. The 104 SS type of fuel-moderator elements consist of a homogeneous mixture of uranium-zirconium hydride in which the H-to-Zr atom ratio is 1.7 to 1, with 20 % U-235 enrichment.

TRIGA-2000 reactor uses 3 types of TRIGA fuel i.e. Standard-8.5 (catalog no 104), Standard-12 (catalog no 106) and Standard-20 (catalog no 118). The core contains 116 fuel elements. The TRIGA fuel specification data is shown in Table 2.

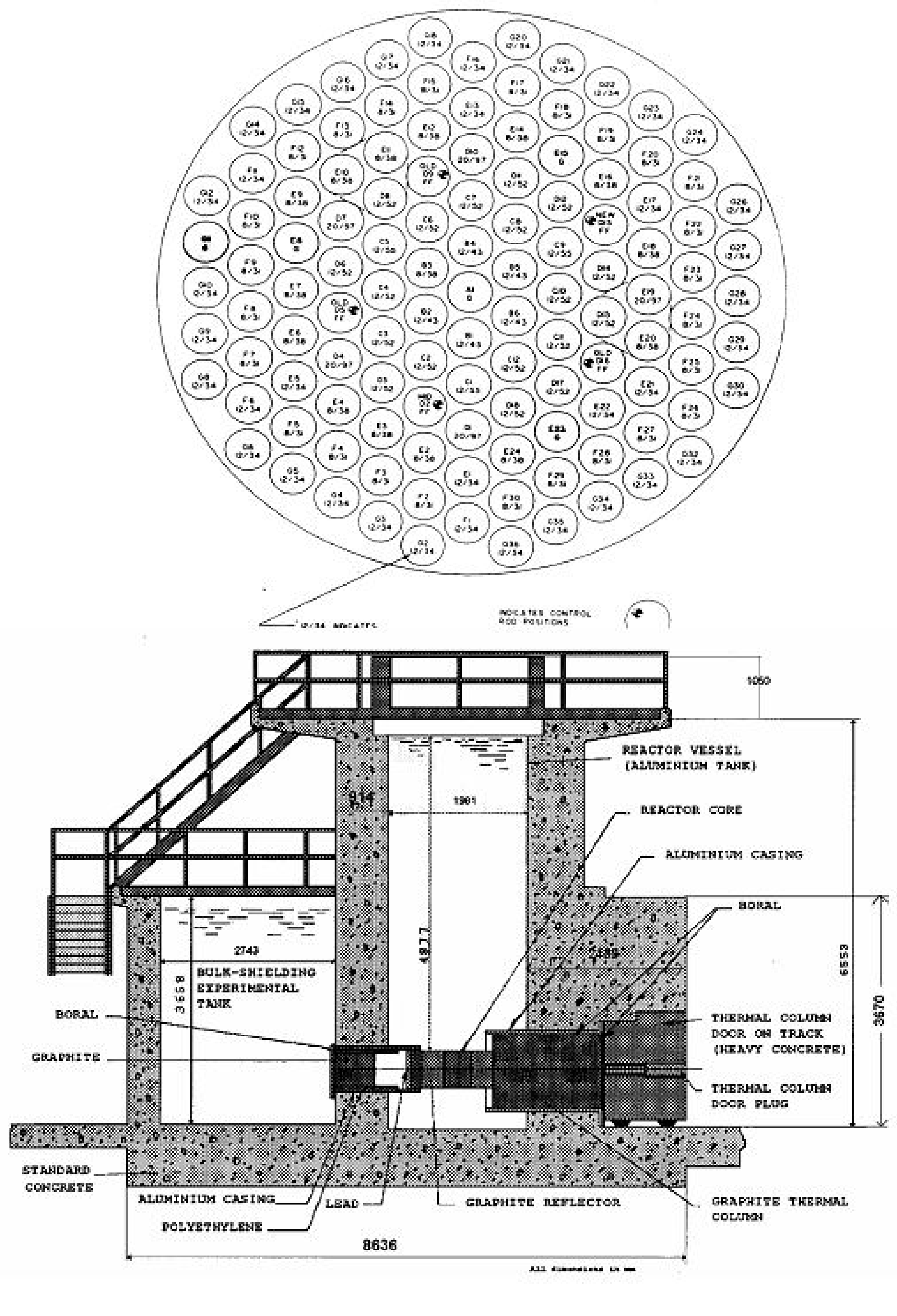

FIG. 3. Core grid plate and pool of TRIGA reactor.

USA Party:

- US Department of Energy (US DOE): take back SNF from reactor site to SRS/INEEL, USA.
- US Embassy in Jakarta: support financial for Security in Indonesia.

Supported by:

US Contractors:
NAC International: handling and loading SNF to TC and transportation the SNF.
P.T. Easternindo Carmitra Lintas (PT. ECL): supporting tools, equipment and vehicle loading and transport operations.

3.3. Coordination Activities

A National Team of BATAN was set up by the Chairman of BATAN wwith the task to coordinate, implement and control all FRRSNF shipment activities of the three reactors, RSG-GAS, TRIGA 2000 and Kartini, covering contracting, licensing, preparation works, scheduling, safety and security.

Internal BATAN coordination started on August of 2003 to prepare both technical, contractual and management aspects.

External coordination meetings were frequently conducted to realize and review security plan, licensing and surveys. The main activities involved:

Scheduling: A continuous coordination activity schedule was developed and periodically kept updated as shown in Table 3.

Licensing: License documents needed for conducting the shipment were:

(a) Safeguards report
(b) Fuel transportation permit
(c) Arrival and re-export permit of equipment/tools
(d) SNF re-export permit
(e) Permit for SNF Transportation to Port.
(f) Certification of the Transport Cask

Procedures: Some procedures had to be prepared in order to obtain the license or permit, as follows::

- SNF Loading procedures
- Procedures for Fuel element cropping
- SNF transport procedure
- Emergency response during transportation
- Physical protection during SNF transportation
- Safety Analysis Report of SNF Transportation from reactor to Port.
- Radiation protection procedures

Security Plan: Coordination was focused on security plan for land and sea transportation of SNF. On land security it was coordinated by Regional Police of Banten, Jaw Barat, DKI, Yogyakarta and JawaTengah provinces. Security for sea transportation was coordinated by RI Navy.

All parties, including Police, Army, Navy, BAPETEN, US-DOE, US embassy as well as forwarder PT ECL, agreed that the implementation of the SNF shipment operation should be finished before the start of the Election Campaign, i.e. 11 March 2004.

TABLE 3. REVISED SCHEDULE OF SNF SHIPMENT OPERATION

No.	Activities	February 2004											March 2004											
		19	20	21	22	23	24	25	26	27	28	29	1	2	3	4	5	6	7	8	9	10	11	12
0	**Vess approx. Indonesia Territory**																							
1	**Vessel arrive at Jkt Port**																							
	a. Container to Serpong																							
	b. Container to Yogyakarta																							
	c. Container to Bandung																							
2	**SERPONG SITE**																							
	a. Cropping Operations (2 days)																							
	b. Equipment unpack, Fuel loading in																							
	Cask testing,equipm pack (9 days)																							
	c. Cask sent to Cigading																							
3	**BANDUNG SITE**																							
	a. Equipment unpack, Fuel loading in																							
	Cask testing,equipm pack (5 days)																							
	b. Cask sent to Serpong																							
	c. Tools sent to Yogya																							
4	**YOGYAKARTA SITE**																							
	a. Equipment unpack, Fuel loading in																							
	Cask testing,equipm pack (5 days)																							
	b. Cask sent to Cilacap																							
5	**SNF SHIPMENT**																							
	a. Serpong to Cigading (4 cask + tools)																							
	=5 hours=																							
	b. Cigading to Cilacap																							
	=26 hours=																							
	c. Yogya to Cilacap (1 cask + tools)																							
	=10 hours=																							
6	**CUSTOMS CLEARANCE**																							
7	**VESS HEAD FOR US**																							
	a. Leave for Cilacap Port																							
	b. Leave for Indonesia Territory																							

TABLE 2. SPECIFICATION OF TRIGA FUEL ELEMENT

No	N a m e	Specification				Unit
1	Catalogue number	102	104	106	108	—
2	Total length tube	75.5	75.5	75.5	75.5	cm
3	Outer diameter	3.65	3.56	3.75	3.75	cm
4	Fuel length	35.56	35.56	38.1	38.1	cm
5	Fuel composition	U Zr H	U Zr H	U Zr H	U Zr H	—
6	Weight U^{235}	37	38	55	99	gr
7	Weight % U^{235}	8.5	8.5	12	20	%
8	Enrichment	20	20	20	20	%
9	Graphite reflector at the end	10.2	10.2	10.2	10.2	cm
10	Tube material	Al	SS304	SS304	SS304	—

3. Management aspects

3.1. Legal issues

Indonesian Laws and regulations to support and control all activities of shipment operations are already in function. The operation is very much supported by the existence of bilateral cooperation agreements between United States of America and The Republic of Indonesia i.e. Protocol Amending “The Agreement for Cooperation between The United States of America and The Republic of Indonesia Concerning Peaceful Uses of Nuclear Energy” signed in Jakarta on 20 February 2004 that valid until 30 December 2031.

3.2. Contracting parties

Implementation of the second SNF shipment operation was based on the 2 contracts between USDOE and BATAN, namely:

(a) Contract No. DE-AC07-04-ID14557 between Idaho National Engineering and Environmental Laboratory (INEEL), US-DOE and P3TM-BATAN and P3TkN-BATAN, signed on 21 January 2004),
(b) Contract No. DE-GI09-99-SR18920 Modification 1, between Savannah River Operation Office (SRS)-US-DOE and P2TRR-BATAN, signed on 4 February 2004.

According to the contracts, both parties had well defined responsabilities:

Indonesia Party:

BATAN has responsibility to:
- Provide assistance to obtain permit and license of re-export activities, including permit for using Ports
- Provide data necessary to complete the contract Appendix-A: Spent Nuclear Fuel Acceptance Criteria and Appendix-B: Acceptance Criteria of Transport Cask.
- Provide security for entire activities in Indonesia.

Supported by:

BAPETEN: nuclear permit
Custom: export and re export permits
State Police: security fromsite to Port
Navy: security from Port to International water.

4. Preparation works

4.1. SNF Identification

Physical and chemical description, quantity, burn-up, cooling time, water chemistry, fuel physical condition of each fuel element were identified and then submitted to USDOE. These data were then utilized by them for preparing the nuclear safety assessment, choice of container, quantity of SNF that could be shipped on each cask, and handling requirement before shipment, as well as strategy for further disposal. As an example, Table 4 shows a summary of data for RSG-GAS SNF that was agreed for shipment.

TABLE 4. SUMMARY OF INFORMATION OF SHIPPED RSG-GAS MTR SPENT NUCLEAR FUEL

	Fuel Element	Fuel Element Cropped	Control Element	Fuel Plate
Quantity	6	83	22	1
Chemical Form	U_3O_8-Al	U_3O_8-Al	U_3O_8-Al	U_3O_8-Al
Dimension, cm	86.85 x 8.05 x 7.61	66.85 x 8.05 x 7.61	98 x 8.05 x 7.61	72.14 x 8.38 x 8.13
Weight, g	6084.14	5370.24	57.96.61	
Weight of U, g	1265.46	125.46	903.9	
Weight of U-235, g	249.9	249.9	178.50	
Enrichment, %	19.75	19.75	19.75	
Average burn up, %	49.41	49.41	48.64	
Min cooling, days	1548	657	556	
Max cooling, days	1666	2992	2870	

4.2. SNF Removal from spent storage to ISFSF (for RSG-GAS)

As the loading of the SNF was to be conducted from the ISFSF building, a number of 111 SNF stored in the storage pool of the reactor building had to be moved to ISFSF. Also, one fuel plate from Radiomethalurgy Installation was inserted into a Canister provided by NAC and placed in a rack in the ISFSF Pool.

4.3. Logistic and supporting infra structure

Most of all supporting logistic and infra structure was under scope of the US contractor, such as outside crane, forklift and specific handling tools. Site survey and adequate data was submitted to NAC, and considered as an important step.

4.4. Fuel cropping (for RSG-GAS)

A cutting machine was provided by NAC, in order to crop the fuel elements The machine was installed in the ISFSF pool as illustrated in Fig. 4.

It was identified that only 83 SNF of RSG-GAS reactor needed to be cropped by 7 inches of its end fitting. Cropping the lower end of all 83 SNF MTR type was accomplished in 2 days, 26–27 February 2004) of 5 planned days.

FIG. 4. Cutting machine arrangement over the ISFSF Pool.

5. Loading SNF

5.1. Loading activity

The performed loading activities can be summarized as follows:

(a) unpack tools
(b) stand the Transfer Cask (TC)
(c) bring Inner Shielding with basket to ISFSF
(d) load SNF to basket in Inner Shielding (24 SNF TRIGA or 7 SNF MTR types)
(e) insert Inner Shielding to ITS (Intermediate Transfer System)
(f) close Inner Shielding and decontaminate
(g) bring ITS from ISFSF to nearby TC under Dry Transfer System (DTS) by forklift
(h) move Isolation Cask from ITS to DTS
(i) load basket to TC using DTS, and close TC
(j) do leak test for TC
(k) pack TC back to container

Notes:
- One TC could accommodate 6 SNF baskets
- Control of radiation was conducted during each step of activity
- Regulatory Body was also in place during all the activity.

5.2. Implementation

The 9 containers, 5 with transport casks for SNF and 4 with working equipments, arrived at Tanjung Priok Port on 28 February 2004, and were directly distributed to Serpong (5 containers), to Bandung (3 containers), and to Yogyakarta (1 container). Then the activities were as follows:

At RSG-GAS Serpong:

Tools unpack, Load 111 SNF and 1 plate SNF in 3 TC and Leakage Test of transport casks SNF took 8 days, between 1- 8 March
LWT #1: 42 cropped SNF
LWT #2: 41 cropped SNF and 1 Canister of a fuel plate
LWT #3: 6 un-cropped SNF and 22 control element

At TRIGA-2000 Bandung

Loading activity of 111 SNF into 1 TC was executed between 1-5 March 2004.

At KARTINI Yogyakarta

Tools unpack and load 71 SNF into 1 TC, as well as helium leak test, were completed in 4 days, between 8-10 March 2004. There was one day delay because of some damage on the main crane (150 ton)..

The last step prior to transportation, was the inspection from custom and regulatory body officers. The container was then ready for shipment..

6. Transport operation

The main aspect of SNF transport was physical protection during transportation from reactor to Port. The route and procedures of transport were determined based on the results of surveys in which the closed and static security was in positions. The arrangement of transport was separated in 2 convoys with a speed of not more than 40 km/h. The main convoy consisted of a police car, folowed by the, SNF trucks, Radiation Protection, Security (Brigade Mobile), and another police car at the end, as illustrated in Fig. 5. The second convoy was made of supporting vehicles, with tool containers and officers. The plan was prepared by Security and Safety Group of BATAN, BAPETEN, Police, Army and US-DOE Officers.

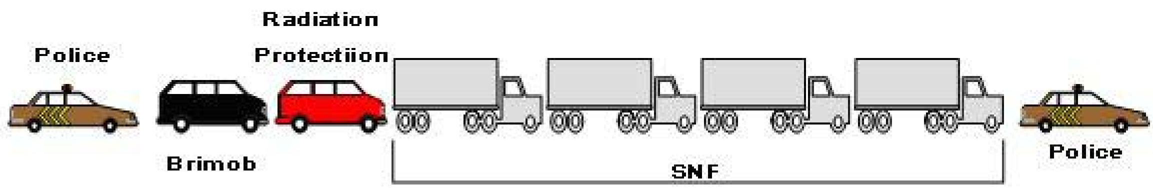

FIG. 5. Main convoy of SNF transportation.

Bandung to Serpong (ca. 244 km)

Transport of 1 SNF container and 1 Tools container to Serpong on 5 March 2004 and reached Serpong at 05.00 of 6 March 2004, with a speed of trucks around 40 km/h.
One container of tools was shipped to Yogyakarta for accelerating the work in the reactor.

Serpong to Cigading Port (ca. 130 km)

SNF from Serpong and Bandung were transported to Cigading Port on 9 March 2004. The 4 containers with SNFl and 3 containers with equipment arrived at 03.30.
Loading to Sea Bird ship was completed around 12.00, and the ship left for Cilacap Port about 285 miles away, RI NavyTeuku Umar.

Yogyakarta to Cilacap Port (ca. 250 km)

The transport of 1 container with SNF and 2 containers with tools from Yogyakarta to Tanjung Intan Cilacap Port was done on 10 March 2004 at 22.00 and arrived at 04.30. After loaded with all 3 containers, MV Sea Bird left for US on 11 of March 2004 at 20.00, escorted by RI- Navy Ship Teuku Umar untill international water.

7. Conclusion

Shipment operation of SNF from three research reactors in separated regions of Indonesia was successfully implemented under the US FRRSNF Acceptance Program, in safe, efficient and effective manner. Good cooperation and communication, as well as previous experience were the keys of the success. Extension of this program is considered to be important.

REFERENCES

[1] NATIONAL NUCLEAR ENERGY AGENCY, Reekspor Bahan Bakar BekasRSG-GAS ke Savannah River Site Amerika Serikat, Activity Report No. TRR.OR.18.04.43.04, BATAN, Jakarta (2004).

[2] NATIONAL NUCLEAR ENERGY AGENCY, Reekspor Bahan Bakar BekasReaktor KARTINI ke INEEL Idaho, Amerika Serikat, Activity Report No. 02/SNF/II/2004, BATAN, Yogykarta (2004).

[3] NATIONAL NUCLEAR ENERGY AGENCY, Reekspor Bahan Bakar BekasReaktor TRGA-2000 ke INEEL Idaho, Amerika Serikat, Activity Report No. 02/SNF/II/2004, BATAN, Bandung (2004).

Experience on return of research reactor spent fuels in Japan

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Abstract. In Japan, 1 712 of Research Reactor Spent Nuclear fuels (RRSNFs) have been transported to the US successfully in accordance with the foreign research reactor spent nuclear fuel acceptance policy that started in 1996. In special, Japan Atomic Energy Agency (JAEA) carried out eight shipment operations to the United States of America, transporting 1 283 fuel elements in total, and Kyoto University (KURRI) shipped 331 MTR-type fuel elements in six shipment operations. For each operation the operator is required to make many kinds of procedures to comply with requirements from government offices, and to perform regulatory inspections in Japan. It takes about one year to carry out the transportation, including preparation works and necessary procedures. In 2004, the deadline for US DoE to receipt the FRRSNF was extended to May 2019. It is understood that the reactor operator in association with DOE and the tranport company will continue to transport RRSNF safely from a viewpoint of the nuclear non-proliferation policy, considering the legislations and making the best use of former experiences. This paper describes experiences of RRSNF transportation to the US in Japan.

1. Introduction

The government of the United States of America started receiption and management of foreign research reactor spent nuclear fuel in the US in 1996 subsequent to completion by the DOE of an Environmental Impact Statement. The subject of receiption was established as US-origin spent fuel which was taken out from a reactor core by May, 2006 and which should arrive at the US by May 2009 considering cooling period and shipment arrangement for three years. The title of a spent nuclear fuel belongs to the US when the fuel is unloaded from a ship.

JAEA, Japan Atomic Energy Research Institute (JAERI) at that time, concluded the contract of receipt with DOE in 1997 and carried out the first transportation of the spent nuclear fuels from the JMTR research reactor in the same year. On the other hand, KURRI concluded the contract with DOE in 1998 and the first shipment of Kyoto University Reactor (KUR) spent nuclear fuels was carried out in 1999. Afterward, RRSNFs in Japan were transported to the US periodically. As of the end of 2004, 1 712 of RRSNFs were transported successfully. Especially, JAEA carried out eight shipments to the US accumulation a total of 1 283 spent nuclear fuel elements transported in total, and KURRI carried out six shipments of 331 KUR spent nuclear fuel elements in total so far.

2. Outline of spent nuclear fuel transportation to the United States of America

In most of the shipment operations from Japan, spent nuclear fuel elements are enclosed in a transportation cask which has enough strength and shielding, and conveyed to an available port by land with an exclusive trailer, and then transported to the port in the United States of America by way of the United Kingdom. Once the cask arrives at the US port it is conveyed to the Savannah River Site (SRS) by land.

It is important for a cask to secure the safety because RRSNF is highly activated. Therefore, a cask is designed and fabricated carefully in accordance with Japanese legislations and the IAEA Regulations for the Safe Transport of Radioactive Material. The safety of a cask is examined strictly by the Ministry of Education, Culture, Sports, Science and Technology (MEXT) and the Approval of Package Designs is issued. When transporting RRSNF, several kinds of inspections are carried out on the spot to confirm the adaptation to the approved technical standards by MEXT and the Ministry of Land, Infrastructure and Transport (MLIT).

3. Outline of RRSNFs

The main research Japanese reactors that have shipped RRNSF to the SRS are now JMTR, JMTRC, JRR-3, JRR-4 of JAEA and KUR of Kyoto University.

JMTR is the most powerful testing reactor (50 MW) dedicated to irradiation tests of materials and nuclear fuels in Japan. JMTRC was a critical facility assembled to evaluate the core characteristic of JMTR and to carry out reactor physics experiments, but it has been shut down in 1994. JRR-3 was the first Japanese designed and constructed reactor (10 MW) and it was modified in 1990 as a high-performance, multi-purpose research reactor with a maximum output of 20 MW. JRR-4 was constructed to test the reactor shielding of the first Japanese nuclear ship “Mutsu” and it was also modified in 1996 as a multi-purpose research reactor with a medical irradiation facility.

KUR (5MW) is a multi-purpose research reactor with HEU fuels used for neutron beam experiments, activation analysis, medical irradiation and so on. It was temporarily shut down in February, 2006 and it is now being modified so as to use LEU fuels.

All fuels of these research reactors are MTR plate type. The outline of these research reactors is shown in Table 1 and for example, the structure drawing of fuel element of JMTR is shown in Fig.1.

Outline of these fuels are as follows [1].

3.1. Spent fuel of JMTR

In JMTR, HEU fuel of 93% enrichment was used since its operation started and HEU fuel of 45% enrichment, named MEU (Middle Enrichment Uranium) fuel conveniently to distinguish from the 93% enriched fuel, was used from 1986 as a switchover period to LEU fuel. Finally, the full conversion to LEU fuels for the reactor core was achieved in 2001. The fuel of JMTR is a plate type. The HEU fuel was a uranium-aluminum alloy, the MEU fuel was a uranium aluminum dispersed type (aluminide fuel) and the LEU is a uranium silicon aluminum dispersed type (silicide fuel). The average burn-up of the MEU spent fuel and the LEU spent fuel is about 20% and 35%, respectively. The necessary cooling period for the MEU spent fuel and the LEU spent fuel is more than 360 days and 420 days, respectively. On the other hand, all of the HEU spent fuels have been already transported to SRS by 1998.

3.2. Spent fuel of JMTRC

In JMTRC, HEU fuel of 90% enrichment was used and its burn-up is negligible small because JMTRC is a critical facility. Twenty fuel elements were transported in 2003 for the first time.

TABLE 1. OUTLINE OF MAIN RESEARCH REACTORS IN JAPAN

	JMTR	JRR-3M	JRR-4	KUR
Owner	JAEA	JAEA	JAEA	Kyoto Univ.
Max. Power	50 MW	20 MW	3.5 MW	5 MW
Reactor Type	Tank	Pool	Pool	Tank
Utilization	Irradiation test of materials and fuels Power ramp test RI production	Neutron beam experiments Si doping NAA NRG	NAA Si doping Training BNCT	Neutron beam experiments NAA Training BNCT
Fuel	93% U-Al (from 1968.3) 45% UAlx-Al (from 1986.7) 20% U_3Si_2-Al (from 1994.1)	20% UAlx-Al (from 1990.3) 20% U_3Si_2-Al (from 1999.9)	93% U-Al (from 1965.1) 20% U_3Si_2-Al (from 1998.7)	93% U-Al (from 1964.6) 20% U_3Si_2-Al (2 elements) (from 1991.4)

3.3. *Spent fuel of JRR-3M*

In JRR-3M, LEU aluminide fuel with enrichment less than 20% was used until 1999 and then it has been converted to LEU silicide fuel. The average burn-up of the aluminide spent fuel and the silicide spent fuel is about 40% and 55%, respectively. The necessary cooling period for both types of spent fuels is more than one year.

3.4. *Spent fuel of JRR-4*

In JRR-4, HEU fuel of 93% enrichment was used until 1996 and LEU fuel with enrichment lower than 20% is used since 1998. The average burn-up of the LEU spent fuel is about 23% and the necessary cooling period is more than one year.

3.5. *Spent fuel of KUR*

In KUR, HEU fuel of 93% enrichment is used and Kyoto University is to convert the KUR to use LEU fuel in the near future. The average burn-up of the HEU spent fuel is about 23% and the necessary cooling period is more than one year. So far, Kyoto University carried out the shipment of spent fuel elements six times.

4. Outline of transportation cask

A transportation cask is examined and inspected strictly by MEXT in accordance with the technical standards which provide structure and performance. The cask for JMTR spent fuels is shown in Fig.2 as an example [2].

This cask consists of a main body, a lid, a fuel basket, shock absorbers, a drain valve and a vent valve. This cask is cylindrical with an outer diameter of 1.9 m and a height of 2.0 m, and its total weight is about 18.5 tonnes. The main body is fabricated by forged stainless steel, and its inner diameter, shell thickness and baseplate thickness are 66 cm, 33.4 cm and 36 cm, respectively. The lid is also made from stainless steel with a thickness of 37 cm and fastened to the main body by 24 bolts. The sealing performance is secured by double O-rings. This cask is a dry type without water as a coolant. The heat generated from spent fuels is transferred to the main body by natural convection and radiation and released to the atmosphere through the fins which are attached to the main body. Aluminium plates containing sintered B_4C are installed in the fuel basket as a neutron absorber to secure subcriticality.

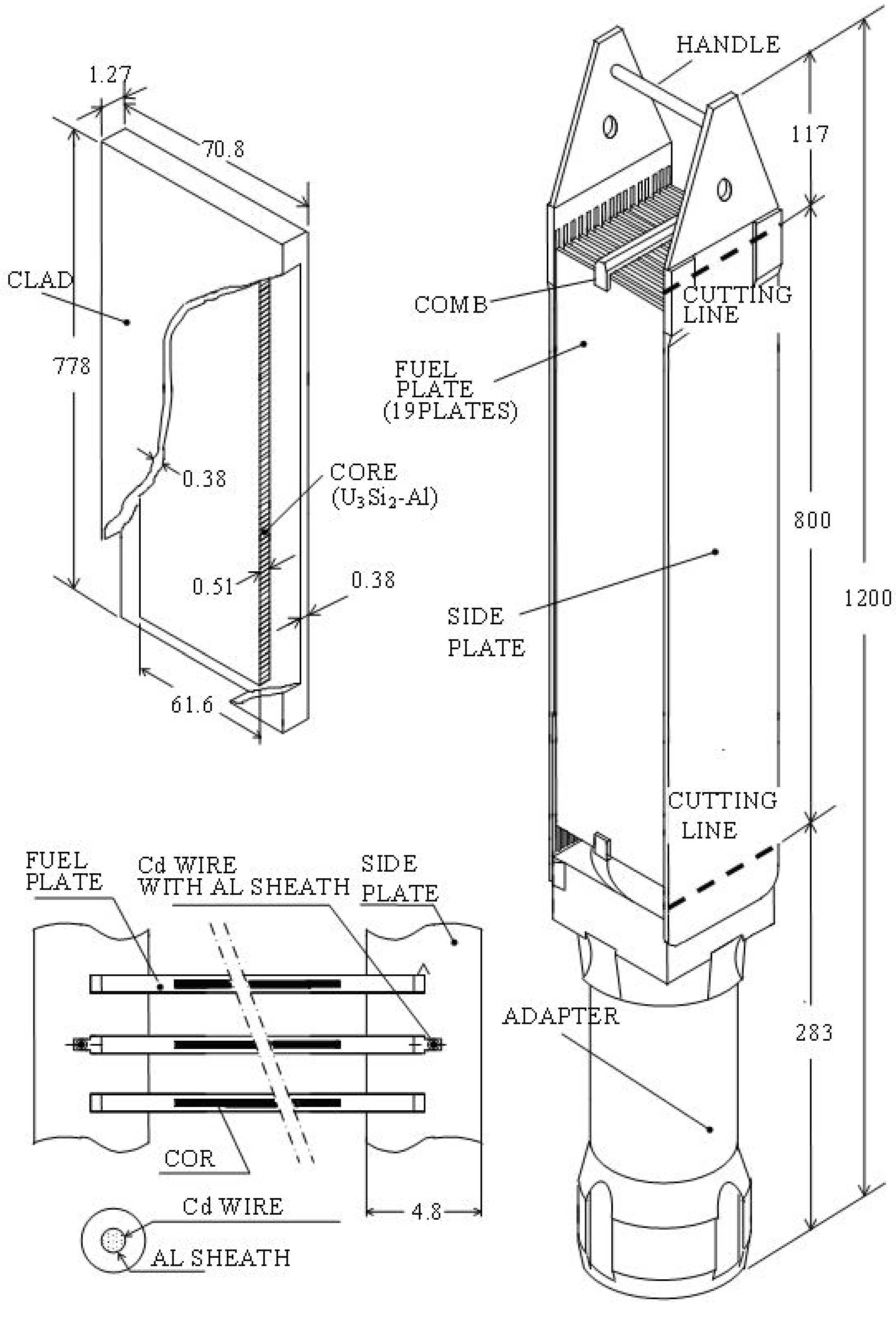

FIG. 1. Construction drawing of fuel element of JMTR.

The operator has to carry out a periodical self-inspection of the cask once a year in Japan, and has to obtain the Approval of Package Design of the US authorities every three years.

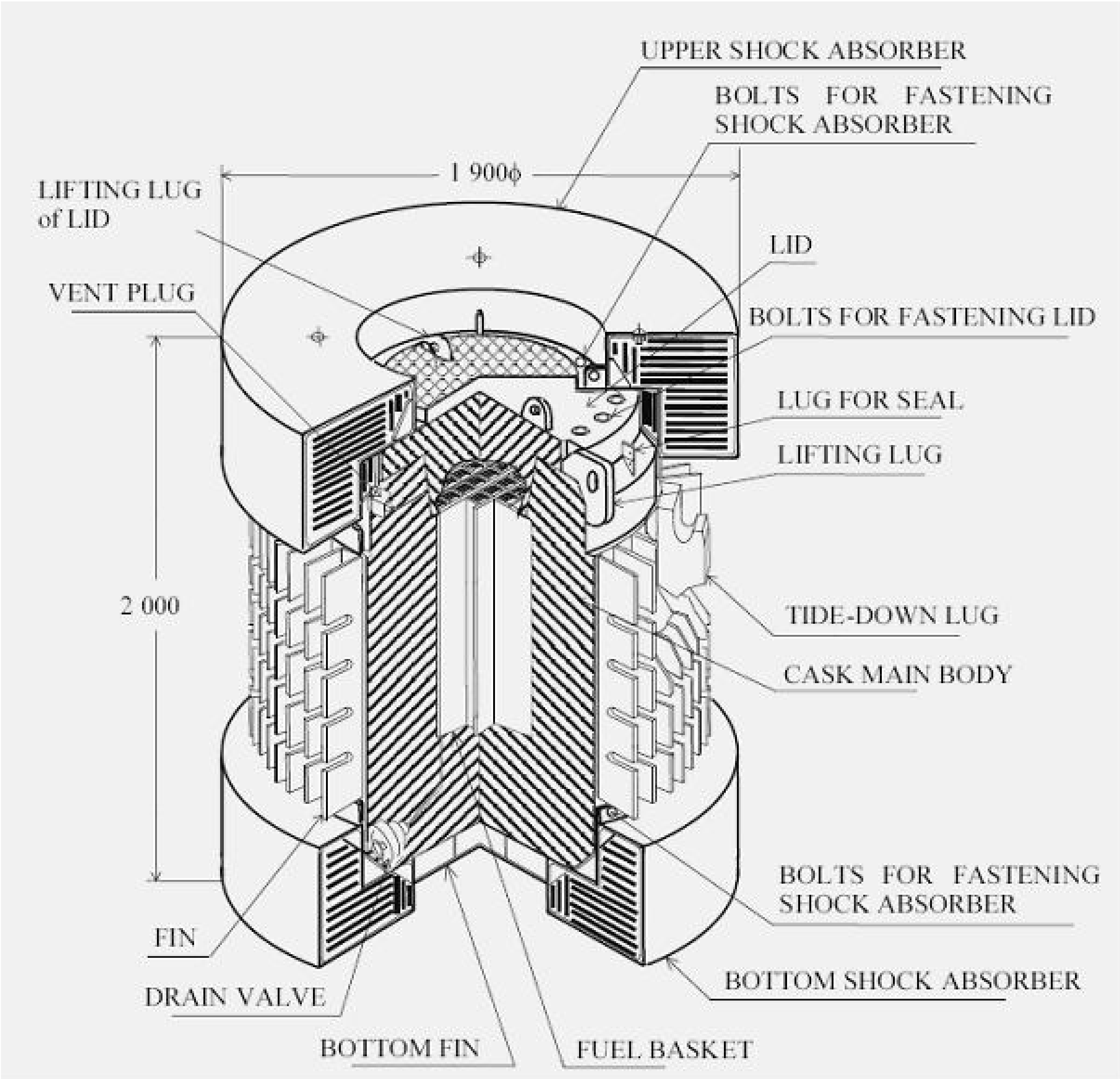

FIG. 2. Construction drawing of transportation cask (JMS-87Y-18.5T).

5. Procedure of RRSNF transportation

5.1. Preparation prior to transportation

5.1.1. Approval of the USDOE

The specifications of RRSNFs such as a type, quantity, composition, dimensions, enrichment, activity, amount of produced plutonium, operation log, etc. are submitted to the USDOE in order to obtain its approval prios to shipment. In Japan, the activity of spent fuel is calculated using the ORIGEN code [3],and the produced Pu is calculated using the SRAC code [4].

5.1.2. Preparation of cask

Prior to loading the RRSNFs, the periodical self-inspections including a visual test and an air tight test are carried out to confirm the soundness of a cask. Then the cask is sunk in the fuel storage pool with the depth of 6m. The water in the pool is controlled with an ion exchange resin so that the conductivity of water is less than 10 μS/cm, (2 μS/cm in the case of JMTR), and the pH is kept in the range of 5.0 to 7.5.

5.1.3. Loading of RRSNF

The top and bottom parts of a fuel element are cut off by using a cutting machine with a disk cutter to reduce fragments except nuclear materials as much as possible and the weight is measured for every cut fuel element. A neutron source of americium-berylium (Am-Be) is inserted into the fuel basket and neutron measurement is carried out to confirm subcriticality whenever each fuel element is loaded to the cask. The loading is carried out with a handling tool manually. After loading all fuel elements, a lid is installed and fastened underwater and the cask is taken out from the storage pool. Then the decontamination process of the external surface of the cask is carried out. The water inside of the cask is extracted through a drain line to measure radioactive nuclides. After confirming that there is no radioactive nuclide in the sampled water, which means that loaded fuel elements are sound, the water in the cask is fully drained.

5.2. Transportation

5.2.1. Land transportation

Before the package (cask and radioactive contents) is loaded on a truck trailer, it is checked *on the spot* by MEXT, to verify if the cask is manufactured in accordance with approved package design, if the radioactive contents are within acceptable limits of the approved package design, and if the dose rate on the surface of the package, surface contamination, containment system, etc. satisfy the approved technical standards. After loading the package on the trailer, and obtaining MEXT approval, the loading methods and dose rates around the trailer are checked to confirm conformation to the technical standards by MLIT. Experts for nuclear material control and radiation control have to accompany the transport. Necessary equipments such as radiation detectors, an extinguisher, etc. are also carried in land transportation.

The main processes and inspections required for land transportation are illustrated in Fig. 3.

5.2.2. Sea transportation

After the package is loaded from the trailer to the ship, the loading methods and dose rates around the ship are checked again by MLIT. So far, the package was transported to the available port of the US by way of the UK and then it was conveyed to SRS by land. The reason why the package stopped in the UK is that other nuclear materials of Europe are transported to the US together with the package. However, the transportation without stopping in the UK must be considered in the future from the viewpoint of the physical protection.

5.2.3. Organization when transportation

During transportation a local headquarter is organized to provide actions against an accident because when an unusual event occurs at the transport of radioactive materials, the operator must notify the fact to related government offices immediately, and must take emergency measures for safety, as necessary, such as deployment of watch-persons, enforcement of no-entry measures, decontamination, relief measures, etc. in accordance with laws and regulations.

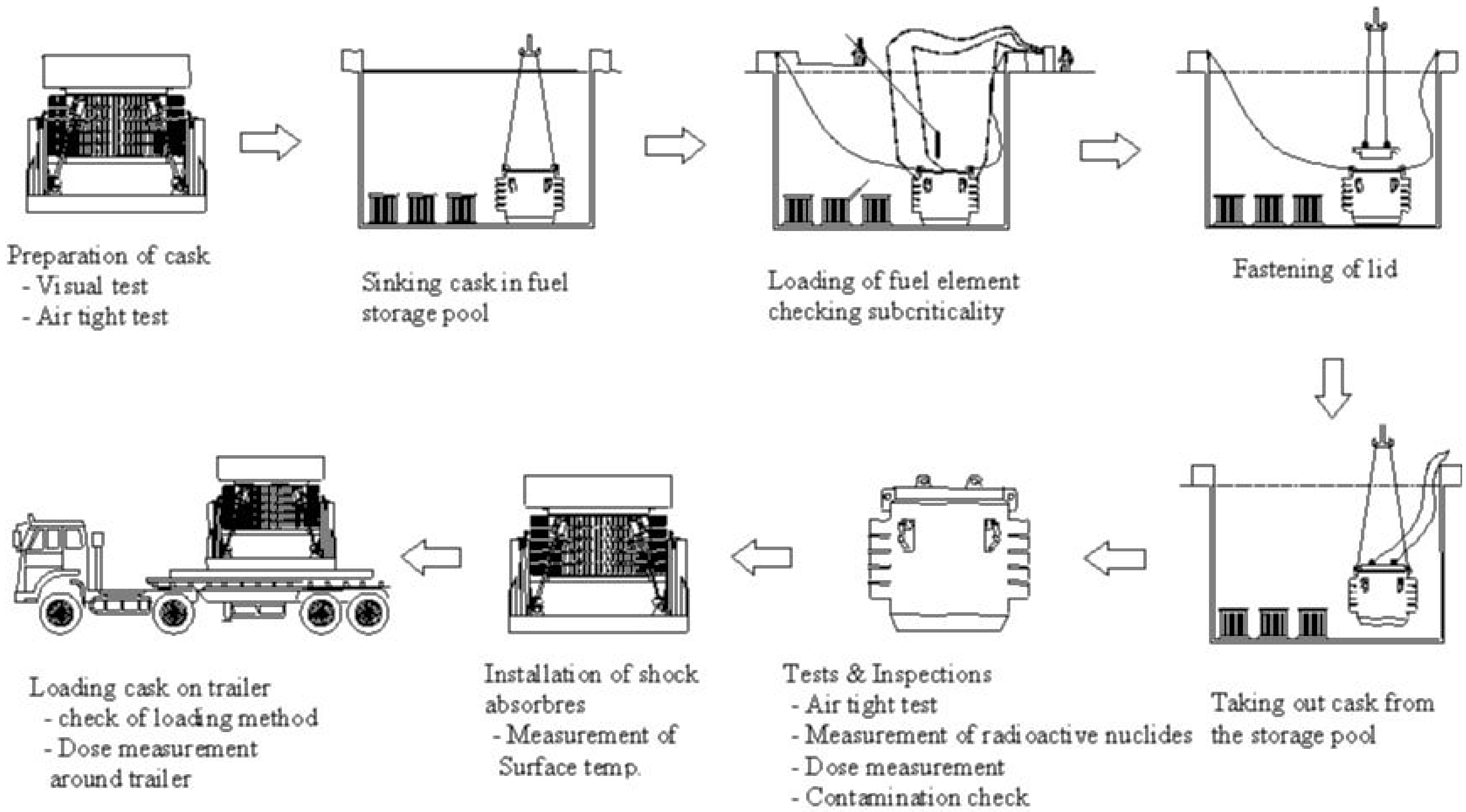

FIG. 3. Main processes of RRSNF transportation until land transportation.

5.3. *Return of cask*

The cask used to transport the RRSNFs is decontaminated in the SRS Laboratory, and returned to Japan. Although the cask is sufficiently decontaminated, it is treated as a radioactive material. Therefore, it is returned to Japan with a limited ship which is authorized to transport radioactive material.

6. Conclusion

RRSNF transportation to the US has been carried out, with each operation taking about one year, including preparation works in accordance with the legislation and paying careful attention to the safety regulations. So far, more than 1 700 RRSNFs were transported to the US successfully. The experience of shipment to the SRS is shown in Table 2.

TABLE 2. THE EXPERIENCE OF RRSNF SHIPMENT TO THE US IN JAPAN (AS OF THE END OF 2004)

Organization	Number of transported spent fuel element	Fuel type	Remarks
JAEA	1 283	Plate type	8 times since 1997
Kyoto University	331	Plate type	6 times since 1999
Toshiba Co., Ltd	27	Plate type	in 2003
Rikkyo University	71	Rod type	in 2003

The decision to extend of the deadline to receive FRRSNF was decided by the US Government in 2004. According to the decision, to be eligible for the programme the spent fuel must be taken out from the reactor core by May, 2016 and arrive at the US by May, 2019. Moreover, the legislation was amended on a basis of the INFCIRC/225/Rev.4 and enforced in December, 2005. The management of information and measures related to physical protection were reinforced in the amended legislation.

Finally, transporting RRSNF to the US is a very expensive operation. The organizations under the control of the Government such as JAEA and the Kyoto University, have their budget provided from the Government. This means that the Government considers the return of RRSNF an important thing from a viewpoint of the nuclear non-proliferation policy, and this support, is an assurance that the operator of the research reactors in Japan will continue RRSNF transportation safely, considering the legislations as well as making the best use of former experiences.

REFERENCES

[1] NAKAGOME, Y., UNESAKI, H., SAGAWA, H., Status of Reduced Enrichment Program for Research Reactors in Japan, Proc. 27th International Meeting on Reduced Enrichment for Research and Test Reactors, Boston, United States of America, November 06-10, 2005, Argonne National Laboratory, Argonne (2006) available at: http://www.rertr.anl.gov/RERTR27/PDF/S2-4_Nakagome.pdf

[2] MIYAZAWA, M., et al., Drop Tests of Type-B Packaging Under -40°C Conditions, Proceedings Volume I, 171-177 PATRAM'95, Las Vegas, USA, December 3-8, 1995.

[3] KOYAMA, K., YAMANO, N., MIYASAKA, S., "ORIGEN-JR : Computer code for calculating radiation sources and analyzing nuclide transmutations", JAERI-M 8299 (1979).

[4] TSUCHIHASHI, K., ISHIGURO, Y., KANEKO, K., IDO, M., "Revised SRAC code system", JAERI 1302 (1986).

The experience on the return of research reactor spent fuels of Korea

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Abstract. In 1997, the HANARO Steering Committee agreed to ship the TRIGA spent nuclear fuels back to the country of origin according to the Record of Decision (ROD) on a Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel, which was issued by DOE (US Department of Energy) in 1996 and established the foreign research reactor spent fuel acceptance program. On April 1998, a total of 299 spent fuel elements discharged from the KRR-1 (200 kW TRIGA Mk-II) and KRR-2 (2 MW TRIGA Mk-III) were returned to the US without any charge. INEEL/Lockheed Martin Idaho Technologies Co. managed the transportation and Nuclear Assurance Company (NAC) took on the shipment.

1. Introduction

In 1962, TRIGA Mk-II was constructed for the first time in Seoul, Korea, which was operated at 100 kW in thermal power and updated to 250 kW in 1969. Ten years later, a TRIGA Mk-III, with higher thermal power (2 MW) was built near the TRIGA Mk-II site. Recently they were named as KRR-1 (Korea Research Reactor No. 1) and KRR-2 respectively. KRR-1 and KRR-2 had been operated for 34 and 24 years respectively, until their permanent shutdown in 1995. The total operation time and cumulative power generation of the two reactors were 36 535 hours and 3 735 MWH, and 55 226 hours and 68 740 MWH respectively. As the new high performance research reactor HANARO came to be realized in 1995, KAERI (Korea Atomic Energy Research Institute) made a decision to stop the operation of two TRIGA reactors and submitted the reactor shutdown report of KRR-1 and KRR-2 to MOST (Ministry of Science and Technology) in 1996. At first the spent nuclear fuels from these two research reactors were planned to be stored at the spent fuel storage pool of the HANARO facility until the plan for the permanent disposal of them was established. While making a plan to move the spent fuels to HANARO in Daejeon, which is about 170 km away from Seoul, the Korean government agreed with the US DOE about the acceptance of foreign research reactor SNF policy. As a result all spent nuclear fuels were returned to US on April 1988 but the costs for the transportation were exempted based on the economic position of Korea in World Bank Development Report. It was the first shipment of SNF out of Korea.

2. Status of TRIGA fuels

2.1. Fuel description

The TRIGA fuel element is a solid, homogeneous mixture of hydride uranium-zirconium alloy containing 8 percent by weight of uranium. A standard type fuel element is enriched to 20 percent in U-235 and consists of fuel meat having a diameter of 3.58 ~ 3.64 cm and a length of 35.6 ~ 38.1 cm. It has two graphite reflectors above and below of the fuel meat and 0.7 mm thick aluminum or 0.5 mm thick stainless clad. A FLIP (Fuel Life Improvement Program) type element has the same geometry as stainless steel clad standard element, but it is highly enriched to 70% and loaded with natural erbium as burnable poison [1].

Some fuel element have three chromel-alumel thermocouples embedded along the vertical centerline of the element. The lead wires are protected and guided up to the surface of the pool by a long tube which is sealed with the plug at the top of the fuel element. This instrumented fuel element was used in both KRR-1 and KRR-2.

Fuel followed control rod (FFCR) is boron carbide loaded rod which has a fuel portion below the poison part to prevent hard distortion of neutron flux when it is being withdrawn from the core. The fuel portion is identical to the FLIP fuel element and it was used in KRR-2 only.

2.2. Fuel inventory

A total of 299 spent fuel elements were produced from KRR-1 and KRR-2 during their operations over 30 years. They consist of 178 LEU (standard) and 121 HEU (FLIP) elements. Of the LEU elements, 69 are aluminum clad and 109 are stainless steel clad. Of the 299 spent fuel elements, 186 fuel elements were stored on the floor and along the wall of the KRR-2 reactor pool using available storage racks. Other 113 fuel elements including, 9 canned elements, were stored in 4 dry storage casks at the storage facility in Daejeon. They were used at KRR-2 and suffered variable (large and small) damage due to the leakage of secondary cooling water into the reactor pool through some penetrated parts of the heat exchanger. Table 1 shows a summary of the spent nuclear fuel elements.

TABLE 1. INVENTORY OF TRIGA SPENT NUCLEAR FUEL ELEMENTS

Type	Location		Sub-total	U-235 (g)
	Seoul	Daejeon		
Standard (Al-clad)	61	6	67	2 613
Standard (SS-clad)	19	84	103	4 107
Standard Instrumented (Al-clad)	2	0	2	78
Standard Instrumented (SS-clad)	1	1	2	78
Standard FFCR (SS-clad)	1	3	4	128
FLIP (SS-clad)	92	18	110	14 960
FLIP Instrumented (SS-clad)	6	1	7	945
FLIP FFCR (SS-clad)	4	0	4	448
Total	186	113	299	23 214

3. Shipment of the spent fuels

3.1. Preparing

In 1996 US DOE/INEEL and NAC visited Seoul and Daejeon in order to introduce and discuss the US FRRSNF acceptance program, and also to assess the fuel storage condition. KAERI completed the INEEL questionnaire providing contact points, preliminary fuel data and drawings, and post irradiation data including operating history, cooling time and estimated plutonium production of each fuel element. Through the discussion between KAERI and INEEL, INEEL established the schedule for fuel examination, cutting, canning, loading and shipment [2][3].

Because the fuel elements were stored in different areas, the arrangements for shipment were progressed in both Seoul and Daejeon. Most fuels in Seoul were generally in good condition except two ruptured fuel elements. Since they were well arranged in the reactor pool of KRR-2, no special preparatory works were needed for their shipment. There were some difficulties, however, in accessing into the KRR-2 building due to a narrow paved road. In addition, the insufficient capacity of the overhead crane at the reactor hall made it impossible to handle the heavy shipping cask inside of the reactor hall.

In Daejeon, to provide suitable working place for the examination, cutting, canning, and loading to the transfer cask, the dry casks containing damaged fuels were moved to a transfer canal available between the service pool and the SNF storage pool of the HANARO facility. After that all fuel elements were removed from the casks and rearranged in the temporary storage racks within the canal. The 30 ton overhead crane at the HANARO facility was sufficient for all operations

3.2. Examination

The examination of the fuel elements was done by visual method based on acceptance criteria of INEEL. After setup and test of the equipment for fuel examination, the inspection proceeded with identification of each element, then dimensional and straightness checks were performed along with video recording, photographic and written documentation of the fuel condition by sector. Samples were taken from the storage pool water and from the pool liner. From the examination made in Seoul the fuel elements were found to be acceptable for shipment except 2 ruptured stainless steel clad fuels and 18 aluminum clad fuels, which presented suspected corrosion or cladding penetration. While the examination in Seoul was being conducted, the unloading of the dry casks and rearranging of the fuels in the storage racks on the HANARO canal were carried out in Daejeon, prior to the arrival of the examination team. In the process of unloading the fuel elements, some fuels could not be immediately unloaded from the grid of the dry cask due to some deformation of the fuel, and required additonal work to be unloaded. Similar examination as done in Seoul was performed in Daejeon and it was confirmed that 105 fuels were accetable for shipment without any supplementary means. They included 34 stainless steel clad fuels which were founded to be deformed in the upper or lower graphite section and 3 canned fuels. The remaining 8 fules, including 6 canned fuels, had to be re-canned for shipment due to their severe damages such as rupture, cracks or corrosion.

3.3. Shipment and transportation

NAC provided the transporting service of returning KRR SNFs to INEEL, and three NAC-LWT shipping casks were used. These casks were specifically configured to load all type of TRIGA fuels and canned fuels as well. Five modular baskets can be piled up vertically in a shipping cask, and up to 24 fuel elements could be loaded in each basket. To accommodate instrumented fuels and FFCR, the baskets were of three different lengths up to 45 inches long. The fuels were loaded into the baskets under the water and the basket was transferred into a transfer cask through a shielded funnel. The transfer cask was then mated with the shipping cask at the top entrance and the basket inserted into the shipping cask. Due to the insufficient capacity of the overhead crane and space limitations in KRR-2, NAC manufactured new loading equipment. NAC also prepared an under-water saw and a shear to cut the instrumented fuels and FFCRs. They also prepared cans to accommodate the damaged fuels.

The shipment was done first in Daejeon, on the Hanarao facility. The first shipping cask was loaded with one basket containing FLIP fuels and four empty baskets then it was sent to Seoul. Then the second cask was loaded with five baskets containing the remainder fuels in the HANARO facility and also sent to Seoul. The FLIP fuels stored in Seoul were loaded into the vacant baskets of the first cask and the other fuels were loaded into a third shipping cask which had been delivered to Seoul directly. All casks were transported to Incheon port by trucks for sailing to the US.

4. Result

In Korea, two TRIGA-type research reactors are in the process of decommissioning. The spent fuels of the reactors were already sent back to the US through the US DOE’s foreign research reactor spent fuel acceptance program. The new research reactor, HANARO uses low enriched uranium silicide fuels. The spent fuels are being stored at the spent fuel storage pool of HANARO. The storage capacity of the spent pool was designed to store 20-years of produced spent nuclear fuels. However, HANARO is expected to operate for more than 40 years and currently no other temporary or permanent disposal facilities are planned. In 2001, HANARO Steering Committee agreed again to ship HANARO spent nuclear fuels back to US to ensure the availability of enough storage space. However, the economic status of Korea has changed to high-income country in 2002, and this has caused some difficult in consolidating any new shipment operations, because now we have to bear the shipping costs. As a result, instead of returning the SNFs we recently proposed an upgrade of our storage rack in order to store more SNFs in the HANARO facility.

REFERENCES

[1] TOMSIO., N., "Characterization of TRIGA Fuel", General Atomics, GA-C18542, (1986).

[2] COLE, C.M., et al., "Assessment Result of the South Korea TRIGA SNF to be shipped to INEEL" (Proc. RERTR Int. Meeting, Jackson Hole, Wyoming, USA, 1997).

[3] PARKER, D., et al., "Operational Aspects of TRIGA Shipment from South Korea to INEEL" (Proc. RERTR Int. Meeting, Sao Paulo, Brazil, 1998).

The Philippine experience on return of research reactor spent nuclear fuel to the country of origin

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Abstract. In 1999, the Philippine Nuclear Research Institute returned all of its research reactor spent nuclear fuel to the U.S.A. under the latter's program to accept spent nuclear fuel of U.S. origin. The reactor had no built-in capability to load the shipping casks and the spent nuclear fuel elements were stored in an unfavourable location, and the work had to be done within a few days. The work was done safely and on time using rented common construction equipment. This paper describes the careful preparation and the on-site work that enabled the shipment to succeed.

1. Introduction

The Philippine Research Reactor (PRR-1) was obtained from the U.S.A. under the Atoms for Peace program by the Philippine Atomic Energy Commission, which later became the Philippine Nuclear Research Institute (PNRI). PRR-1 operated at up to 1 MW (thermal) from 1963 to 1984 with fuel elements sourced entirely from the U.S.A. The reactor was shut down in 1988. In January 1998, under the initiative of the U.S.A. to accept foreign spent research reactor fuel of U.S. origin, a contract was signed between the PNRI and the U.S. Department of Energy (U.S. DOE) to return to the U.S.A. all the spent nuclear fuel of the PRR-1 reactor. Philippines was qualified as a developing country under the U.S. initiative and the shipment was to be fully paid for by the U.S. DOE. The U.S. DOE chose the company NAC International Inc. (NAC) as its contractor. NAC and its subcontractors worked closely with the staff of PNRI to prepare the spent nuclear fuel elements for shipment [1].

2. The spent fuel

The description of the fuel element used in the PRR-1 research reactor, as follows, is quoted from the reactor's 1960 Safety Analysis Report [2].

"The fuel is of the General Electric Flat Plate type as shown in Fig. 1[1]. Each element consists of two aluminum side plates and 18 equally-spaced flat fuel plates. The meat of the fuel plate is a uranium-aluminum alloy, about 30w% uranium enriched to 20% U-235[2], sandwiched between 15-mil aluminum cladding on each side[3]. The plate is fabricated by the so-called picture frame technique. The fuel plates are 0.060 inch thick, 2.79 inches wide, and 25 inches long. The active, fuel bearing length is 24 inches. When assembled in the fuel element, plates are separated by a 0.1-inch gap for water passage. Two identical end boxes position the fuel element in the grid and provide handles for refuelling. Including end boxes, the elements are nearly 40 inches long. The elements may be inverted and rotated to achieve more efficient utilization of the fuel."

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[1] The number of the figure on the original text is "3", and it was modified to "1" on this paper.
[2] 20 of the later fuel elements were enriched to 93%
[3] 20-mil cladding in the 93% enriched plates

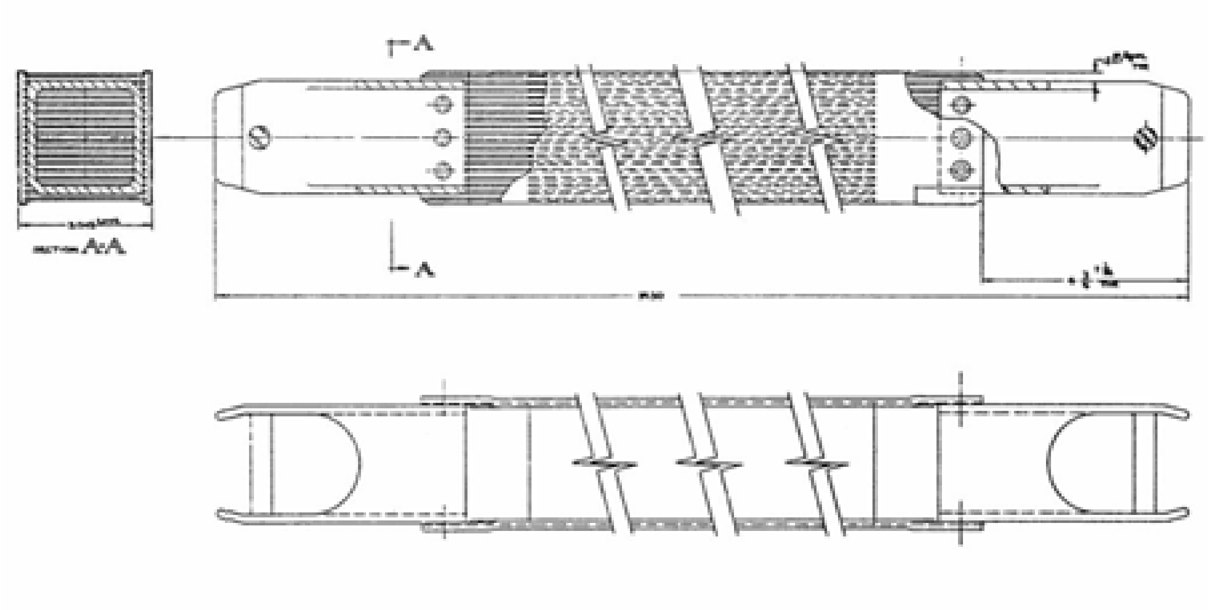

FIG. 1. The PRR-1 plate-type fuel element.

The characteristics of the spent nuclear fuel are given in Table 1. In addition to the 50 spent nuclear fuel elements, the PNRI also agreed to return to the U.S.A. a set of 18 loose plates containing 93% enriched uranium, equivalent to one fuel element. Those loose plates were intended to be used for reactor criticality experiments but were never actually irradiated. The loose plates were boxed and placed in the same shipping cask as the spent nuclear fuel elements.

TABLE 1. CHARACTERISTICS OF THE SPENT NUCLEAR FUEL FROM THE PRR-1 RESEARCH REACTOR

	Original Core	*Reload 1*	*Reload 2*
Number of Fuel Elements	30	10	10
Nominal Enrichment	20%	93%	93%
Nominal mass of U-235 Per Element	134 g	137 g	155 g
Total mass of U-235 in Batch	4 040.58 g	1 369.26 g	1 555.21 g
Total mass of U in Batch	20 324.76 g	1 470.02 g	1 669.61 g
Total Spent Fuel Elements	50		
Total Original mass of U-235	6 965 g		
Total Original mass of U	23 464 g		
Integrated Energy Release	617 MW·d		
Estimated Total U-235 Burnup	775 g		
Estimated U-235 Burnup Range	0.2% to 20%		
Estimated Total mass of Pu-239 Content	60 g		

3. The Shipping Casks

The spent nuclear fuel elements were transported to the U.S.A. in NAC-LWT casks provided by NAC and licensed by the U.S. Nuclear Regulatory Commission and by the U.S. Department of Transportation. The description below is quoted from the cask's Safety Analysis Report [3].

“The cask design is optimized for legal weight over the road, with a gross weight of less than 80 000 pounds[4]. The NAC-LWT cask assembly is composed of a package that provides a containment barrier, preventing the release of radioactive material. The actual containment boundary provided by the package consists of a 4.0-inch thick bottom plate, a 0.75-inch thick, 13.375-inch inner diameter shell, an upper ring forging, and an 11.3-inch thick closure lid. The cask lid closure is accomplished using twelve, 1-inch diameter bolts. The cask has an outer shell, 1.20 inches thick, to protect the containment shell and also to enclose the 5.75-inch thick lead gamma shield. Neutron shielding is provided by a 5.0-inch thick neutron shield tank with a 0.24-inch thick outer wall, containing a water/ethylene glycol mixture and 1.0 weight percent boron (58 w/o ethylene glycol); 39 w/o demineralized water; 3 w/o potassium tetraborate. The neutron shield tank system includes an expansion tank to permit the expansion and contraction of the shield tank liquid without compromising the shielding or overstressing the shield tank structure. Aluminum honeycomb impact limiters are attached to each end of the cask to absorb kinetic energy developed during a cask drop, and limit the consequences of normal operations and hypothetical accident events”.

Figure 2 is a diagram of the NAC-LWT cask. For the shipment of the spent nuclear fuel from the PRR-1 reactor, the maximum authorized capacity of one cask was 42 MTR-type fuel elements with end boxes cut off (cropped) in 6 short baskets. So two casks were needed to ship the 51 fuel elements. However, considering that two casks could hold up to 56 uncropped fuel elements in 4 standard baskets in each cask, none of the PRR-1 spent nuclear fuel elements were actually cropped for shipping.

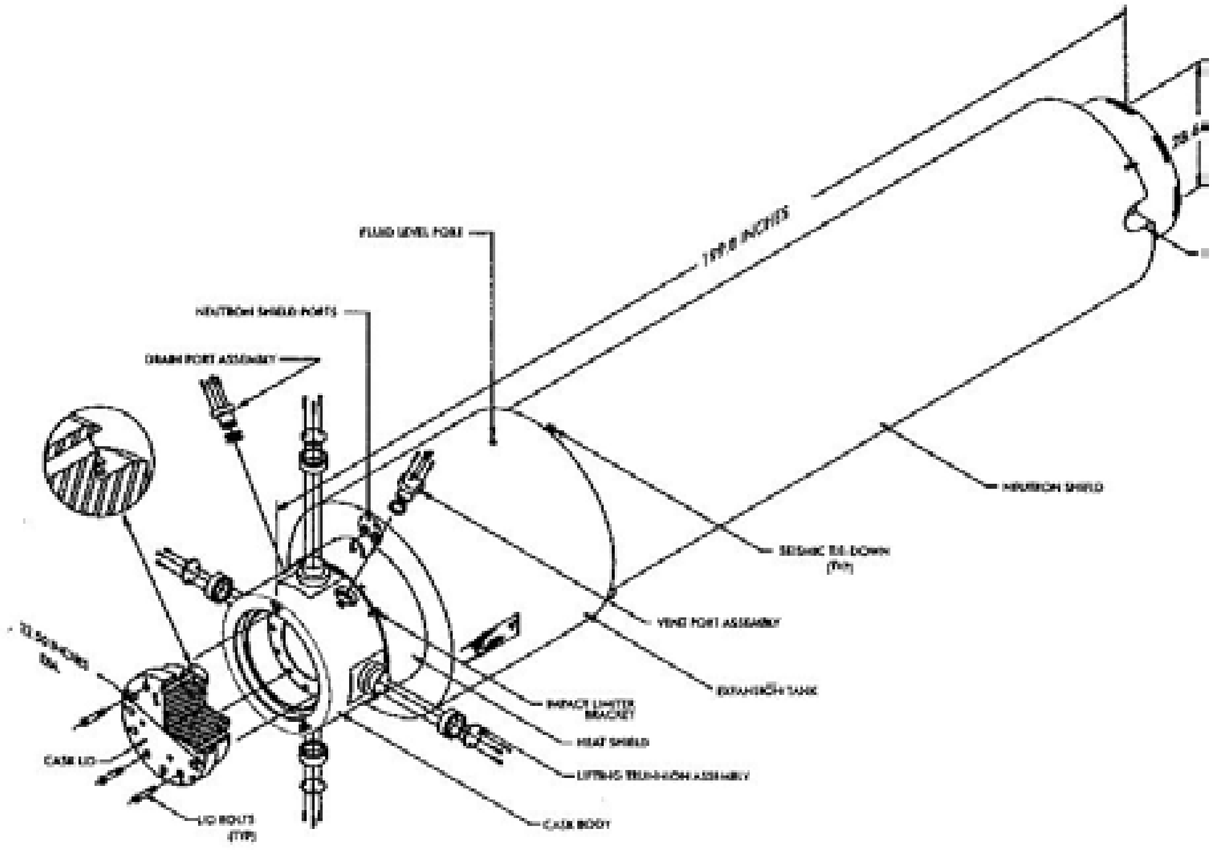

FIG. 2. The NAC-LWT spent-fuel shipping cask.

[4] *The cask alone weighs about 20 tonnes*

4. The spent nuclear fuel location

By 1999, the PRR-1 reactor had been shut down for 11 years, the reactor core had been unloaded and the reactor pool dewatered. The spent nuclear fuel elements were in racks inside of a free-standing stainless-steel tank placed on the Reactor Bay floor, as shown in Figs 3 and 4. The storage tank was 12 feet (3.7 m) in diameter and 16 feet (4.9 m) high. Access to the top of the tank was via scaffolding.

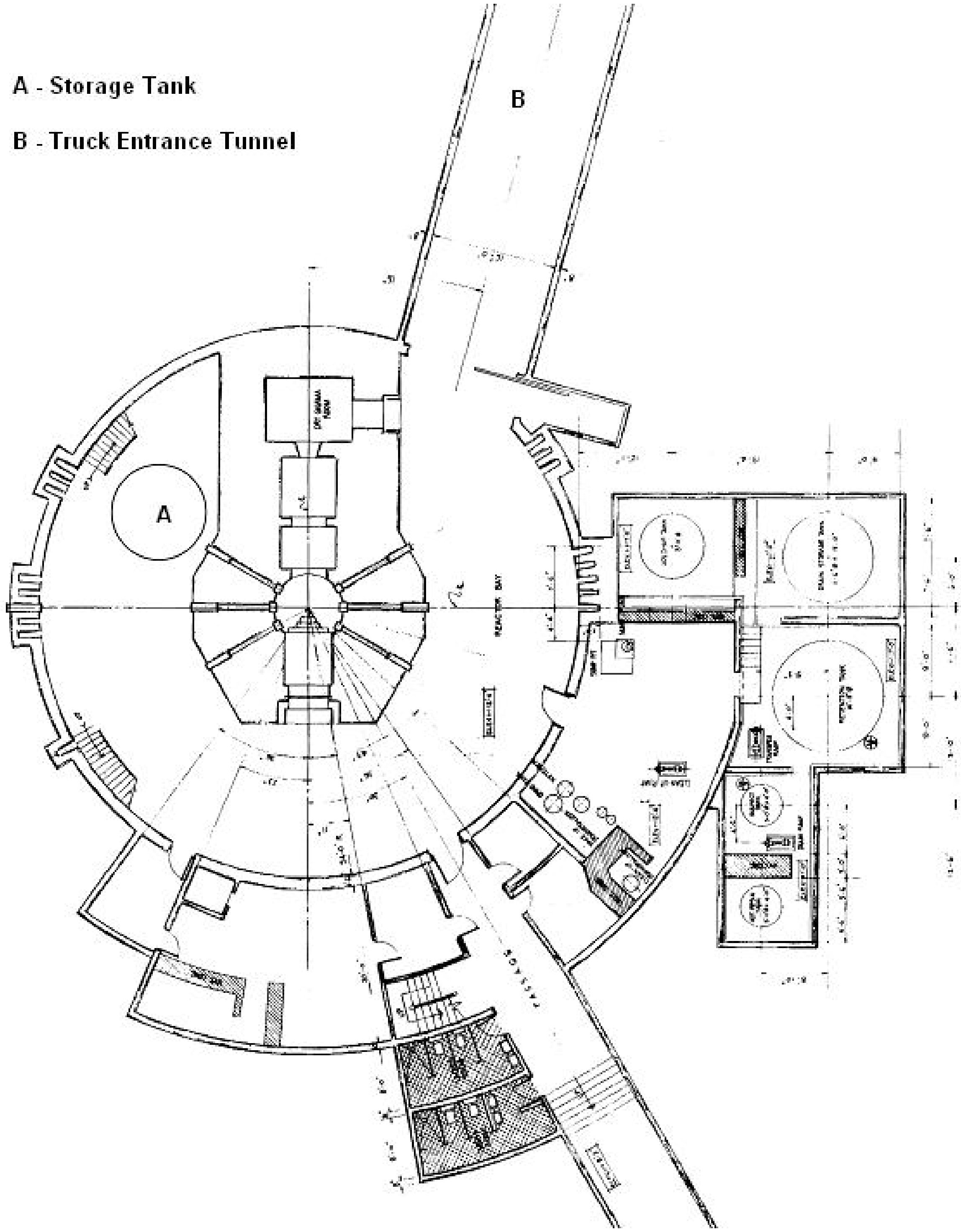

FIG. 3. The floor plan of the Reactor Building.

FIG. 4. The storage tank.

The reactor bay floor was 3.8 meters below ground level, and could only be accessed by heavy equipment through a truck entrance ramp. The ramp had a slope of about 10% for a run of about 32 meters, then made a turn into a tunnel with a slope of about 12% and a run of about 27 meters. The Truck Entrance Doorway to the Reactor Bay was 3.4 meters wide and about 4 meters high.

The reactor building had a crane with a nominal rating of 10 000 pounds (9 tonnes), but it was over 35 years old at the time of the spent nuclear fuel transfer, and it had never being used to lift anything heavier than 5 tonnes. Therefore, it was decided to consider 5 tonnes as the working limit of the building crane.

5. The fuel transfer equipment

The NAC-LWT cask was designed to be raised vertically and submerged into a fuel storage pool for direct underwater loading of the spent nuclear fuel. Unfortunately, that was not an option feasible for the fuel of the PRR-1 reactor, because the weight of the cask greatly exceeded the rating of the building crane and there was no space inside of the building for a mobile crane with the required capacity. Then it was decided to loadn the NAC-LWT cask outside of the reactor building by using a system of transfer casks specially devised by NAC.

5.1. The DTS cask

The DTS (Dry Transfer System) cask, shown in Fig. 5, was designed to carry a single basket of research reactor spent fuel and to mount it on top of an upright NAC-LWT cask that had been fitted with a valve shield adaptor, as shown in Figs 6. The DTS cask is loaded and unloaded with a single basket through its bottom, using an internal grapple and lifting mechanism run by compressed air. A basket can contain up to 7 MTR-type fuel elements, such as the PRR-1 fuel.

FIG. 5. The DTS cask, reclining on its transport skid, being unpacked.

FIG. 6. The DTS cask being mounted atop an upright NAC-LWT cask.
Note the valve shield adaptor between the DTS cask and the NAC-LWT cask.

The DTS cask could directly load a fuel basket if the cask could be positioned over the storage tank. However, the DTS cask weighed 7 tonnes and was too heavy to be lifted by the building crane, and an ITS (Intermediate Transfer System) cask was used.

5.2. The ITS cask

The ITS cask was a top-loading cask designed to mate with the DTS cask in much the same way as the NAC-LWT cask, but it is much smaller than the NAC-LWT cask and can also contain only one basket at a time (Fig. 7).

FIG. 7. The ITS cask on arrival at the work site.
Note the valve shield on top of the cask.

The ITS cask weighed about 10 tonnes, but it could be used with the building crane because it had a separable inner cask called "Inner Shield", which weighed about 2 tonnes (Fig. 8). The building crane was capable of moving the Inner Shield with a fuel basket inside of the Storage Tank, where the basket could be filled with spent fuel under water.

FIG. 8. The inner shield being lifted away from the ITS cask.
Note the lid on the inner shield.

6. The rented equipment

6.1. The mobile crane

A crane was needed to unload the casks and tools from the ISO-standard containers in which they were shipped, raise the NAC-LWT cask to the upright position, move the DTS cask between the ITS and NAC-LWT casks and do general positioning of heavy items. A mobile construction crane with an 80-tonne lifting capacity, together with operating personnel, was rented for the purpose (Fig. 9).

6.2. *The forklift*

It was decided to use a rented forklift to move the ITS cask between the interior of the Reactor Building and the outdoor loading area. The forklift should have enough carrying capacity to safely transport the ITS cask up and down the Truck Entrance Ramp, but it also had to fit through the narrow Truck Entrance Doorway. A forklift with a 10-tonne capacity fitted easily through the doorway. However, when it was tested with the empty ITS cask, the driver felt that he was at the edge of losing control while travelling through the ramp. Then it was decided to use another rented forklift, shown in Fig. 9, with a rated carrying capacity of 18 tonnes. The larger forklift fitted through the doorway with only a few centimeters to spare, but easily carried the ITS cask up and down the ramp.

FIG. 9. The 80-tonne mobile crane and the 18-tonne forklift.

6.3. *An air compressor*

A source of compressed air was needed to run the mechanism of the DTS cask, and later to prepare the NAC-LWT for shipment. The reactor building had a compressed air supply system, but running an air hose the distance to the loading area might have reduced the working pressure too much, so a large engine-driven mobile air compressor, normally used to run jackhammers and air tools in construction, was rented.

7. Preparation for fuel transfer

A paved area of approximately 500 square meters, shown in Fig. 10 and just off the truck entrance ramp was chosen as the area where the NAC-LWT casks were to be loaded. Careful study of the loading procedures and the work area revealed a few shortcomings of the site that had to be fixed. Part of a traffic island on the road to the work area was removed to allow the heavy equipment to pass. Some of the iron grates over the drainage gutters across the truck entrance ramp were replaced with solid concrete blocks to bear the weight of the loaded forklift that was going to enter the reactor building.

The crane in the reactor building and its accessories like lifting slings were verified to be in safe and reliable working condition. The reactor's demineralizer, needed to supply purified water to test the NAC-LWT casks after loading, was also verified to be working properly. It was also verified that the water in the storage tank was not contaminated with radioactivity from a leaking fuel element and was safe to work with it.

Other details were not overlooked. Those included providing strong underwater lights for the atorage tank and verification that the fuel grapples were in good working condition. Additional security personnel were added and night lighting was installed in the loading area.

FIG. 10. The loading area, with the NAC-LWT casks in their shipping containers, with impact limiters installed. The truck entrance ramp is at the left, hidden by foliage

8. The fuel transfer

The transfer of the spent fuel from the storage tank to the NAC-LWT casks was done over five days, Monday to Friday, 8-12 March 1999. For safety, the work was only done during daytime, but there was a tight schedule to meet the arrival of the ship schedulled to carry the spent fuel away. A single ship was iused to take the spent nuclear fuel from 4 countries in one trip, and the Philippines was the third on the sailing route.

8.1. Raising the NAC-LWT casks

The two NAC-LWT casks arrived in individual ISO-standard 20-foot shipping containers on 3 March. The DTS and ITS casks and other tools arrived in another two 20-foot shipping containers on 6 March. Work started in the morning of 8 March, a Monday. Unpacking the shipping containers took half of the day, after which setting up the baseplates of the NAC-LWT was started. Heavy rain started falling around noon, a rare occurrence in March which is in the dry season in the Philippines. The shipment of spent fuel was deliberately scheduled in the dry season to avoid the typhoons of the rainy season, which might have made the outdoor work dangerous and might also have made sea transport difficult. Fortunately no rain was to fall during the rest of the week. The baseplates were completely set up by Tuesday morning, and the raising of the NAC-LWT casks was started. The NAC-LWT casks were completely raised, as shown in Fig. 11, by Wednesday

FIG. 11 The two raised NAC-LWT casks, with the reactor building in the background. The cask at the left has the valve shield adaptor installed.

8.2. Taking the ITS cask inside the reactor building

The ITS cask was loaded on the 18-tonne forklift. The blade slots at the skid of the ITS cask were too narrow for an 18-tonnene forklift, but a field solution was found by securing the blades on the skid with straps, chain and turnbuckles.

Because the ITS cask can take only one basket with 7 fuel elements per trip, the ITS cask had to make many trips between the reactor building and the NAC-LWT casks. On each trip, the forklift proceeded backward down the truck entrance ramp and tunnel because there was no room to turn around in the reactor building. It would also not have been safe to carry such a heavy load in the front while going down an incline (Figs 12 and 13).

FIG. 12. The forklift carrying the ITS cask backward down the truck entrance ramp. The forklift is about to enter the truck entrance tunnel.

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FIG. 13. The forklift entering the reactor building at the end of the truck entrance tunnel. Note the extremely tight fit at the doorway.

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8.3. Lifting the inner shield into the storage tank

Inside of the reactor building, an empty fuel basket was loaded into the inner shield of the ITS cask and the 68-kilogram lid was placed on the inner shield. The inner shield was then lifted by the building crane into the storage tank (Figs 8 and 14).

FIG. 14. The inner shield being lowered into the storage tank.

8.4. Loading fuel elements into the basket in the inner shield

Under water and inside of the storage tank, the lid of the inner shield was taken off, exposing the fuel basket. Spent nuclear fuel elements were then taken out of their racks and placed inside of the basket, with the inner shield still suspended from the building crane (Figs 15 and 16).

After the fuel basket was completely filled, the lid of the inner shield was replaced and the inner shield was lifted out of the storage tank and returned into the ITS cask waiting in the reactor bay. The lid was then taken off the inner shield remotely with the crane, with personnel avoiding direct exposure, and the valve shield of the ITS cask was closed. The forklift then took the ITS cask out of the reactor building back to the NAC-LWT loading area.

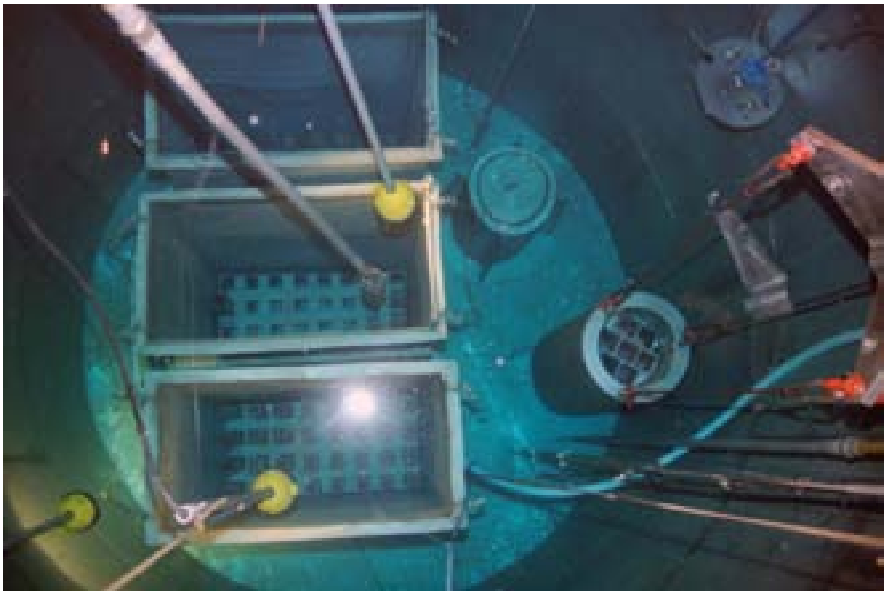

FIG. 15. A fuel element being withdrawn from its storage rack. The waiting inner shield with a fuel basket inside is at the right. The suspended lid of the inner shield is at the upper right.

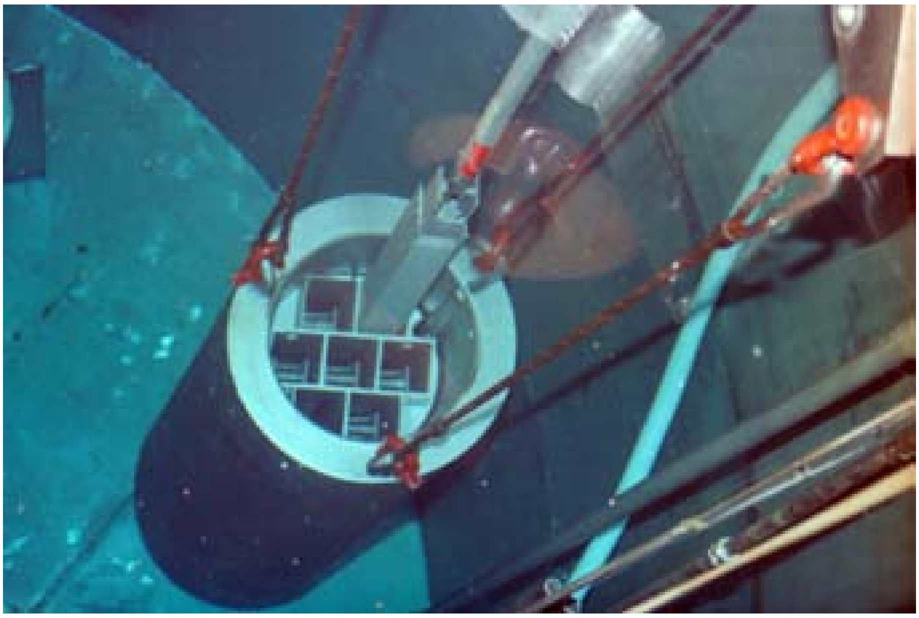

FIG. 16. A fuel element being loaded into the basket in the inner shield.

8.5. Moving the loaded basket into the NAC-LWT cask using the DTS cask

In the NAC-LWT loading area, the 80-tonne crane mounted the DTS cask on top of the ITS cask. The loaded basket was transferred into the DTS cask using the grapple and lifting mechanism inside of the DTS cask. The crane then transferred the loaded DTS cask to the adaptor on top of the upright NAC-LWT cask. The valve shields were opened, and the DTS mechanism lowered the loaded basket into the NAC-LWT cask. The grapple was then withdrawn and the NAC-LWT valve shield was closed. The DTS and ITS casks were ready for the next basket (Figs 17 and 18).

*FIG. 17. The DTS cask carrying a fuel basket from the ITS cask to the NAC-LWT cask.
The forklift had just transported the ITS cask from the Reactor Building.*

FIG. 18. The DTS cask, suspended from the crane, transferring a loaded fuel basket into the NAC-LWT cask. Note the other NAC-LWT cask laid down in its shipping container, already full. Its empty baseplate is at the left, just in front of the crane.

9. Preparing the casks for transport

All the fuel elements were loaded into the NAC-LWT casks by Thursday, but the loaded NAC-LWT cask still had to pass several tests before the transport could be approved. The fuel leak test was conducted after the closure lid of the cask had been installed, with the cask still on the upright position. Hoses were connected to the drain and vent ports of the cask. Purified water provided by the reactor's demineralizer was pumped to the cask through the drain port until the vent port dumped water, indicating the cask was full. A sample of the LWT cask water was taken immediately, another between 4 and 8 hours later, and a third 12 hours later. The samples were analyzed for Cs-137 activity. The regulatory limit for the increase of activity between the first and third samples was 1 325 dpm/ml.

The NAC-LWT casks loaded at PNRI met the limit by very wide margins. One cask had an increase of 3.77 dpm/ml and the other 12.74 dpm/ml. None of the samples actually had a Cs-137 activity higher than 18.5 dpm/ml, or 8.33 nanocuries per liter. This low level was not unexpected, as none of the PNRI fuel assemblies was known to have failed cladding. It was assumed that the small Cs-137 contamination that was detected in the samples probably came from previous use of the casks and not from the PNRI spent nuclear fuel.

After the 12th-hour sample had been taken, the water in each cask was ejected by introducing compressed air into the vent hose. The water was collected in a waste tank for proper disposal. The cask was then laid down on its shipping container. The cask interior was dried by attaching a vacuum pump and running it until the moisture level was negligible (Fig. 19).

Each cask was then filled with helium and leak tested with instrumentation attached to ports in the closure lid of the cask. Helium was supplied from a pressure cylinder equipped with a regulator. After the leak test, the helium atmosphere was retained inside of the cask for transport (Fig. 20).

The radiation dose rates at the exterior of the NAC-LWT casks as well as the amounts of removable surface contamination were measured, to verify that the casks complied with transport regulations.

The first cask completed testing on Thursday, and the second NAC-LWT cask completed testing on Friday. The impact limiters were put back on the NAC-LWT casks and the shipping containers were closed. The other casks and the tools were packed into their own shipping containers. All four containers were ready for shipping by the end of Friday (Fig. 21).

FIG. 19. Vacuum-drying the interior of the NAC-LWT cask. The vacuum pump is at lower right. Note the helium tanks at left, used later to leak-test and fill the NAC-LWT cask for transport.

FIG. 20. Leak-testing the NAC-LWT cask.

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FIG. 21. Spent-fuel shipping containers ready for transport.

10. Transporting the spent fuel to the ship

The loaded containers were shipped back to the U.S.A. through the port of the city of Manila. According to the original planning, the containers were to be taken by public roads through the center of the city to the port, about 15 kilometers from the PNRI. The plan was to do the transport at around midnight, to avoid heavy traffic and unwanted attention. Packing was done by Friday afternoon and the ship was already at the harbor, but the trip was not to be done that night at the advice of the police. Friday was the end of the work week and traffic would be very heavy deep into the night. The loaded containers were taken to the port at the next night.

Extra guards were placed around the loaded containers while waiting during Friday night and during Saturday. At about 12:00 P.M. Saturday, the technical staff required for the transport and the police escort assembled at the reactor site. There were nearly a dozen vehicles in the convoy, including the four loaded tractor-trailers, two spare tractors, and police cars. There were also a number of motorcycle police to control traffic. The convoy left the PNRI at about 1:30 A.M. Sunday, 14 March 1999 (Fig. 22). Travelling at a slow but steady 20 to 30 kilometers per hour, with the motorcycle police clearing the way through very light traffic, the convoy reached the dock in less than an hour.

At about the same time as the convoy was leaving the reactor site, the ship was moving from the harbour to the dock with its hatches already open. Customs formalities were prearranged and over quickly. The four containers were loaded aboard as soon as they arrived beside the ship (Figs 23 and 24). The PNRI Director signed papers with the ship captain formalizing the transfer of the spent fuel. The ship left after spending less than 3 hours on the dock, closing its hatches underway.

FIG. 22. One of the spent-fuel containers leaving the PNRI.

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FIG. 23. The port crane about to grab one of the spent-fuel containers.

FIG. 24. One of the spent-fuel containers going aboard ship.

11. Conclusion

The transfer of the spent fuel elements from the Storage Tank to the NAC-LWT casks was accomplished on schedule without incident and injury.[4] This was due to the months of careful preparation for the task, as well as the skill of the people who did the job, including the people from NAC, PNRI, and the local and foreign subcontractors. Some of those people are shown in Fig. 25.

FIG. 25. The loading crew posing in front of the last shipping cask to be filled.

REFERENCES

[1] YOSHISAKI, M.B., LEOPANDO, L.S., A Report on the Transport of MTR-Type Spent Fuel Assemblies of the Philippine Research Reactor, Philippine Nuclear Research Institute, Quezon City, Philippines (March 1999). Internal report.

[2] GENERAL ELECTRIC COMPANY, Final Safeguard Report for Philippine Open Pool Reactor, GEAP-3383, San Jose, California. (1 April 1960).

[3] NAC INTERNATIONAL, Safety Analysis Report for the NAC Legal Weight Truck Cask, Revision 23 (November 1998).

[4] LEOPANDO, L.S., The Philippine Experience in Shipping Spent Research Reactor Fuel (May 1999), Presented at the IAEA Interregional Training Course on Technical and Administrative Preparations Required for Shipment of Research Reactor Spent Fuel to its Country of Origin, Argonne National Laboratory. (May 1999).

Portuguese experience on return of spent fuel to the United States of America

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Abstract. In 1999 Portugal declared its interest to participate in the Foreign Research Reactor Spent Nuclear Fuel Acceptance Programme (FRRSNF) of the United States Department of Energy (DOE). A commitment was made to stop using the current HEU fuel and return it to the US until 12 May 2009. The LEU assemblies used from 1961 until 1987 were returned to the US still in 1999. Prior to the shipment all assemblies were visually inspected for corrosion and sipped for determination of fission product leakage. Limitations on the floor loading of the reactor building and on the capacity of the crane prevented loading the transport cask inside of the containment building. Instead, a structure had to be built outside and the assemblies transferred individually. An excellent collaboration with the Ministry of Defence allowed the use of military facilities for the shipment. In this paper we review the actions taken, as well as the lessons learnt

1. Introduction

The Portuguese Research Reactor (RPI) is owned and operated by "Instituto Tecnológico e Nuclear" (ITN). It was built by AMF Atomics during the period of 1959-61 and its design follows closely the one of the Battelle Research Reactor. The activities currently underway in the RPI cover a broad range from irradiation of electronic circuits [1] to calibration of detectors for dark matter search [2], with a strong emphasis on neutron activation analysis [3].

In 1999, Portugal declared its interest to participate in the FRRSNF. A commitment was made to stop using HEU and return this fuel until 12 May 2009. The LEU assemblies used until 1987 were shipped back to the US in the summer of 1999. The RPI is now undergoing conversion back to LEU fuel [4] and is getting ready for the shipment of the HEU fuel which will take place on a date to be agreed with the management of the FRRSNF acceptance programme.

2. Preparation for the shipment

2.1. Data collection

DOE's receipt process requires the fuel proposed for shipment to be classified based on the materials of construction, physical dimensions, decay heat load, dose rate, fissile content, selected isotope content and physical condition [5]. These parameters are determined from the fabrication drawings and specifications, the fabrication quality control records, and the operating history of the fuel assembly. RPI had three types of LEU assemblies: standard, control and partial assemblies. All assemblies were of U-Al alloy, 1100 grade aluminum cladding, enriched to 19.83 ± 0.1% in ^{235}U. Table 1 summarizes the main data on these assemblies. All quoted uncertainties represent $1\cdot\sigma$ values.

The RPI produces an annual integrated power up to 60 MW·d and thus the uranium burnup rate is relatively small. Core management is reduced to shuffling of assemblies, in order to get comparable burnup levels, and addition of fresh assemblies whenever necessary. A typical LEU core configuration is shown in Fig. 1. It had 19 standard assemblies and 5 control assemblies [6].

TABLE 1. DESCRIPTION OF THE 3 TYPES OF LEU FUEL ASSEMBLIES

	LEU standard	LEU control	LEU partial
Nr. of assemblies	28	9	2
Nr. of plates/assembly	12	6	6 + 6 Al dummy
Overall weight (g)	5016	4814	4634
Total U weight (g)	906.1 ± 7.9	456.3 ± 3.8	454.0 ± 3.2
Total U-235 weight (g)	179.7 ± 1.5	90.5 ± 0.8	90.0 ± 0.9

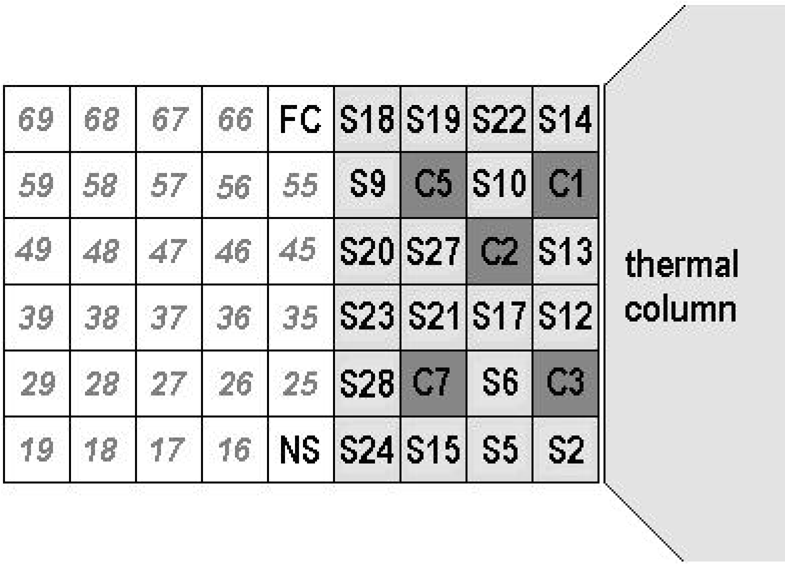

FIG. 1. Typical LEU core, with reference 7-I. Legend: FC=fission chamber for source range channel; S_i=standard assemblies; C_i=control assemblies; NS=Sb-Be neutron source.

Out of the 28 standard assemblies, 17 were put into use in April 1961 and the remaining later, as necessary. By the end of 1967 all standard assemblies were in the pool. Assembly S11 was unloaded in July 1978 as it presented a small pit corrosion in the cladding of one outer plate. As for the control assemblies, 5 of them were put into use in April 1961. The remaining 4 were loaded into the core in July 1978. Assembly C1, used for the regulating rod, remained in place throughout the whole service life of the LEU fuel. As for the partial assemblies, P1 was put into use in April 1961 and was several times introduced and removed from the core. Assembly P2 was put into use in December 1972.

2.2. Determination of fissile content and decay heat

The fissile content was determined using data from ANL/RERTR/TM-26 [7], Table 4 (MTR Fuel, 19.75% enrichment). The use of this data requires interpolation in both the burnup and the initial mass, as it is evident from Fig. 2, which plots the data for the total amount of Pu, as function of burnup and initial ^{235}U mass. A FORTRAN program was written to do the interpolations using the divided-difference technique [8].

Decay heat values were determined using the PHDOSE code [9] taking into account the burn-up and decay for each assembly. Due to the relatively long decay time (at least 11 years) the only relevant activity was due to ^{137}Cs, with average and maximum values of 66 Ci and 90 Ci, respectively. The total ^{137}Cs inventory of the 39 assemblies was 2 570 Ci. The average and maximum decay heat values were 0.28 W and 0.38 W, respectively. The total decay heat value for the 39 assemblies was 11 W. Tables 2 and 3 compile the average fissile content and relevant data for decay heat values, for the 3 types of assemblies.

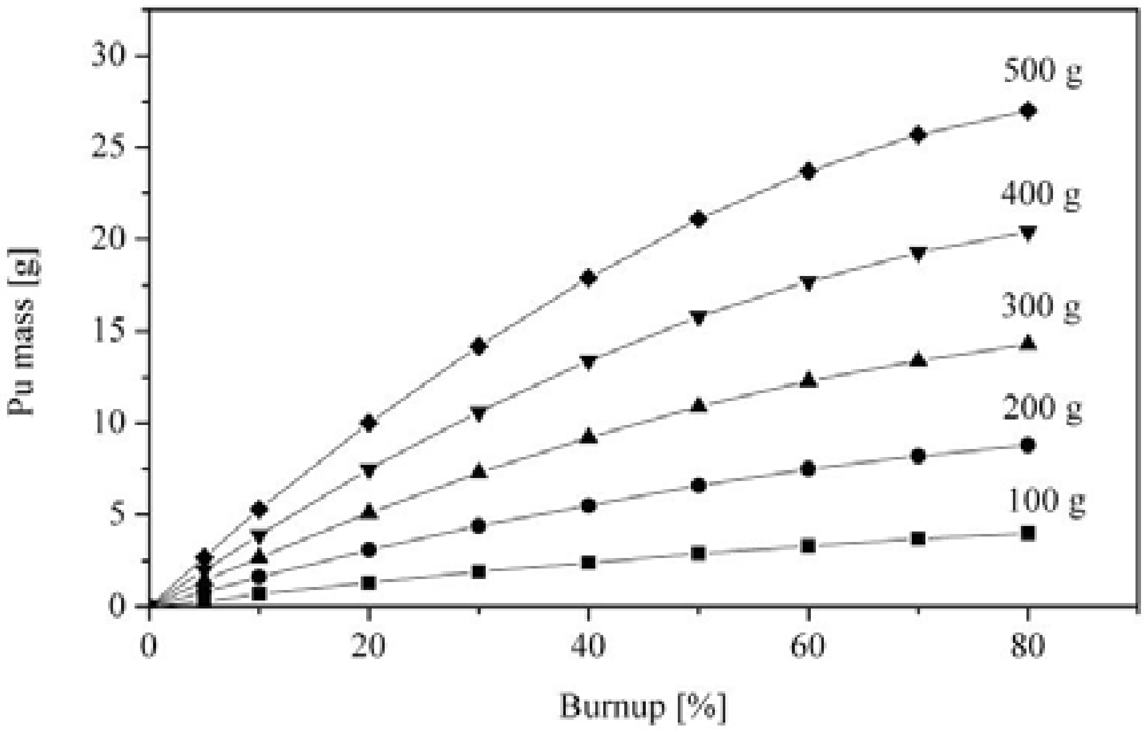

FIG. 2. Total amount of Pu for MTR fuel, 19.75% enriched in ^{235}U, as function of burnup and initial ^{235}U mass. Data from ANL/RERTR/TM-26. The lines were drawn to guide the eye.

TABLE 2. AVERAGE FISSILE CONTENT FOR THE 3 TYPES OF LEU ASSEMBLIES

	LEU standard		LEU control		LEU partial	
	Average (g)	Uncertainty (±g)	Average (g)	Uncertainty (±g)	Average (g)	Uncertainty (±g)
Total U	866.8	12.2	442.0	7.6	445.3	6.3
Total Pu	3.1	0.6	1.0	0.7	0.7	0.1
Total Np	0	0	0	0	0	0
U-235	137.8	8.6	74.9	9.9	79.9	3.0
U-236	6.6	1.3	2.6	1.6	1.0	0.1
Pu-239	2.7	0.5	0.9	0.6	0.7	0.1
Pu-241	0	0	0	0	0	0

TABLE 3. DATA RELEVANT FOR DECAY HEAT AND DOSE RATE FOR THE 3 TYPES OF LEU ASSEMBLIES

	LEU standard		LEU control		LEU partial	
	Average	Uncertainty (±%)	Average	Uncertainty (±%)	Average	Uncertainty (±%)
Cs-137 (Ci)	81.7	21	27.0	64	20.3	21
Dose (rem/h)	29.4	21	8.0	84	7.0	20
Decay heat (W)	0.34	21	0.11	64	0.09	22

2.3. *Inspection and sipping procedures*

About 60% of the LEU assemblies were in the pool since 1961, i.e., for 38 years, initially raising some concerns about their condition. Therefore, all assemblies had to be visually inspected for corrosion and sipped for the determination of the eventual fission product leakage rate before the shipment [10].

The assemblies had always been stored in 6061 Aluminium alloy racks placed within the reactor pool. During the refurbishment of the reactor (1987/90) the LEU assemblies were temporarily kept inside of a double-walled container, connected to the water demineralizing system [11]. In the late nineties the assemblies were transferred to two new 6061 Aluminium alloy racks, each one with the capacity to store up to 20 assemblies.

The water demineralizer system of the RPI runs permanently which means that about 108 m^3 of water run daily through the system (nearly ¼ of the pool water). On the other hand the pH and conductivity have always been maintained around 6.0 and < 1 μS/cm, respectively, well within the recommended range for aluminide fuels [12],[13]. The RPI is currently using Amberjet 1600-H cation exchange resin and Amberjet 4400-OH anion exchange resin, from Rohm & Haas. It previously used Amberlite IR120-H and IRA410-Cl resins. The pH is measured using a Metrohm 713 meter and the conductivity using a Metrohm 712 meter. These parameters were recorded 80 times in 2005, from which one obtains average values at the pool surface of 5.6 ± 0.2 (pH) and 0.93 ± 0.05 μS/cm (conductivity).

For the visual inspection all assemblies were individually taken to an examination station, consisting of a table and an extension, which was anchored to the wall of the pool. The table with 1.5 × 0.8 m in size, was located 2 m deep in the water and was used to place the assemblies horizontally and rotate them to have a complete external examination. The extension part of the examination station was suspended from the side of the table. It was used to place the assemblies vertically, as a transit point between storage and washing, and also as an observation point to look for eventual obstructions between plates. Photographs were taken using a digital camera, model Olympus C-1000L with 1 024 × 768 pixels resolution. The distance between the camera and the fuel assembly was approximately 2.5 m. A plexiglass plate was used to reduce the reflex on the water surface. For each assembly photos were taken showing i) one of the side plates with the identification of the assembly, ii) convex outer plate (plate 12) top and bottom sections, and iii) concave outer plate (plate 1) top and bottom sections. Additional photos showing details were taken whenever necessary.

No nodular corrosion or exposed pits were found. Only minor problems, namely:

- Several assemblies had superficial scratches in plate 12 (convex);
- Two assemblies had small deformations on the top of plate 12, above the fuel region;
- One assembly S1 had a lump in the corner of plate 12 with a side plate; this lump was partially removed with a Teflon plate attached to an aluminum handling tool.

Once the visual examination was completed, each fuel assembly was placed in the washing and sipping system. This system, shown in Fig. 3, consists essentially of an aluminum pipe (3 m long, 153 mm internal diameter), closed on the botton, and which can be moved up and down along a guide. The top end of the guide was fixed to the top of the pool wall, while the bottom was inserted in the orifices used to place spent fuel for the gamma irradiation facility. The pipe is placed in the lower position of the guide for insertion and removal of the fuel element, and it is raised to the upper position for the remaining operations. When it is in the upper position, the top of the pipe is above the water level of the pool and the water in the pipe can be replaced with water taken from the demineralized water supply system, and homogenized by bubbling air through it. The admission of water and air are made through a plastic tube connected to the bottom part of the pipe. This system is similar to the one previously developed by IPEN, São Paulo, Brasil [14].

The procedure was to place the assembly inside the pipe, raise the system to the upper position and wash the assembly with an amount of water of at least three times the volume of the pipe, i.e. 150 liters of water, which meant leaving the water running for about 10 minutes. After washing, the water kept in the pipe was homogenized by passing compressed air through it for about 5 minutes. At this stage a sample of about 1.5 liter was taken for background determination. The system was then left in the upper position, isolated from the pool water, for a resting period of at least 4 hours (some assemblies were left overnight). At the end of the resting period the water was homogenized again and another sample of 1.5 liter was taken.

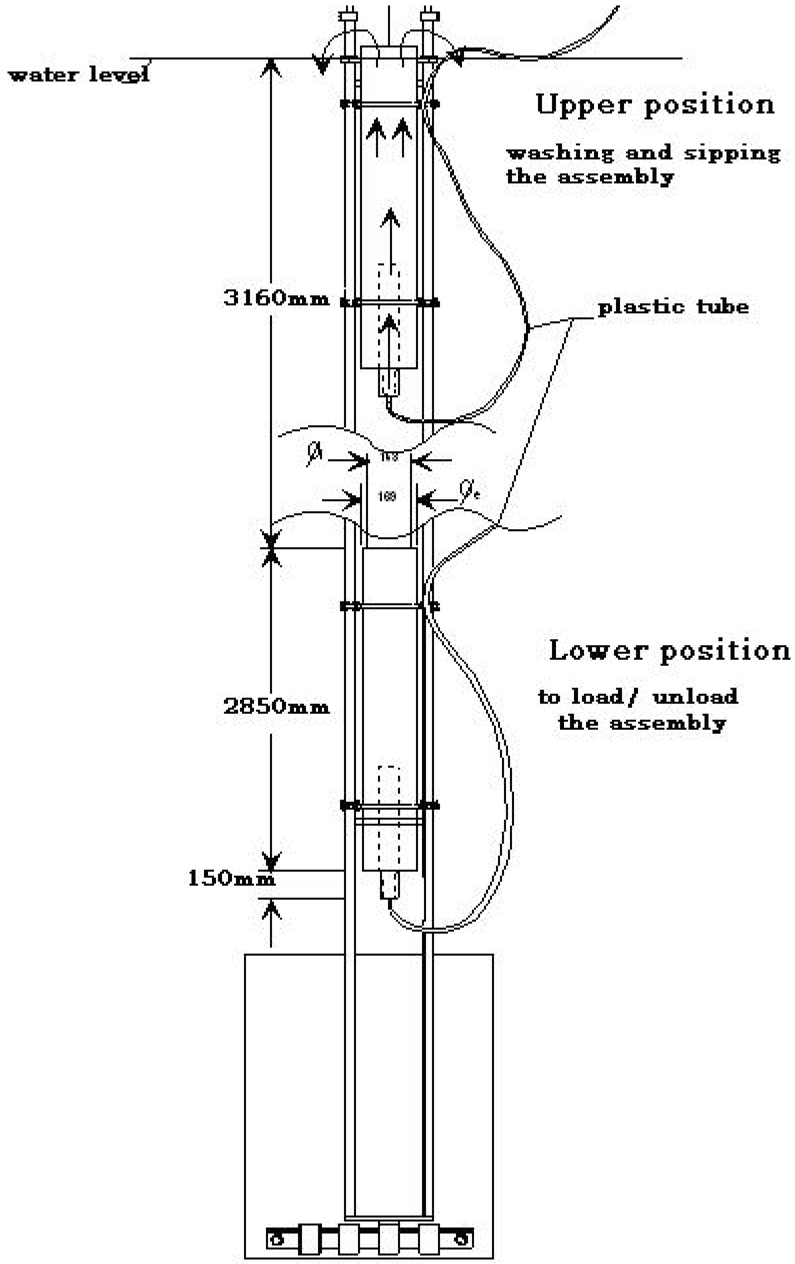

FIG. 3. Washing and sipping system. The upper and lower positions are shown.

To measure the activity of the water a HPGe detector (25% efficiency relative to NaI) was used. The detector was placed inside of a cylindrical Pb piece, assuring a 5 cm thick shield around the detector. The detector viewed the sample through a collimator with a 4 cm diameter opening. The complete system was installed in the side of a clean handling cell with a 5 cm thick Pb wall. One liter samples were used in all cases. The calibration of the measuring system was performed with a reference sample, also with 1 liter volume, prepared from a calibration solution. This reference sample was measured several times to control the long term behavior of the measuring system. The water samples were measured for at least 3 hours. The analyses of the collected spectra were done using a commercial code (SAMPO 90, from Canberra), which determines the intensities (in counts per second) of the peaks and identifies the isotopes present. In several cases observation of the spectra has shown that there were small increases in the counting of the channels in the region of the 137Cs peak not identified by the program. It was therefore decided to perform a manual determination of the excess counting in the region of the 662 keV peak for every spectrum, calculating both a net peak counting rate and its uncertainty. The net area was calculated using a channel by channel background subtraction assuming a linear background variation along the peak region.

Normally two fuel assemblies were tested per day. One assembly was taken to the examining station in the morning, examined, photographed, installed in the washing system and left there to rest after the background sample was taken. Around the middle of the afternoon the water was homogenized and sampled and the assembly was removed and replaced by another one which would rest in the pipe overnight. The gamma spectrometry system was running continuously and normally two spectra were collected: one during the normal period of operation of the reactor and another overnight. The relevant sipping results are compiled in Table 4.

TABLE 4. LEAKAGE RATE RESULTS. ONLU NON-ZERO VALUES ARE SHOWN

Assembly	Leakage rate 1st run (Bq/h)	Uncertainty (±%)
S1	177	10
S5	113	37 a)
S17	24	20 a)
S26	114	14
C1	11	13

a) Photopeak area and error determined manually.

Only 5 assemblies have shown 137Cs activity above background. The washing and sipping procedures were repeated for those assemblies. In all cases the second and following samples yielded lower leakage values. This seems to indicate that the high values measured in the first sample were due to surface contamination that was essentially removed during the first washing and sipping period. Specifically for fuel assembly S1, the first measurement gave the value 177 ± 10% Bq/h, while the following six averaged 24 ± 5% Bq/h. No values above background were found for the assembly taken out of the core in 1978.

2.4. Preparation of a loading station

The cask selected for the transport was a Transnuclèaire IU04 with an AA267 basket capable of storing up to 40 assemblies. The IU04 cask weights 19 t and has a 1.6 m diameter. The low capacity of the reactor crane (10 t) and the characteristics of the floor at the service entrance which was planned for a load inferior to 7 t/m2, prevented that the cask could be handled inside of the reactor building. Instead, a loading station had to be erected outside of the building, about 10 m away from the service entrance [15].

The loading station is shown in Fig. 4. The cask rested on I-beams placed inside of a basin connected to the radioactive waste piping. The whole system was resting on a concrete support and was covered with plastic covers to guarantee that any spilled water would go to the basin. The cask was provided with a loading skirt, adaptable to its upper part, which was used to create a water basin where the assemblies could be manipulated. All manipulations inside of the cask were done from a platform 2.7 m above of the concrete support. The operators were protected by the water inside of the cask and the loading skirt, whose Pb shield (8 cm thick) extended from the top of the cask until 0.2 m above the platform. Since the shipment would take place during summer, no special measures were necessary regarding meteorological conditions. In any case, a cover for the whole structure was on stand-by and could be mounted within a few minutes.

To carry out the operations inside of the reactor building a supplementary structure was placed on the NW side of the pool, to secure the fuel handling tools. The platform previously used for the visual inspection of the assemblies was then used as an intermediate station in the movement of the fuel assemblies out of the pool. The region of the floor close to the service entrance was reinforced with 1 cm thick steel plates to better distribute the loading.

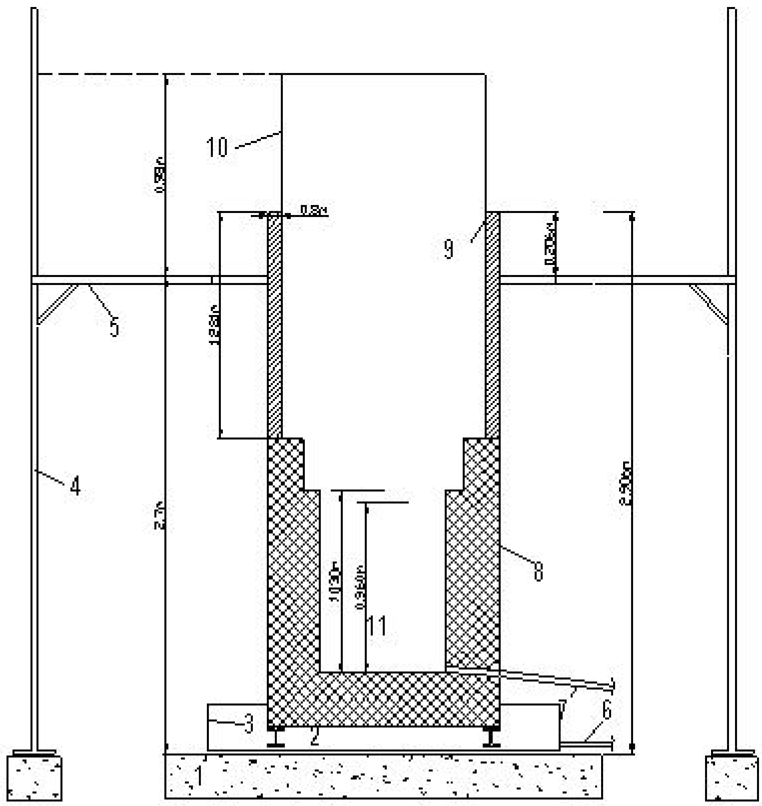

FIG. 4. . Loading station built around transport cask. Legend: 1=concrete base, 2=support for cask, 3=basin for liquid waste, 4=support structure, 5=working deck, 6=basin outlet, 7=cask outlet piping, 8=transport cask, 9=biological protection, 10=loading skirt, 11=basket.

Loading an assembly on the transport cask required a series of steps of which the most relevant were the loading and unloading of the transfer cask (6 t weight), the identification of the assembly and its placement in the AA267 basket, according to a previously established loading diagram. Each assembly was taken from the storage racks to the auxiliary platform where its identification was confirmed. The transfer cask, held by the reactor crane, was placed above the assembly with its bottom about 1 m below the water surface. Then the assembly was raised into the transfer cask by means of a cable provided with a hook which grabbed the assembly by the handling rod. The admission of the assembly was done via a small door at the bottom of the cask, which was closed once the cable was secured.

Before leaving the pool area the surface of the transfer cask was dried to minimize water spillage. The cask was then lowered to the entrance level of the building and loaded on a forklift, which was used as a shuttle between the reactor hall and the transport cask. A special basin was constructed to ensure the fixation of the cask during the movement of the forklift and to collect any spilled water. Outside of the reactor building all the handling was performed with a mobile crane. Each assembly was put initially in a temporary position above the AA267 basket, where its identification was rechecked before going to its final position. The transfer of control assemblies required the replacement of the hook by a piece capable of handling these assemblies, with the stainless steel shock absorber still in place. These were only removed after the placement of the assembly in the IU04 cask. The transfer cask was then used to carry the absorber back to the pool. The 39 assemblies were transferred within two days.

The radiation exposure of the staff had to be estimated in advance in order to optimize the handling procedure. Since this was the first time that spent fuel was shipped from the RPI no previous data on the dose rates occurring during this type of operation was available. MCNP simulations confirmed that the main exposure was expected to occur during the transfer of the assemblies from the transfer cask to the transport cask, since shielding at this stage was provided only by the water-filled skirt and the biological shield [16]. These simulations were useful to optimize the loading procedures. No dose values exceeding the 0.20 mSv minimum recordable value for individual dosimetry were detected for the operators involved in the operation.

Once loading was completed, water samples from the transport cask were taken at regular intervals. A maximum 137Cs leakage rate of 63 Bq/h was determined, which was well below the limit imposed by DOE for acceptance of the shipment.

The position of the transport cask outside of the reactor building created difficulties on the use of the Cherenkov viewing device during the safeguards inspection. It was necessary to switch off or cover with black cloths all lights around the cask and wait until about 10 pm for it to be sufficiently dark. After the safeguards inspection, the IU04 cask was closed, the water was drained and the required dryness and containment tests were performed. The outer surface of the cask was finally washed and cleaned in fulfillment of the transportation requirements. Finally it was placed inside of a 20 foot ISO container kept close to the reactor building under permanent surveillance by armed police officers, in addition to the normal security arrangements of ITN.

3. Shipment

Portugal was planned as the last stop of the ship carrying the fuel before leaving to the US. This meant that upon its arrival in Lisbon it would be carrying spent fuel from other facilities. In order to minimize the risk for leaks of information it was decided not to use any of the commercial port facilities in or around Lisbon. The Ministry of Defence kindly offered the use of a naval base located on the south of Lisbon. Access to the naval base was granted to ITN and Customs staff, a food supplier for the ship and the mandatory stevedoring staff.

The truck transport of the cask from ITN to this base was done overnight in a military convoy, via a route that included the Vasco da Gama Bridge, avoiding areas of large population density. The ship had its own crane facilities and left less than 3 hours after the arrival of the convoy. It was escorted by a Navy vessel while in Portuguese waters.

Press coverage was minimal – only one newspaper, "Correio da Manhã", reported in its last page "a depleted uranium shipment of uncertain origin" on the day of the shipment. The same newspaper published the following day, well into the inner pages, a short note from the Ministry for Science and Technology on the shipment.

4. Final notes and conclusions

Initial contacts with DOE were done as early as 1996. The then Supervisor of the RPI participated in the training course organized in Argonne in January 1997. Informal discussions were held during the RERTR meeting in Jackson Hole, Wyoming (US) in October 1997. The information exchanged in this way led to the starting of inspection of the assemblies in 1998. Formal conversations were held shortly after the RERTR meeting in São Paulo, Brasil, in October 1998. Diplomatic Notes were exchanged in early 1999. Appendix A data was sent to DOE on 15 March 1999 and approved on 26 July 1999. The shipment occurred in early August 1999.

Back in 1999, there was no entity responsible for authorizing transports of nuclear fuel in Portugal. Since 2002, the General Directorate for Geology and Energy is responsible for such authorization, while ITN, through its Department for Radiological Protection and Nuclear Safety is responsible for evaluating and inspecting the safety conditions during the transport (Decree-Law 165/2002). The Independent Commission on Radiological Protection and Nuclear Safety, created in 2005 (Decree-Law 139/2005), is an independent supervisory authority.

All operations went essentially as planned. There was a special concern to document with detailed photos all procedures. The shipment occurred at a transition time between two generations of reactor staff, which ended up being positive in the long run, as it represented a chance for the new staff that will handle the next shipment to get experience. The excellent collaboration with the Ministry of Defence was essential for the success of the shipment.

REFERENCES

[1] FRANCO, F.J., et al., Radiation Tolerant D/A Converters for the LHC Cryogenic System, Nucl. Instr. and Meth. A 553 (2005) 604.

[2] GIRARD, T.A., et al., SIMPLE Dark Matter Search Results, Phys. Lett. B **621** (2005) 233.

[3] MACHADO, A., FREITAS, M.C., PACHECO, A.M.G., Relative Response of Biological and Non-biological Monitors in a Coastal Environment, J. Atm. Chem. **49** (2004) 161.

[4] MARQUES, J.G., et al., Core Conversion of the Portuguese Research Reactor: First Results (Proc. 27th Int. Meeting on Reduced Enrichment for Research and Test Reactors, Boston, 2005), Argonne National Laboratory (in print).

[5] UNITED STATES DEPARTMENT OF ENERGY, Guidelines Document on Preparatory Work Prior to Return of Spent Fuel of US-Origin from Foreign Research Reactors, http://www.rertr.anl.gov/FRRSNF/CHEKLIST.html.

[6] RAMALHO, A.J.G., et al., Calibração Inicial do Reactor Nuclear do Laboratório de Física e Engenharia Nucleares (RPI), LFEN-NI.10α, Sacavém, 1961 (in Portuguese).

[7] POND, R. B., MATOS, J. E., Nuclear Mass Inventory, Photon Dose Rate and Thermal Decay Heat of Spent Research Reactor Fuel Assemblies, ANL/RERTR/TM-26 (Rev. 1), Argonne National Laboratory (1996).

[8] CONTE, S.D., DE BOOR, C., Elementary Numerical Analysis – An Algorithmic Approach, 3rd Edition, ISBN 0-07-012447-7, McGraw Hill, Singapore (1981) 47.

[9] POND, R.B., MATOS, J.E., Photon Dose Rates from Spent Fuel Assemblies with Relation to Self-Protection, ANL/RERTR/TM-25, ANL (1996).

[10] RAMALHO, A.J.G., MARQUES, J.G., CARDEIRA, F.M., Some Aspects of the Fuel Management in the Portuguese Research Reactor (Proc. Int. Symposium on Research Reactor Utilization, Safety and Management, Lisboa, 1999), IAEA, Vienna, 2000, IAEA-SM-360/40P.

[11] CARDEIRA, F.M., MENEZES, J.B., Modifications and Modernization of the Portuguese Research Reactor (Proc. Symp. Management of Aging of Research Reactors, Geesthacht, 1995), IAEA, Vienna and GKSS, Geesthacht (1995) pp. 438-450.

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Corrosion of Research Reactor Aluminium Clad Spent Fuel in Water, Technical Reports Series No. 418, IAEA, Vienna (2003).

[13] INTERNATIONAL ATOMIC ENERGY AGENCY, Understanding and Managing Ageing of Material in Spent Fuel Storage Facilities, Technical Reports Series No. 443, IAEA, Vienna (2006).

[14] PERROTTA, J.A., TERREMOTO, L.A.A., ZEITUNI, C., Experience on Wet Storage Spent Fuel Sipping at IEA-R1 Brasilian Research Reactor, Ann. Nuc. Ener. **25** (1998) 237.

[15] RAMALHO, A.J.G., MARQUES, J.G., CARDEIRA, F.M., Return of Spent Fuel from the Portuguese Research Reactor (Proc. 4th Int. Meeting on Research Reactor Fuel Management, Colmar, 2000), European Nuclear Society, Berne (2000) 75-79.

[16] KLING, A., MARQUES, J.G., RAMALHO, A.J.G., Monte Carlo Simulations for the Estimation of Dose Rates during Handling of Spent Fuel at the Portuguese Research Reactor (Proc. MC2000, Lisboa, 2000), ISBN 3-540-41795-8, Springer-Verlag, Heidelberg (2001) 529-534.

First shipment of TRIGA 14MW research reactor highly enriched uranium spent fuel to the United States of America

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Abstract. The TRIGA 14MW Research Reactor has a unique design of core and fuel, with an exceptionally long life. This means long time in-core utilization, leading to a high burnup. The peculiar characteristics of the fuel and reactor facility design made the first shipment dissimilar from the other TRIGA reactors or aluminium plate type shipments. The paper presents the legal framework, regulatory activity, licensing, agreements, contracts, training prior to shipment. The shipment was considered a large coordinated project requiring preparatory activities, resources, national and international cooperation. The overall project time schedule is presented, as well as the diagram of the activities with intervening groups, organization and logistics, the unforeseen events being also mentioned.

1. Introduction

Following the non-proliferation policy, the Department of State and the Department of Energy established a plan for acceptance, receipt and management of Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) under certain conditions, specified in a Record of Decision (ROD).

Authorized Material is FRRSNF which has been discharged from the Research Reactor and which is covered by the eligibility criteria specified in the Environmental Impact Statement (EIS). Adhering to the international non-proliferation policy by practical actions provided for in the treaty, cooperation agreements or contracts, Romania is involved in the general effort to reduce and eliminate Highly Enriched Uranium (HEU) from civilian use. To carry out this goal Romania sustained and performed an intensive activity of cooperation with the IAEA and the Department of Energy over the last decade, resulting in the full conversion of 14MW TRIGA Research Reactor Core and carrying out of the first shipment of FRRSNF.

2. Project of first shipment of TRIGA 14MW research reactor spent fuel from Romania to the United States of America

The completion of the above project was made possible by an extended cooperation between the US Government, Romanian Government and the International Atomic Energy Agency. During the last decade the cooperation addressed the diplomatic level involvement, legal and regulatory activities, technical and commercial relationships. Each shipment project is a complex one, demanding a sustained collaboration among all entities under a tight time-schedule and inter-conditioning activities, based on reciprocal understanding and definition of interfaces and responsibilities.

The shipment project is complex due to the relatively large number of entities involved at different levels of decision, by different geographical areas of development, by the large number of areas of knowledge and expertise of nuclear activities entailed by the project, such as:

- Mechanical Engineering
- Research Reactors Technology
- Nuclear Fuel for Research Reactors
- Nuclear Law – Quality Assurance

- Nuclear Safety
- Physical Protection
- Radiological Emergencies
- Radiation Protection
- Radioactive Waste Management
- Safeguards
- Transport of Radioactive Materials

All shipment projects for FRRSNF are similar in their general structure, using almost the same patterns (i.e. format) of documents for specifications, requirements, authorization, being developed and managed in limited real time. At the same time, each shipment project is distinctive and unique, depending on knowledge resources, engineering experience, local safety and quality culture, capability of communication among project partners and inside every organization involved. Each shipment project became peculiar by typical project situations encountered by local project manager, by unforeseen events and by the aptitude to deal with typical barriers, limited resources or experience, cultural diversity. The common perspective of all shipment projects is to accomplish the return of the nuclear spent fuel at a high level of safety and security in order to reduce the global threat of non-peaceful utilization of highly enriched uranium. To this purpose, the exchange of experience under the auspices of the International Atomic Energy Agency will sustain this activity and will enhance the future project communicating insight about project problems and solutions.

Following this exchange of experience, some guidelines, which have proven successful, can be used for general provisions/specifications of project/contract improvement as a learned lesson which project managers can implement within their responsibilities.

Initially, the entire project had been established for a period of 10 years, encompassing a pre-contractual activity and a contractual one.

Pre-contractual activity started in 1996 and was dedicated to bilateral exchanges of agreements and commitments between Governments and an exchange of technical information, which paved the way to a formal contract for “Terms and Conditions for the Acceptance of FRRSNF” signed in July 1999.

The most important activities of the project are outlined in Table 1.

3. General framework of project development in Romania

3.1. National legislation

The peculiar case of Romania in transition is reflected in the deep modification of national legislation due to:

- new decentralized administrative and economic system
- adherence to the provisions of international practices, agreements and conventions
- integration in enlarged European Union
- development of new bilateral relationships.

In 1996 a new Romanian Nuclear Act on peaceful and safe utilization of radioactive materials and radiation sources was promulgated. Since 1996 the Act had been occasionally amended and updated to incorporate the provisions of conventions and agreements to which Romania adhered. It is worth mentioning the Convention of Nuclear Installations Safety and the Convention for Safe Management of Radioactive Waste and Spent Fuel. A large number of regulations and norms have been promoted by the entities operating in nuclear activities. Several laws and regulations reinforced environment protection. Each type of activity required an environment license released by Regional authorities or by National authority, strengthened by a Government Decree.

TABLE 1 OVERALL PROJECT TIME SCHEDULE

Documents/Activities	Date
Environmental Impact Statement EIS-DOE	1994
Record of Decision ROD-DOE	July 1996
IAEA Training Course on Technical and Administrative preparation for R.R.S.N.F. return	January 1997
Commitment of Government of Romania to: ➢ participate in F.R.R.S.N.F. project ➢ complete conversion of R.R. core to LEU till 12 May 2006 ➢ to not use HEU as fuel in reactor after 12 May 2006	September 1998
Cooperation agreement for peaceful utilization of Nuclear Energy between Governments of USA and Romania	September 1998
Pre contractual activities – first draft	February 1999
"Appendix A" preparation	February 1999
Inspection equipment shipped to Romania	March 1999
Fuel characterization	April 1999 (3 days)
Communication of Government of Romania that the all HEU will be available for shipment after 12 May 2006 and will be shipped till 12 May 2009	May 1999
Cask validation by CNCAN	May 1999
Shipping cask and equipment arrival on site	June 1999
Contract signature	30 June 1999
Shipment documentation approval by CNCAN and ANCEX	July 1999
Physical Protection Plan approved by Romanian authorities	July 1999
➢ Spent Nuclear Fuel from 14 MW TRIGA Reactor loaded in shipping cask ➢ Shipment of Authorized Material ➢ Transfer of title at border	July 1999
Last utilization of HEU in 14 MW TRIGA Reactor	10 May 2006
Removal of all HEU from core	12 May 2006
Complete conversion of core	
All HEU available for shipment to the USA	
Preparation of a new shipment ➢ Characterization of fuel by ICN and INEEL ➢ Perform all fuel shipments	May 2009

3.2. *Institutional and political aspects*

The Romanian Regulatory Authority was settled by Law in 1966, as State Committee for Nuclear Energy. In 1990 the institution was reorganized as National Commission for the Control of Nuclear Activities (CNCAN), reporting to the Government of Romania. In 2002 the Romanian Nuclear Agency was founded, whose major responsibility is the promotion of Nuclear Energy and Nuclear Application in life and economy, reporting directly to the Government of Romania. In 2003 the National Agency for Management of Radioactive Waste (ANDRAD) was created, subordinated to the Ministry of Economy and Commerce. In 1994 the Romanian Agency for Exports and Imports Control was founded. The main responsibility of the Regulatory Authority is to reinforce nuclear safety in all areas where radioactive materials and radiation are used. This is why all such activities are subject to a recurrent process of licensing of facilities, activities and a large category of personnel. The performance of activity is subject to periodic reporting and on-site inspections by the Authority inspectors. All Romanian Authorities presented above are entitled to issue National Standards, norms and regulations for its own field of activity, in consultation and correlation with other administrative entities.

The development of nuclear energy in Romania and the application of safety regulations for all aspects of activity, starting with research until decommissioning of nuclear installations, was actively sustained by all Governments since 1990, with noticeable results such as: commissioning in 1996 and safe operation of CANDU Unit 1/700 MW in Cernavoda Power Plant; construction of Unit 2; planning of the investment for Units 3 and 4, to be commissioned in 2012; final shut down of 2MW RR in Bucharest and beginning of decommissioning; first shipment of TRIGA spent fuel to the US (1999); and full conversion of 14MW TRIGA using LEU fuel (May 2006).

3.3. *Legal issues*

As regard the shipment of the TRIGA 14MW spent nuclear fuel containing Highly Enriched Uranium enriched in the US, there are not any legal issues. The import was done on the basis of a Supply Agreement between the US Government, the Romanian Government and the IAEA in 1973. The export takes place in the framework of the Cooperation Agreement between US Government and Romanian Government on peaceful applications of nuclear energy and Contract for acceptance of FRRSNF.

3.4. *Safeguards and physical protection*

All nuclear fuel was continuously under IAEA safeguards provisions. An inventory unit of TRIGA research reactor fuel is one fuel rod recorded on the basis of shipper’s data. No mechanical or other events affected the integrity of the fuel rods and they are shipped in the same conditions of integrity.

In 1999 Romania adhered to Additional Protocol, which is in force since 2001.

The fist shipment was done with the complete observation of safeguards provisions by the reactor operator and by the Regulatory Authority, in close cooperation with IAEA Safeguards representative. The title of Authorized Material passed from the Institute to DOE representative at the border of the country.

Physical Protection of Nuclear Materials and Nuclear Facilities was provided following regulations and means available in the second half of 1999, following the provisions of IAEA, INFCIRC 225 Rev.3, contract provisions and Romanian CNCAN regulations in force at that time. The permanent plan for Physical protection of Research Reactor and nuclear fuel is one section of the operating license of the TRIGA-14MW Reactor. Additional plans for Physical protection of installations and fuel were established and approved for fuel handling and loading into the new confinement, i.e., the shipping cask. Another temporary Physical protection plan for domestic transportation was commonly established by the Ministry of Industry and Commerce, Ministry of Transportation and Ministry of Internal Affairs, then approved by Romanian Regulatory Authority – CNCAN.

The Regulatory Authority performed on-site inspections related to the preparation of shipment, fuel loading and departure from site.

3.5. Public acceptance in Romania

Nuclear energy development in Romania and related activities are accepted by the general public and by some non-governmental organizations for several reasons:

The actual share of nuclear energy produced by Cernavoda NPP Unit 1 is 10% of the total consumption in Romania and is expected to grow to 20% in 2007.

The safety of the power plant and of other applications is demonstrated and is not a reason for public concern;

The provisions of Romanian Nuclear Law calls for transparency openness and public hearing for all matters concerning nuclear activity and nuclear safety.

A continuous effort for education in high school, college and universities deployed by the Ministry of Education and other entities succeed in developing a positive attitude towards nuclear power.

We have no record of revolt or protests against nuclear sites, shipments, etc.

4. Nuclear safety, quality assurance and training for project

4.1. Nuclear safety of project

The overall Nuclear Safety of the project was analyzed and established by "Mitigation Action Plan" and by Environmental Impact Statements (EIS) documents settled by the Department of Energy, covering all aspects of Research Reactor Spent Fuel Acceptance, Transportation and Management in the US.

The peculiar aspects of nuclear safety during spent nuclear fuel handling and shipment in the country concern:

- On-site Safety of Research Reactor Facility;
- Safety of fuel handling and cask loading;
- Out-of-site safety of transportation to the border.

During on-site and domestic transportation nuclear safety was subject to Romanian Nuclear Regulations, applied by the Operating Organization, i.e. the Institute for Nuclear Research (in site) and by Licensed Transportation Organization (on road), from the site to the border.

Nuclear safety on site and during transportation concerns fuel handling, storage in transit and cask transport conditions, formalized in the supplementary analysis of reactor safety and in the operating procedures and instructions.

The role of the above analyses and procedures is to:

- minimize radiation exposure;
- prevent inadvertent criticality;
- limit uncontrolled increase of temperature;
- store and content Irradiated fuel;
- prevent mechanical or corrosive damage of nuclear spent fuel in any circumstances.

In order to minimize radiation exposure a specific Radiation Protection Plan was prepared in 1999, to cover special circumstances of fuel handling in the reactor pool, to transfer the cask out of the reactor hall, and to load the transport cask, in order to survey all exposure and possible contamination of personnel, equipment and any facility.

To prevent inadvertent criticality, provisions were established during the design phase of the facility, storage pool and storage raks The fuel handling procedures were also established to prevent criticality.

The design of the shipping cask with the purpose to facilitate fuel loading, transport, receipt handling and storage of Authorized Material was licensed by US Nuclear Regulatory Commission and accepted by Romanian Regulatory Authority in a legal process of validation satisfying all criteria specified above for fuel handling in reactor facility and during transport.

Additional analyses were performed by the Institute with the real data of the fuel batch that was subject to shipment, in order to reevaluate inventory, criticality, fuel temperature, dose at shielding and at 1 m. distance. The real data of the fuel batch to be shipped, as well as those of all the reactor fuel HEU and LEU, are available following the research reactor fuel management procedures. This means operational data and post-irradiation data concerning burnup, isotopic composition and fission product inventory, considering the history of utilization and cooling time, as well as the dimensional modification of fuel rods such as diametral growth, bending, elongation, etc.

4.2. Quality assurance

The Institute developed and maintained, initially in 1978, a QA Plan for commissioning and operation of the Research Reactor. This was a good training and, in 1984, the QA Plan was extended to other activities in the Institute, such as post-irradiation examination, treatment and conditioning of research radwaste, design and fabrication of nuclear equipment.

Presently, the Quality Management System of the Institute is accredited by Lloyd’s Register, as per ISO 9001:2000. During the spent nuclear fuel shipment, as new and unique activities for the Institute, several actions were carried out:

(a) Shipment contract analysis in order to identify:

- Contractor requirements,
- Customer obligations and responsibilities,
- Customer resources,
- The extent of application of existing regulation procedures and necessity to elaborate new procedures and to endorse Contractor procedures,
- Needs for training and retraining.

(b) Considering all provisions and requirements of the Contract, a “General Control Issue list” was established at the level of project management on behalf of the Institute. This list ensures immediate tracking of any open issue to settle solutions and timely actions communications to project team, status of issues as new; under consideration; in work; or closed.

(c) Shipment of FRRSNF within the Institute was defined as a complex process for the accomplishment of an elaborated international project. By analyzing the overall process the set of sub-processes already existing in the Institute were identified, as well as the group of persons, divisions, offices and associated procedures.

(d) Each item of the General Control Issue list was communicated to internal staff to be solved in due time.

(e) By the decision of the Institute management a project team was assigned and also a project manager to carry out the entire responsibility.

(f) Several check-lists were issued for the main sub-processes.

(g) Weekly meetings were arranged with Project management team and with representatives of subcontractors and intervening authorities to evaluate project status, identifying new issues and find solutions.

(h) Each activity was terminated with a short report supplying project status evaluation, increase of transparency, and tracking of new issues in order to prevent conflict situations.

(i) Documents management and control on the basis of existing procedures, considering present documentation and newly produced documentation in the scope of project.

4.3. Training for the project

The IAEA organized in 1997 an Interregional Training Course on Technical and Administrative Preparation Required for Shipment of Research Reactor Fuel to its Country of origin, held in Argonne, Ilinois, USA between 13 – 24 January 1997. The course was useful for the needs of participants and countries which intended to ship the fuel to the country of origin. The quality of the Training Course, the technical content and the experience of lecturers contributed to the development of knowledge related to technical, administrative, legal and institutional requirements for project accomplishment. The course provided practical skills specific to the project of fuel shipping.

Specific training of the project team was performed in the Institute, using some course documents and internal procedures. Technical procedures for access, fuel handling, heavy loads handling were reviewed as well as the technical documentation presented by subcontractors. Special training sessions took place before starting fuel characterization, before fuel loading in cask, before shipment proceeded from the Institute. The lectures were held by subcontractor specialists and by members of the project team. The project management team was instructed so as to receive information concerning the context of project development, considering organizational, political and psychological constraints that might have an impact on the project.

5. Logistics supporting infrastructure

The contract requirements did not arise difficult issues with respect to in-site logistics and infrastructure. Country routes and bridges or the access road to the Institute meet the general standards for transport of heavy cargos.

The communication systems are acceptable and the contacts between participants in the contract proved reliable.

A large effort was made by the contractor (INEEL) in terms of coordination and correlation of shipment in order to ensure a tight time-schedule with allowance of hours to fulfill various customs requirements, to comply with all applicable international and US Federal and State laws. These laws and requirements included the EIS, ROD, and Mitigation Action Plan, regulation of country where the Authorized Material was located or through which it would be transported, IAEA and IMO regulations.

The Institute was responsible for the entire technical and mechanical labor and equipment on the reactor site, necessary to assist fuel examination or cask loading operation.

The Institute made available an extra crane installed outside of the reactor building and operated by the crane owner, on the basis of a sub-contracted activity.

The Institute has its own department of transports which supplied permanent support and transport means within the country for the entire staff, both Romanian and foreign, and ensured handling means and in-site transport.

The general services for reactor operation were also available for the timely accomplishment of all tasks related to fuel loading, examination and characterization.

The on-site infrastructure was established even from the design phase of the Institute and of the reactor and it was built, licensed and has been operating continuously since the reactor commissioning.

The 14MW TRIGA Research Reactor is located in a large pool of 300 m^3 of water, as shown in Fig. 1.

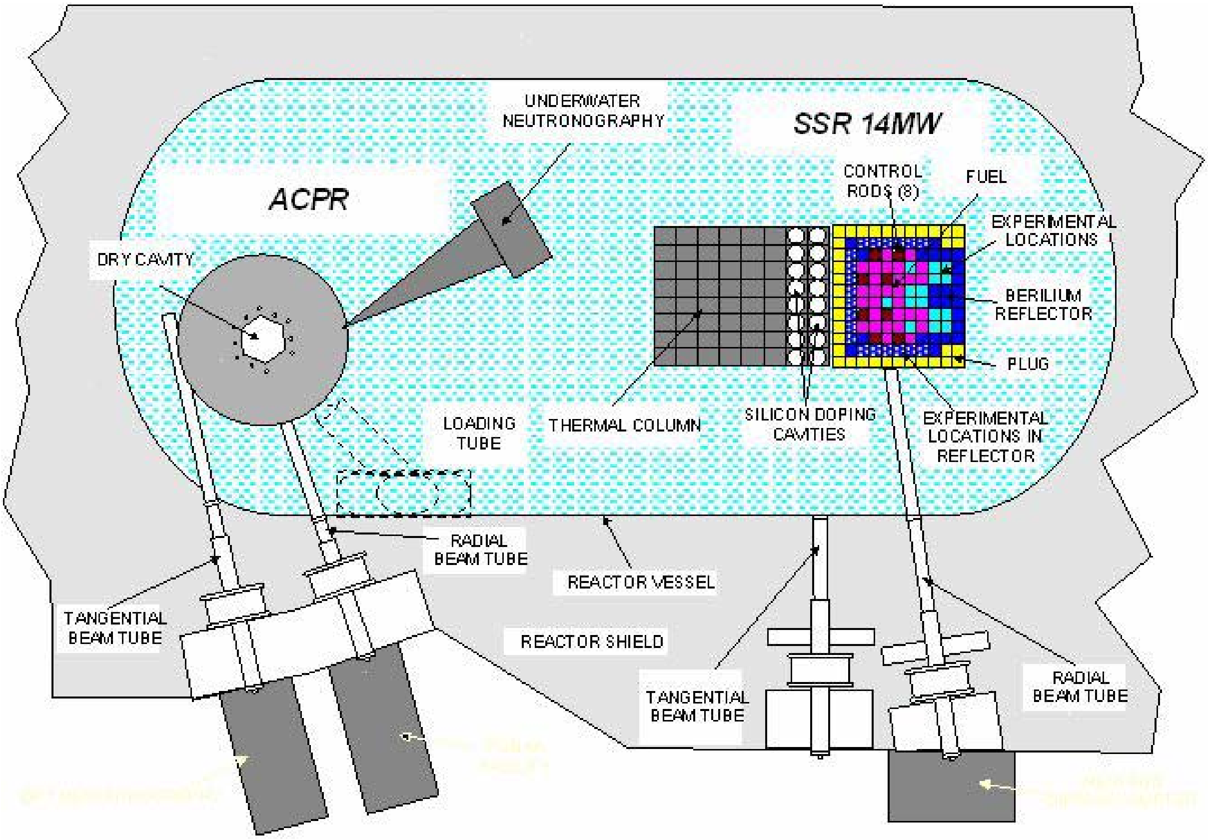

FIG. 1. The 14MW TRIGA research reactor.

The large reactor hall accommodates equipments and a bridge crane with 2 hooks of 20/5 metric tonnes. The reactor floor is designed for 5 tonnes/m^2 and allows the access of a large truck. The double access door to the reactor hall constitutes a large airlock to prevent complete opening of the hall during transfer. The area of unloading shipment equipment –2 twenty-feet containers with the cask and supporting equipment was selected outside of the reactor building, on a large reinforced concrete platform (600 sq.m) connected to 2 access ways and big enough to allow temporary installation of a 50-tonnes crane with a 40 meters arm, enough to cover the platform surface and allow unloading and loading the twenty-feet containers from and on the trailers, unloading and installation of equipment and cask, handling of the transfer cask without any difficulty.

The platform was lighted up during the night, being organized as a protected working area with permanent guard posts, limited access control, radiation and contamination monitoring, rain water collection and control, collecting points for trash and disposed materials to prevent uncontrolled spreading of contamination – if any.

In case the crane fails, one of the ways allows withdrawal of the defective one and the installation of a new one at any moment, without interference with other equipment or other heavy materials already in place.

6. 14MW TRIGA research reactor spent nuclear fuel characterization prior to shipment

6.1. Description of the first fuel batch shipped

The peculiar aspects of 14MW TRIGA fuel shipment are related to the unique design of this type of fuel. The fuel material is a high melting point alloy of uranium, zirconium and erbium. The alloy is hydrided in controlled atmosphere of hydrogen at high temperature. The precise grinded pellets are cladded in Incoloy-800 thin-wall tube, with welded plugs, 25 such fuel rods being assembled in a fuel bundle.

The design demonstrated an exceptionally long life of in-core utilization of the fuel at a mean burnup of 68%. The reactor core was heavily used, ensuring long-life stable parameters for irradiation, refueling being forecast with 3 to 5 fuel bundles per year. The fuel behavior under irradiation was within the foreseen limits of bending and elongation, without any clad failure since the commissioning in 1979. For this reason the frequent cooling water gamma spectrometry analysis shown only a low level of corrosion activated materials, but no fission product. The careful in-core fuel management, periodic interim examination of the fuel by pool-side facilities and in the hot cells of the post-irradiation laboratory ensured the assessment of operating conditions and safety margins.

The very good quality of the water in the primary cooling system is constantly maintained below 1 μS with associated parameters below designed limits. Under these circumstances there are no traces of corrosion or crud deposition on the clad surface.

The operational data of the research reactor and interim examination data for each fuel element, the correlation with burnup and history of utilization, provided a preliminary fuel characterization prior to shipment. The spent nuclear fuel was stored in a spent-fuel rack located in the pool supplied with some water, as primary cooling system.

6.2. Fuel characterization before shipment

The fuel characterization before shipment was a pre-condition of the contract, having a safety significance for the transport and the final storage at destination. The objectives of the examination performed by INEEL staff were:

- To evaluate TRIGA fuel rods conditions by visual methods to support acceptance of fuel condition to the INEEL and to decide for eventual canning requirements;

- To document visual examination information;

- To evaluate previous storage conditions on the basis of operational records for water chemistry and determine present water chemistry and microbial content.

The examination process and procedures refer to:

- All fuel rods were photographed

- All fuel rods were video-taped under water, in order to detect visible damage such as blistering, rupture, bowing out of operational limits, pin holes/pitting, cracks, mechanical damage (large dent, deep scratch, corrosion products, scale).

All damage-free fuel rods were checked with go-no-go gauge to simulate fuel transport cask insert tolerances. The accepted fuel was documented and prepared for transfer into the shipping cask by making each fuel element from 5 x 5 fuel assembly in a 4x4 basket to be inserted in to shipping cask.

The results of fuel characterization showed that all the examined fuel rods were acceptable for shipment. The fuel cladding was clean and free from corrosion products, the only images of records showing superficial scratches were due to handling of fuel rods in between grid spacers in fuel shrouds. The accepted fuel did not require canning. The entire batch was shipped without canning due to the good fuel condition. The underwater examination equipment was supplied by INEEL on the basis of temporary importation. The Institute was responsible for customs clearance and for providing documentation for non-contamination of equipment.

6.3. Fuel Preparation for Shipment

Following the fuel condition determined by fuel characterization, no special measures were needed in preparing the fuel for shipment.

The Certificate of Compliance for the shipping cask Model no. NAC-LWT, drawing of internal arrangement and basket were delivered after fuel characterization.

The Institute staff in charge with fuel loading noticed a difference between the length of the medium basket and the length of irradiated fuel pins.

To solve this issue two solutions were anticipated:

(a) redesign of the medium basket and to amend the cask license to include this design modification;

(b) cut the lower locking pin of a quarter of irradiated fuel elements to reduce the length by some 20 mm.

The root cause of this issue was found in the design process, when the length of fresh unirradiated fuel was considered following the manufacturer design. Solution (a) above seemed to be too long in time and would affect the entire shipping schedule. Solution (b) in principle reqired to bring each fuel rod in the hot cell to use hot cell equipment for cutting. This process was evaluated and found to be long, producing unexpected radioactive waste and contamination of fuel elements. An alternative cutting solution was suggested by the Institute staff, who designed, built and tested a sharing device operated under the pool water without grinding chips or hazard of contamination. Cutting was performed with minimum fuel handling, when fuel rods were ready to be inserted in the medium basket.

Further actions concerning fuel preparation for the next HEU shipment concern the modification of the medium basket in the LWT cask, to increase the amount of uranium content in the fuel element, to allow shipment of LEU fuel, to decide upon a technical solution for instrumented fuel rods, which are larger than standard fuel elements and need to be cut and canned or to be provided with a new upper plug by remote welding in hot cell to be transformed into standard fuel elements.

7. Fuel loading process and procedures

The equipment for fuel loading was supplied by INEEL and its subcontractors on the basis of temporary importation.

The NAC-LWT container was installed in vertical position outside of the reactor building, on the platform presented in Chapter 5. The transfer cask was used to load the fuel basket with designated fuel elements underwater, being handled by the reactor hall crane, to be placed on a platform mounted on a 7-tonne truck. The truck was used between the reactor hall and the platform, where the 40-tonne crane rised the transfer cask atop the shipping cask, to allow fuel loading.

Handling and operation procedures were provided by the cask owner. The operations were jointly performed by the staff of NAC and by the Institute trained personnel.

All operations took place smoothly and without incidents. At the end of the loading the cask was brought to horizontal position and prepared for shipping, using another set of procedures for tightness control, contamination control, and shock absorbers installation.

An important volume of documentation was produced for shipment and transport:

- written certification by the Institute on the amount and condition of the Authorized material;
- Shipper Training and Safety;
- Emergency Response Plan;
- Inspection Plan during transport;
- Radiation protection Plan;
- Export and Import documents, as per IAEA and USA requirements;
- Commercial Vehicle inspection plan for truck shipment;
- Results of radiation and contamination surveys prior to departure and in transfer points.

8. Description of transport

The transport from the reactor site to the border was performed by a licensed subcontractor.

The Institute provided physical protection during transportation by a police patrol squad and radiation protection support. The departure time and transport schedule were coordinated by INEEL representative in order to meet the overall plan of shipment requirements.

9. Conclusions

The first shipment of TRIGA 14MW Research Reactor Highly Enriched Uranium Spent Nuclear Fuel to the United States of America in 1999 and the Full Conversion of TRIGA-14MW Research Reactor from HEU to LEU Fuel in May 2006 are the result of the accomplishment of large internationally coordinated projects requiring preparatory activities, resources, national commitment and international cooperation.

The above achievements demonstrated the feasibility of the non-proliferation policy and responsibility of the Member States which actively sustained the projects for the reduction of HEU utilization in civilian use.

The project also offered the opportunity to apply safety standards developed by the IAEA, by USA and Romania and by other international organizations, which revealed the safety of this shipment that took place in 1999.

Having in mind this experience and lesson learned about coordination, new shipments are foreseen in the near future in order to completely abandon HEU utilization or storage in Romania.

Return of spent TRIGA fuel

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Abstract. Spent fuel from J. Stefan Institute TRIGA reactor was successfully shipped to the US in 1999. Totally 219 standard TRIGA fuel rods used in the reactor from 1966 to 1991 were shipped. Together with the experience interesting for other reactors preparing for shipment, the following aspects of the project are explained: training of all persons involved, organization (QA, responsibilities), pre-preparation of the fuel, characterization of the fuel elements (burn-up determination, inspection of physical integrity), technical preparation for the shipment, administrative preparation (environmental impact report, safety report, operating and emergency procedures, qualification of equipment, permit), loading of the shipment containers, transfer of the containers to the port, signing of the bill of lading and transfer of liability. The role of main parties involved (J. Stefan Institute, US-DOE, IAEA, NAC) is explained. According to the contract covering the first shipment, we intend to return also the remaining fuel elements after 2016.

1. Introduction

On 28 July 1999 at 5:00 a.m, totally 219 standard TRIGA fuel elements from J. Stefan TRIGA reactor were loaded on a special ship in port of Koper, together with the spent fuel shipments from Rome, Italy and from Pitesti, Romania. The shipments were coordinated to arrive at the port simultaneously as a single convoy. The one-truck convoy from Romania joined the two-truck convoy loaded with Slovenian fuel at J. Stefan reactor site, to share the police and radiological escort, and left the reactor site at approx. 2:00 a.m. The one-truck convoy from Rome joined at the Italian border near Koper. At approx 6:00 a.m. the ship left the territorial sea, concluding the responsibility of Slovenian side for the shipment. The spent nuclear fuel shipment project was effectively accomplished after 3 years of preparations.

The fuel shipment project was technically coordinated and organized by J. Stefan Institute (JSI) project team established for this purpose. In this paper, only the most important segments of the project are described in fashion of a ‘lesson learned‘ as observed from the perspective of the J. Stefan Institute project team leader. The purpose of the presentation is to help other teams responsible for future shipments in small reactors to avoid mistakes, delays, unnecessary work and to improve efficiency and safety of the shipment operation.

J. Stefan reactor is a 250kW TRIGA Mark II reactor with annular core. It is situated in Brinje near Ljubljana. It is in operation since May 1966. The irradiated fuel elements can be stored in the racks in the reactor tank or in the spent nuclear fuel pool in the basement of the reactor building, accessed through a opening in the reactor hall floor (capacity: 1 600). The reactor is equipped with crane and two transport containers (1.5t and 3.5t) for fuel transportation within the reactor building.

More information about the reactor can be found at http://www.rcp.ijs.si/ric/index-a.htm

The reactor uses standard rod-type fuel elements. Several fuel element types have been used:

standard LEU, 20% enriched, 8.5w% U, Aluminum cladding.
standard LEU, 20% enriched, 8.5w% U, Stainless-steel cladding
standard LEU, 20% enriched, 12w% U, Stainless-steel cladding
standard LEU, 20% enriched, 20% U, Stainless-steel cladding
FLIP, 70% enriched, 8.5% U, Stainless-steel cladding.

The following spent nuclear fuel elementes were returned to the United States of America:

standard LEU, 20% enriched, 8.5w% U, Aluminum cladding, totally 67
(64 Type 102, one 3/4, one 1/2, one 1/4)

standard LEU, 20% enriched, 8.5w% U, Stainless-steel cladding, totally 125
(119 Type 104, 3 Type 204-instrumented, 3 fueled follower CR)

FLIP, 70% enriched, 8.5% U, Stainless-steel cladding, totally 26
(24 Type 110, 2 Type 210-instrumented)

standard LEU, 20% enriched, 12w% U, Stainless-steel cladding, totally 1
(ACPR Type)

2. Pre-preparatory phase

Already in 1996, soon after the US originating research reactor fuel 'repatriation' act was passed by the government of the United States, Slovenian government decided to accept the initiative and assigned J. Stefan Institute to organize and carry out the shipment of spent TRIGA fuel. Ample funds (approx. US $400 000) were allocated for the project, mainly to cover the costs of (preparatory) work on Slovenian side. The main costs, i.e. the fuel inspection, loading operation, hiring of the shipment containers, transportation and permanent disposal, were covered by the US government.

The following activities immediately started at J. Stefan Institute:

- appointment of the project leader and the project team
- organization of the project
- training of the project team and gathering of information
- contacts and exchange of information with Idaho National Labs (INL) and with the shipment companies (in particular NAC International)
- establishiment of the list of regulatory requirements
- preparation of the contract with DOE
- technical preparation of fuel.

Project organization and project team

The project team was established from the reactor operation staff (4 operators, head of division) and from reactor physics division (2 researchers). A researcher from reactor physics division experienced in leading the fuel management projects in nuclear power plant Krško was appointed as project leader. He was reporting directly to the director of the Institute. He was also allowed (and responsible) to engage (and commercially hire) other experts from the Institute or from other companies, in particular for preparing the licensing documentation. All workers were engaged only part time on the project as their main duties (reactor operation, research work) remained practically unchanged. Total extent of work on the project was estimated to be approximately 5 man-years, spread over 3 years. Establishing the project team and additional funds allocated for the project proved to be essential for the project as the amount and nature of work was such that could not be realized within normal reactor operation activities.

Quality assurance

The project team started to prepare the program of the project in compliance with the Institute's QA program. Main elements were: general organizational scheme, personal responsibilities and time schedule. Strict implementation of QA principles and personal responsibilities proved to be another important element for the successful realization of the project, in particular as the workers in a research institute are inclined to the 'scientific' approach of the work, often bordering on improvisation and negligence.

Information and training

Gathering of information and training was surprisingly difficult at the initial stage of the project. Practical experience from similar reactors was not available. There was only one IAEA training course in that period, however, the candidate from the JSI project team was not admitted to participate. According to IAEA selection rules, the participation was awarded to another person from Slovenia who was not involved in the shipment project at all (also in later stages of the project we did not receive any training or assistance from IAEA). The only source of useful information at the initial stage were direct personal contacts with organizations involved in the shipment program (EDLOW Co., INL, DOE, NAC). However, all institutions and individuals involved were extremely cooperative to provide information and the problem soon became irrelevant.

Regulatory requirements

Main regulatory requirements for transportation permit were the following:

- safety analysis and licensing of the shipment container
- environmental impact report
- written operating procedures
- emergency procedures.

The requirements were defined clearly by the Slovenian Nuclear Safety Administration (SNSA) from the very beginning of the project. SNSA also acted as the coordinator between JSI and the governmental bodies, in particular with the police. SNSA also solved all formalities with respect to the international safeguard requirements. Their role in the shipment project was for this reason very important. Their professional and efficient work essentially contributed to the success of the project.

Contract DOE-JSI

The draft contract between US DOE, Idaho Operations Office and JSI, on terms and conditions for the acceptance of the fuel at the Idaho National Engineering and Environmental Laboratory, was being prepared. However, it was signed only one month before the date of the shipment, when it became certain that both parties would be able to keep the terms. The element of uncertainty due to the unsigned contract was present through the entire project, but did not affect the process on the Slovenian side as we were covered by the decree of the government and the funds were provided from the very beginning.

Technical preparations

Technical preparation of the fuel consisted mainly of gathering the fuel elements in the spent fuel pool. All elements prepared for the shipment were taken from the reactor core well before the required 3 years' period from shipment. Some of them were stored in the racks in the reactor tank. One was stored in the gamma-scanning device in the reactor hall. All they were transported to the spent fuel pool in a usual way, using the transport container and the crane that are part of the reactor equipment.

3. Preparatory phase

The preparatory phase started approximately one year before the shipment. Entire project team and outside experts were involved and the work became intensive. We worked on the following three main tasks:

- characterization of fuel elements for shipment
- technical preparation of the transport routes
- preparation of documents for transportation permit
- public acceptance.

Characterization of the fuel elements for shipment consisted of two parts: preparation of a report on fuel element basic data for shipment and the fuel inspection.

Characterization: Burn-up, activity and dose rate calculations

According to the contract, the report containing data relevant to fuel transfer was prepared by JSI. Together with general fuel design information (drawings, descriptions), the following data were prepared for each and all fuel elements:

> type (FLIP, standard Al clad, standard SS Clad),
> ID number, weight (g) of main isotopes (U-235, U-238, U-236, Pu-239, Pu-240, Pu-241),
> burn-up (in MWh and in percents),
> cooling time (months),
> activity (Ci) and
> decay heat (W).

For isotopic composition and burn-up calculations we used TRIGLAV code. For activity calculations we used ORIGEN.

Prorated to June 1999 the shipment contained 39 382 g of U isotopes and 132 g of Pu isotopes. Total activity was 19 500Ci, maximum activity of a single fuel element was 196 Ci. Maximum burn-up of a single fuel element was 28.5%. Decay heat of individual fuel elements was much below 1W. We also estimated the dose-rates for planning the fuel element manipulations and to be sure that the dose outside of the transport container would be within acceptable limits. Measurements taken later during the transport container loading confirmed our predictions.

We found preparing the report on fuel element basic data very important for planning and safety of the shipment operation. Experience from other reactors shows that erroneous estimates of fuel burn-up and activity may lead to underestimated dose-rates that may jeopardize the transportation of the spent nuclear fuel. It is our experience, that we could not prepare accurate fuel data if we had not used appropriate fuel management codes (TRIGLAV, ORIGEN, WIMS) which normally is not the case in small research reactor centers.

Characterization: Visual inspection

The visual inspection of the fuel elements was performed with an underwater camera by an expert from INL. We were extremely surprised that 57 fuel elements (out of 219) were found to have (potentially) damaged cladding (small cracks and possibly perforations due to corrosion), as we identified only two leaking elements during reactor operation in 35 years (both were among the fuel elements for shipment) and as the reactor and spent nuclear fuel pool water activity never indicated the leaks, except for these two events. It is our explanation that the leaks appeared during the long term storage in the spent nuclear fuel pool and were not detected because the leaking was too small and consisted only of gaseous fission products (that mainly decayed due to long cooling time). The cladding damages were found in all three fuel types regardless of cladding type. They were also not correlated to fuel burn-up. A large number of damages (several hundreds) were found on the fuel element that was dry stored several years inside of the gamma-scanning device.

On the basis of the spent nuclear fuel pool water samples it was decided that the (potentially) damaged fuel elements did not require over-cladding.

Transportation route preparation

Main technical preparation of the transport routes consisted of the following: construction of a new platform for transport container loading near the reactor building, construction of a new short-cut road from the platform to the reactor building and reinforcing the bridge leading into the reactor hall (to sustain 25 tons of load). The 5t bridge type crane in the reactor hall was inspected as well.

Documentation

Preparation of documents for transportation permit was the most work-intensive task of all preparatory activities, in particular preparation of environmental impact report. Its contents were as follows:

Introduction
Description of the reactor and of the fuel
Benefits of spent fuel shipment
Description of the shipment project
Radiological and safety analysis
Environmental impact assessment
Operating and emergency procedures
Physical protection
Organization and quality assurance
References

The report was prepared by 10 authors, it had 60 pages. It was a public document. Like SAR in case of reactors, it was prepared as a licensing document and the document for public information.

Public relations

All details about the shipment (except the date of transportation) were made available to general public. Several meetings with the local community and with the media were prepared. After these meetings, opposing to the shipment turned into full support by the local community and into a kind of euphoria by the media.

4. Fuel loading and transportation

This phase of the project involved the co-operation of several organizations:

JSI project team: fuel handling inside of the reactor hall, crane manipulation
NAC team: transport from the reactor hall to the transport container and its loading
local truck company: manipulation and loading of trucks (80t truck-crane)
transportation company (2 trucks)
radiological escort – JSI radiological emergency unit
police escort
permanent surveillance by nuclear and radiological safety inspectors.

The fuel loading and transportation phase of the project is described in the chronological order.

Arrival of equipment

The 20ft transportation ISO containers (2) and the tools arrived approx. 2 weeks before the transportation date. Once the containers were on site, additional radiological monitoring and physical protection measures were implemented (additional guards). According to our procedure, the containers and the equipment were radiologically inspected at arrival. The containers were found to be contaminated on the outer surface (Cs). The radiological report was prepared. Later this turned out to be important, as at the entry in US this contamination was again detected. On the basis of the report we were able to prove that the contamination did not happen at our site.

Loading of fuel for transportation

The loading of the containers was performed jointly by JSI reactor operators and NAC team. Our operators were loading the fuel elements from the spent nuclear fuel pool to the transfer basket and carried it by the crane to the transport trolley waiting at the reactor hall entrance. The trolley was towed by a forklift to the platform with transport containers. However, after a few tows it was decided that it was easier to carry the entire trolley by the fork-lift than to tow it (the wheels were too small and the trolley was difficult to steer). The fuel basket was lifted together with its shielding from the trolley by a truck-crane and inserted into the transport container which was in vertical position. The loading of the fuel was completed in 3 days. The transport containers were closed and stored in horizontal position into the ISO shipping containers. ISO containers were loaded on the trucks by the 80t truck crane. All operations were performed without complications. JSI and NAC teams cooperated in excellent way.

The photographs of the loading of the fuel are presented at: http://www.rcp.ijs.si/ric/index-a.htm

Transportation

The transport from the reactor centre to the port was organized by the police. The convoy consisted of 2 police cars, two policemen on bikes, three trucks (2 from JSI and 1 from Rumania) and one radiological emergency unit car. The policemen on bikes blocked the crossings before the convoy so that the convoy did not have to stop until the final destination. The date and hour of the transport was made known to the JSI team 12 hours in advance.

5. Conclusions

The fuel shipment in 1999 was successfully accomplished. 219 TRIGA fuel elements were returned to the US. 94 fuel elements are still at the location. The contract between DOE and J. Stefan Institute leaves open the possibility that also the remaining fuel at our reactor could be returned in the future. It is our intention to use this possibility after 2016, when the reactor is expected to be closed.

Thai experience on return of research reactor spent fuels to the United States of America

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Abstract. The first Thai Research Reactor (TRR-1) was established in 1961 at the Office of Atoms for Peace (OAP) Bangkok, Thailand. It was supplied by Curtiss-Wright corporation from Quehann Pennsylvania, used HEU MTR type fuel, went critical in 1962 and eventually shut down in 1975 depending on the HEU fuel sypply problem and to support the NPT. Since then the spent nuclear fuel has been stored in wet storage at OAP for 24 years prior the shippment to the US in 1999. During the storing period, there was no signs of storage problems neither any serious condition of the spent fuel elements was noticed.

According to the USDOE acceptance programme of research reactor spent nuclear fuel, and the tangible benefit gained from the conversion of TRR-1 reactor core from HEU to LEU in 1977 (RERTR Program). The thirty one (31) of the Thai Research Reactor – 1 (TRR-1) spent nuclear fuels elementes were accepted by the USDOE in accordance with the Record of Decision on 13 May 1996 and transported to Savannah River in South Carolina on 1 March 1999.

1. Introduction

On 20 February 1996, the Secretary-General of OAP received a letter from the Embassy of the United States in Bangkok informing about the decision of the U.S. Department of Energy (USDOE) to accept and manage FRRSNF in the United States, including all the spent nuclear fuel listed in the Environmental Impact Statement (EIS). As stated in the reference document, the fuel, all of which, contains uranium enriched in the United States, would be accepted over a thirteen (13) year period.

Countries listed in the EIS as "developing countries" would have the cost of fuel shipment and storage paid by the United States, and arrangements would be made through the Department of Energy Office of Spent Fuel Management.

On 29 August 1996, the OAP's Secretary-General received a second letter from the Embassy of the United States in Bangkok informing of the Record of Decision (ROD) for an Environmental Impact Statement of a policy to manage foreign research reactor spent nuclear fuel. Additionally, the letter informed the lawsuit situation in South Carolina and also informed the spent nuclear fuel shipments from Chile, Colombia, Germany, Switzerland, and Sweden that arrived in South Carolina in September 1996.

On 13 September 1996, OAP received a fax proposing a U.S. delegation to visit the TRR-1 facility for preliminary site assessment. After the acceptance of OAP, the visiting took place on 14-15 October 1996 at the Office of Atoms for Peace, where the delegation meet with OAP's personnel.

In January 29, 1997 OAP decided to participate in the shipment of spent nuclear fuel from Thailand to the United States. From this date, two main types of activities were implemented (1) Negotiation and Management and (2) Technical Activities.

2. Negotiation and management

In order for OAP to carry out the several and most diverse activities related to negotiations and management of this operations, it was decided to set up a working group comprised of personnel from several areas, including the Regulatory body, and staff with knowledge of Health Physics, Waste Management and the Reactor operator, who was directly responsible for all the spent nuclear fuel while stored in the facility.

The main activities related to this topic were the negotiation of contractual matters, safeguard issues and managerial activities related to transportation, security, budget, fuel technical data and cost scheduling.

Above all these activities were the coordination and collaborative efforts performed between OAP and DOE including Washington DC and Savannah River Site and also Westinghouse Savannah River Company (WSRC). For OAP side, negotiation issues were done and communicated with DOE personnel by Secretary-General and Deputy Secretary-General. At the same time, all technical issues were done in parallel by the reactor manager and by the reactor operator, who constantly reported to the Secretary-General for decision making issues.

2.1. Contractual matters

With respect to the legal framework related to this operation, there were a number of very subtle considerations which needed a quick acquisition and response time. The way this was resolved was through an intense and strong communication link between the Secretary-General, the Deputy Secretary-General at OAP and the USDOE. This was also facilitated and consulted by the Office of Attorney General for OAP side. The issues related to Nuclear Liability and Title Transfer Location were particularly complex and required significant effort from all parties involved.

On 19 February 1999, the U.S. Department of Energy and OAP agreed to sign a contract to transport thirty-one (31) MTR type fuel elements from TRR-1 facility to the Savannah River Site in South Carolina.

2.2. Safeguards

Based on the safeguards agreement between Thailand and the International Atomic Energy Agency (Article 92, INFCIRC/241) it was necessary to carry out basic actions in accordance with the IAEA safeguards regulations. By means, Advance Notification of intended tranfer of nuclear material out of Thailand were made to the Agency within two weeks before the nuclear material was to be prepared for shipment. This nuclear material was under IAEA's control during its entire utilization and decay period at TRR-1 facility. All of HEU fuels, especially three (3) fresh fuels were verified by the IAEA safeguards inspector on 25 February 1999, prior to their load in the shipping cask.

2.3. Managerial matters

Several activities were accomplished to insure the physical protection of the spent nuclear fuel. Arrangements with Custom both, at the Lam Chabang deep seaport and Bangkok International Airport, were made.

Among the physical protection tasks the very first one was to coordinate with the authorities of the Airport, deep seaport, Crime Suppression Police, Highway and Provincial police. For this tasks OAP followed the regulation established in the "Enhancement and Conservation of National Environmental Quality Act" as well as those defined in the International standard Guideline of Nuclear Material Transportation.

The communication at the high level of the Office of Atoms for peace and the authorities mentioned above were facilitated due to the position of OAP within the government structure. As a result of this, all the planning and coordination of the various level was straight forward and expeditions. For this operation, a high degree of coordination and accurate information was needed to provide to the Authorities in charged, and in this way to guarantee a quick response and to maintain the complete operation under a rigorous centralized control.

For Customs we proceeded in the same way; that is, a formal letter from our Secretary-General to the Director General of Customs Department to initiate a process and set up the framework of the custom procedure. All the paperwork and authorization to do the "re-exportation" of the spent nuclear fuel elements, were done efficiently and in prompt manner, in particular, the operations at the Port in all aspects were really impeccable.

3. Technical activities

Together with the negotiation and management activities it was necessary to collect the technical information of the spent nuclear fuel elements and to initiate several technical activities at the TRR-1 research reactor. Among the most important activities were the gathering of the technical information to fill out the Appendix A (Agreement of Spent Nuclear Fuel Acceptance Criteria) of the contract between the United States Department of Energy and OAP. Other activities included the preparation of the TRR-1 SNF storage building, to accept the necessary equipment to transfer the spent nuclear fuel element from the storage pool to the shipping cask situated outside of the storage building. Also, it was necessary to write the physical protection plan for transport the spent nuclear fuel elements to the deep sea port and the operation procedures for removing/loading the HEU SNF into the basket of the shipping cask. Procedures had also to be written for the control element cropping operation.

During the meeting held with the United States Delegation at the OAP on 4 December 1998, it was agreed that 31 spent fuel elements from TRR-1 facility could be sent in the first shipment to Savannah River Site. Another important conclusion was the possibility to remove (or chop off) the end-boxes and/or end fitting from the control elements.

NAC International Inc. provided the necessary equipment for the shipment of the spent nuclear fuel to the United States. These were a Dry Transfer System and associated equipment, NAC LWT shipping cask and cropping machine. To move the heavy materials and facilitate the operations involved in the loading and transport of the spent nuclear fuel elements, a mobile crane of 80 tonnes of capacity, air compressor, helium gas, forklift and scaffolding were provided locally.

3.1. Technical information of the spent nuclear fuel

The physical and chemical characteristics, approximate isotopic composition, dimension and weight of the spent nuclear fuel was given in appendix A of the contract, as required by Savannah River Site for the acceptance of the spent MTR fuel. At TRR-1 reactor the complete history and technical information per each fuel element is permanently tracked and maintained. This includes manufacturing specification, drawings and the fuel irradiation history of the fuel element which made a straight forward task to fill out the information in appendix A.

The HEU MTR type elements used in the TRR-1 research reactor were supplied by Curtiss-Wright from Quehann Pensylvania, USA in 1962. The standard fuel element, half element and control element contained curved 10, 5 and 6 fuel plates designed as illustrated in Fig. 1, respectively. Each fuel plates contains a central core of aluminum-uranium oxide with a pure aluminum cladding. The uranium is enriched to approximately 89.93% U-235 with each element having a U-235 loading as tabulated below. The fuel plates are mechanically bonded to the side plates. A summary description of the fuel elements is shown in Tables 1 and 2.

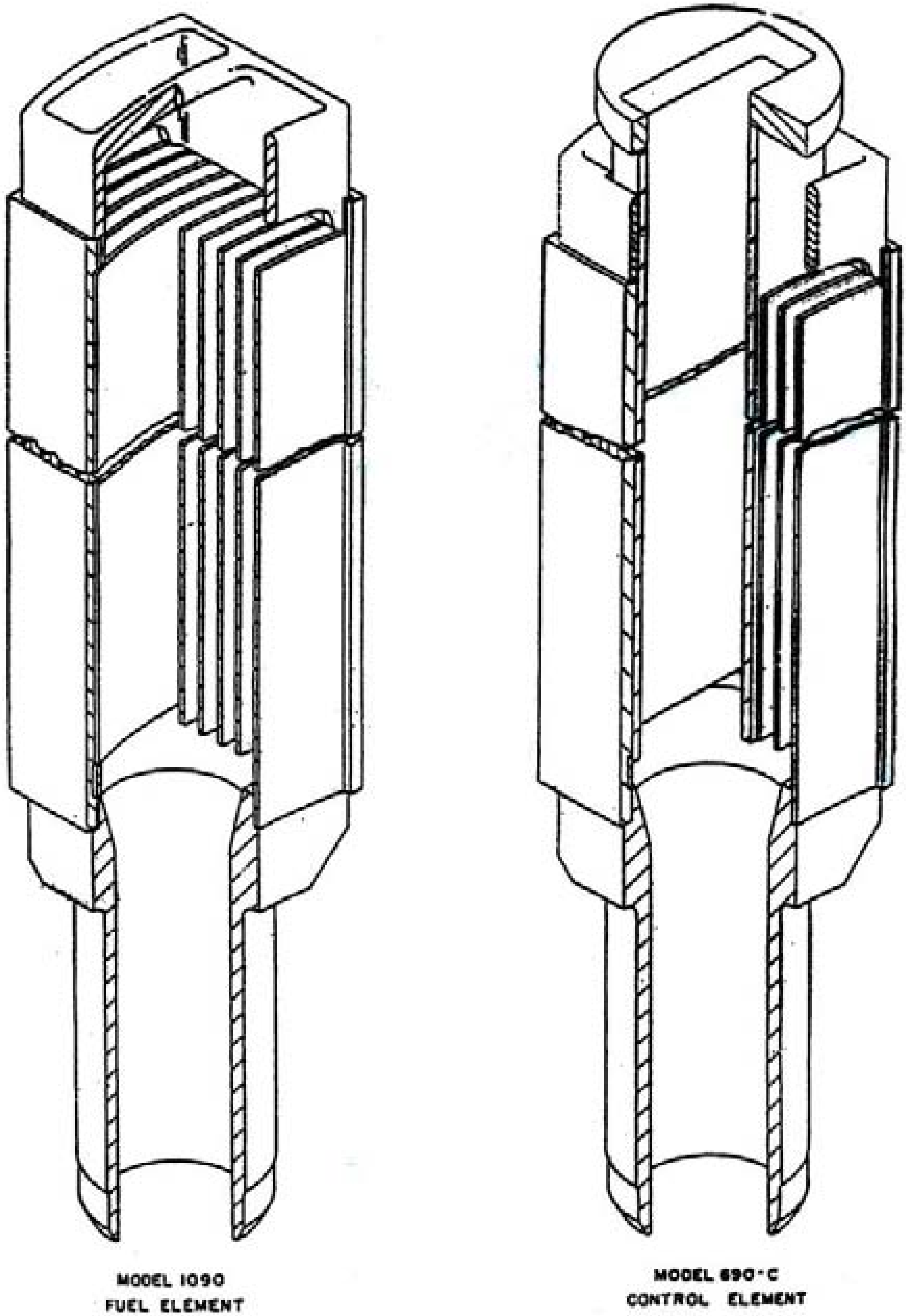

FIG. 1. TRR-1 fuel element.

TABLE 1. MTR ALUMINIUM-BASED SPENT NUCLEAR FUEL ELEMENT

Serial No.	No. of Element	Original Element Weight (gm)	Original Fissile Weight (gm)
HR-1	1	94.59	85.04
F-1	1	188.81	169.70
F-2	1	189.12	170.06
F-3	1	189.04	169.98
F-4	1	189.09	170.00
F-5	1	188.97	169.91
F-6	1	188.89	169.85
F-7	1	188.99	169.86
F-8	1	189.26	170.03
F-9	1	189.45	170.26
F-10	1	189.04	169.98
F-11	1	188.86	169.83
F-12	1	188.97	169.93
F-13	1	189.00	169.95
F-14	1	189.05	169.99
F-15	1	189.08	170.02
F-16	1	189.09	170.03
F-17	1	189.03	169.98
F-18	1	189.07	170.01
F-19	1	188.98	169.93
F-20	1	189.14	170.07
F-21	1	189.00	170.05
F-22	1	189.15	170.08
C-1	1	113.41	101.98
C-2	1	113.38	101.95
C-3	1	113.39	101.96
C-4	1	113.43	102.00
C-5	1	113.41	101.98
Total	28	4820.80	4334.41

Note

1. All spent MTR HEU plates type were kept in spent fuel storage pool
2. Integrated power 482.46 MWD
3. Average enrichment of MTR Spent fuel Approx. 86.34%
4. Total remaining Uranium = 4 298 gm
   U-235 = 3 711 gm
5. HR-1 = Half – 10 plate Rear Fueled
6. F-1 to F-24 = Standard – 10 plate (Full)
7. C-1 to C-5 = 10 – plate Rod element

TABLE 2. INVENTORY OF FUEL ELEMENTS FOR TRR-1*

Type	CW Part No.	Number of Fuel Bearing Plates Per Element	Grams U-235 Per Element (Average)	Number of Elements Available	Total Grams U-235**	Serial No.
Standard-10 Plate (Full)	1090F	10	169.98	24	4 079.48	F-1 to F-24
Half-10 Plate Front Fueled	590F	5	85.03	1	85.03	HF-1
Half-10 Plate Rear Fueled	590R	5	85.04	1	85.04	HR-1
10-Plate Rod Element	A690C	6	101.97	5	509.87	C-1 to C-5

Total Number of Fuel Elements Available…………………………………. .31
Grand Total Grams U-235 Available,……………………………………..4 759.42**

* Initial inventory before the burn-up.
** Taken from CW Report submitted to USAEC as accepted by the THAI AEC (License No. SNM 156).

Base on the irradiation history of each spent nuclear fuel element, the additional information required in Appendix A was determined. Particularly, the following parameters were evaluated: burn up; the content of Special Nuclear Material (SNM) after irradiation; period of time that the fuel element staied in core; irradiation time: cooling time: energy obtained per fuel element: dose rate at 1 meter in air and decay heat. All of this information were calculated using ORIGEN-JR and QAD-JR computer codes and shown in Tables 3 and 4.

TABLE 3. FUEL IRRADIATION HISTORY – GENERAL SUMMARY

Unique ID No.	Total Weight Fuel asse. (g)	Fuel asse. Loaded on dd-mm-yy	Fuel asse. Discharged on dd-mm-yy	Time in Reactor Core (days)	Power Level (Mwd/Asse.)	Cooling Time Jan 1 1997 (days)	Dose Rate At 1(m) in Air (Sv/h)	Decay Heat (W)
F1	4810.8	27-Oct-62	25-Sep-75	4745	32.88	32.88	3.029	1.2239
F2	4810.8	27-Oct-62	25-Sep-75	4745	31.00	31.00	2.856	1.1540
F3	4810.8	27-Oct-62	25-Sep-75	4745	28.28	28.28	2.605	1.0527
F4	4810.8	27-Oct-62	25-Sep-75	3650	30.59	30.59	2.818	1.1386
F5	4810.8	27-Oct-62	25-Sep-75	4745	28.25	28.25	2.602	1.0515
F6	4810.8	27-Oct-62	25-Sep-75	4745	30.13	30.13	2.775	1.1216
F7	4810.8	27-Oct-62	25-Sep-75	3650	29.55	29.55	2.722	1.0998
F8	4810.8	27-Oct-62	25-Sep-75	3650	29.55	29.55	2.722	1.0998
F9	4810.8	27-Oct-62	25-Sep-75	3650	29.00	29.00	2.671	1.0795
F10	4810.8	27-Oct-62	25-Sep-75	4745	29.58	29.58	2.725	1.1010
F11	4810.8	27-Oct-62	25-Sep-75	4745	29.30	29.30	2.699	1.0907
F12	4810.8	27-Oct-62	25-Sep-75	4745	29.73	29.73	2.739	1.0932
F13	4810.8	27-Oct-62	25-Sep-75	3650	26.80	26.80	2.469	0.9975
F14	4810.8	27-Oct-62	25-Sep-75	4745	29.80	26.80	2.469	0.9975
F15	4810.8	27-Oct-62	25-Sep-75	2555	17.75	17.75	1.635	0.6606
F16	4810.8	27-Oct-62	25-Sep-75	2555	17.75	17.75	1.635	0.6606
F17	4810.8	27-Oct-62	25-Sep-75	1095	0.923	09.23	0.850	0.3435
F18	4810.8	27-Oct-62	25-Sep-75	1095	06.06	06.06	0.558	0.2255
F19	4810.8	27-Oct-62	25-Sep-75	1095	04.76	04.76	0.438	0.1771
F20	4810.8	27-Oct-62	25-Sep-75	1095	05.05	05.05	0.465	0.1879
F21	4810.8	27-Oct-62	25-Sep-75	1095	05.34	05.34	0.491	0.1987
F22	4810.8	27-Oct-62	25-Sep-75	1095	05.05	05.05	0.465	0.1879
F23	4810.8	-	-	-	-	-	-	-
F24	4810.8	-	-	-	-	-	-	-

Note: The methodology for the calculation of the Post-irradiation and decay heat by used ORIGEN-JR and QAD-JR Code

TABLE 4. FUEL IRRADIATION HISTORY – GENERAL SUMMARY

Unique ID No.	Total Weight Fuel asse.	Fuel asse. Loaded on	Fuel asse. Discharged on	Time in Reactor Core	Power Level	Cooling Time Jan 1 1997	Dose Rate At 1(m) in Air	Decay Heat
	(g)	dd-mm-yy	dd-mm-yy	(days)	(Mwd/Asse.)	(days)	(Sv/h)	(W)
HR1	4748.15	27-Oct-62	25-Sep-75	1825	11.17	8030	1.029	0.4157
HF1	4748.15	-	-	-	-	-	-	-
	(g)	dd-mm-yy	dd-mm-yy	(days)	(Mwd/Asse.)	(days)	(Sv/h)	(W)
C1	3890	20-Oct-62	26-Sep-75	4745	14.78	8030	1.361	0.5500
C2	3890	20-Oct-62	13-Sep-75	4745	14.78	8030	1.361	0.5500
C3	3890	24-Oct-62	13-Sep-75	3650	9.00	8030	0.829	0.3348
C4	3890	20-Oct-62	13-Sep-75	4745	14.78	8030	1.361	0.5500
C5	3890	20-Oct-62	13-Sep-75	1095	10.4	11315	0.958	0.3871

Note: The methodology for the calculation of the Post-irradiation and decay heat used ORIGEN-JR and QAD-JR Codes

3.2. Technical preparation

Another important technical activity was to prepare the spent nuclear fuel storage building and the surrounding areas to load the spent nuclear fuel elements into the shipping cask by using the NAC's Dry Transfer System (DTS). The Dry Transfer System consists of a transfer cask with MTR fuel basket grapple, a transfer cask carriage, and a cask adapter. Due to physical restrictions and the limited area in the spent fuel pool, the transfer cask was used to transfer spent nuclear fuel from the spent fuel pool to the shipping cask located outside of the spent fuel building.

A complete technical information of the NAC's equipment including SAR of the shipping cask was received at TRR-1 in advance [1]. By using this information, the reactor personnel, together with NAC personnel, prepared the scaffoldings and selected proper location to install the equipment e.g. cropping machine and NAC-LWT shipping cask. All working areas were cleared to prevent accidents and to provide enough space for workers.

To prevent any difficulty during the loading operation, a complete inspection of the 80 metric tonne mobile crane and the air compressor system was done. The latter one was needed for the pneumatic operations and tools.

Demineralized water was also prepared for decontamination purposes, to fill out the cask prior to shipment and with the purpose to take water samples to verify if the Cs-137 concentration was acceptable for shipping. Helium was required for leak testing of the shipping cask closure lid. For this reason, two bottle of Helium were provided at the operation site. Helium was also used to fill the cask cavity and maintain the fuel elements under an inert atmosphere during the transportation.

The Article V.D. and Appendix B (Agreement Transport Package (Cask) Acceptance Criteria) of the contract required 2 x 250 ml of water sample from the storage pool to be shipped to the Savannah River Site in accordance with the instructions in the Appendix B [2], which was done accordingly.

Figure 2 shows the MTR-35 LWT basket loading diagram with a sketch of the fuel element positions, identification numbers and decay heat for the authorized fuel element.

Figure 3 shows details of the 7 element basket.

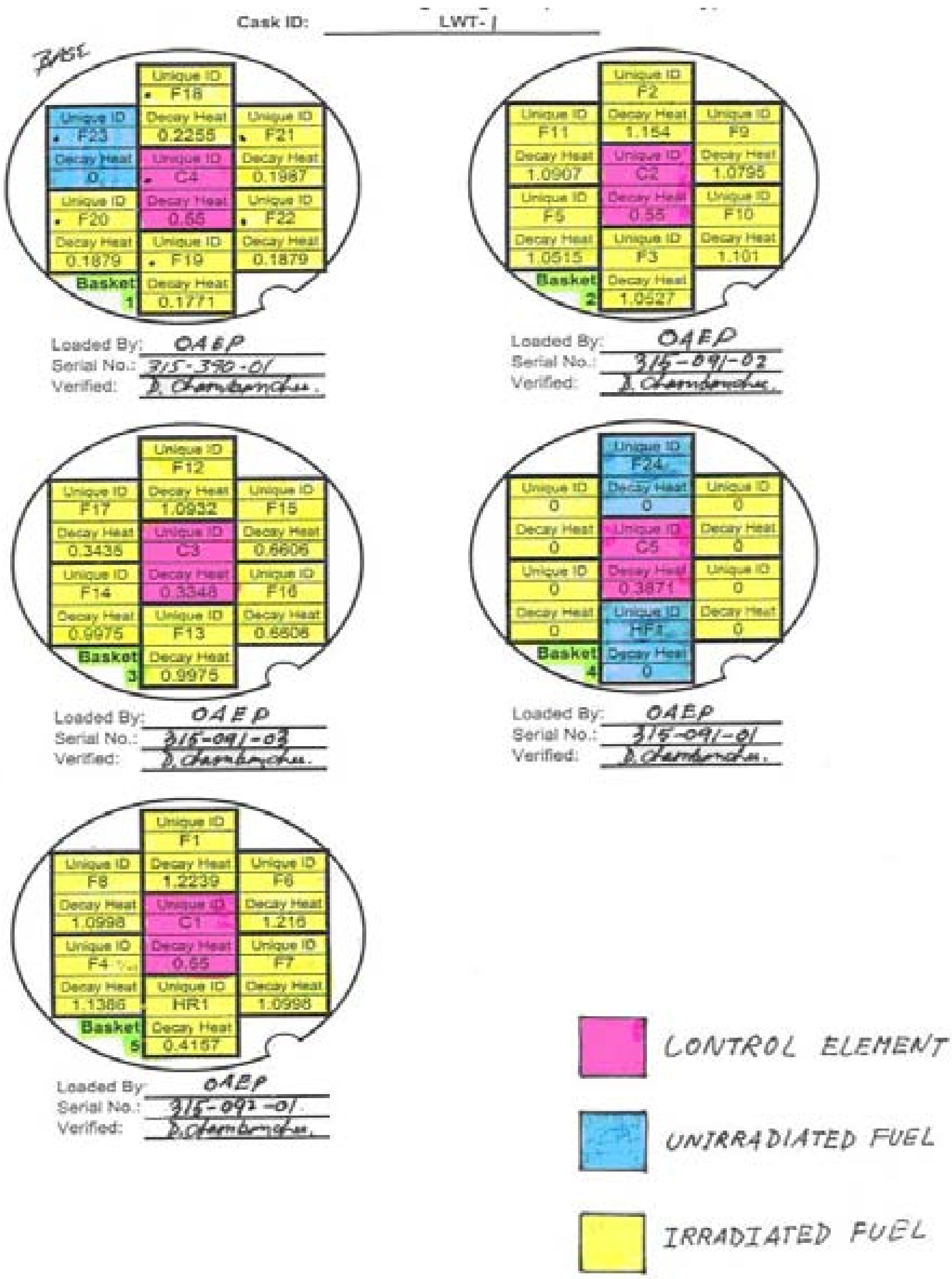

FIG. 2. MTR-35 basket loading diagram (TRR-1 Facility).

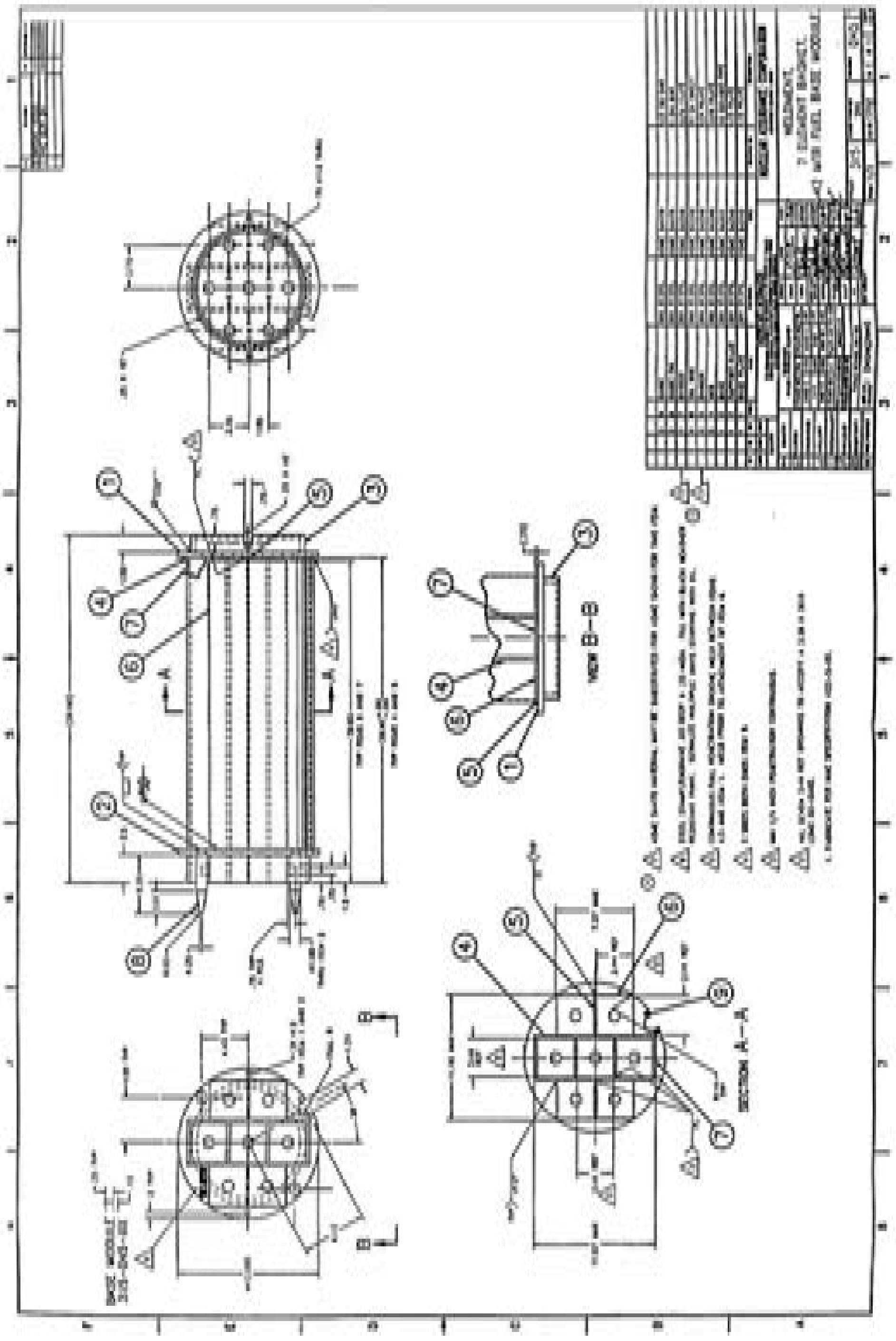

FIG. 3. . LWT cask basket.

3.3. Loading operation

On 13 February 1999, the MV Mahimahi V.029 ship arrive at Lam chabang deep seaport carrying 1 (one) ISO container housing with the empty shipping cask and 2 (two) ISO containers with tools and the necessary additional equipment from the US. OAP's personnal also received the empty basket from the Airport Autority for fuel loading operation.

The loading operation was based on the NAC's procedure for the utilization of the MTR Fuel Dry Transfer System used in conjunction with a NAC LWT shipping cask (Figs 4 and 5). This procedure provided the necessary steps to operate the system, assisting the user to prepare the specifici procedure for the operation, and to inform the user about the operation features of the system. In spite of the details contained in this operating procedure, the presence of qualified personnel of NAC International Inc. was essential to expedite the preparation before and during the loading. In this sense, the pocedure was meant to be utilized as a guide by experienced personnel.

After the inspection for damage, the equipment was removed from the container on 20 February 1999 and set up at the designated location. When the lid from the container was removed, a Health Physicist made a radiation survey of the shipping cask, making an official record of the result.

Once the top and bottom impact limiters from the shipping cask were removed, the shipping cask was carefully raised to upright position on the rear cask support and placed onto the base plate. The pressure in the cask cavity was equalized using a vent value and then the closure lid was removed.

On 22 February 1999, OAP received the written authorization from DOE to load and ship thirty-one (31) SNF assemblies to SRS. Thus, on 23 February 1999 the loading operation of the 31 spent nuclear fuel elements was fully carried out. During the loading process of each spent fuel element, in accordance to the Appendix B of the contract, a description of the observable physical condition was recorded. The results showed no visual evidence of corrosion, pitting cuts or any other physical indication of damage of the authorized fuel elements.

The cask cavity was flooded with demineralized water to allow radiological contamination surveys in accordance with the specifications giving in Appendix B. A water sample was taken on the next day, after 16 hours. The average Cs-137 concentration of the water sample was about 18.5 dpm/ml, an activity far below the maximum required value specified in Table 1 of the Appendix B, which , in the case of NAC LWT shipping cask, the Cs-137 is 278 dpm/ml (4 630 Bq/l)

To remove the demineralized water from the shipping cask, pressurized air was blown into the cavity followed by a vacuum dried process according to the operating procedure. Then the cask cavity was filled with helium and the closure lid was leak tested.

After accomplishing all the above tests, the shipping cask was moved back to the container and the impact limiters were reinstalled. When the container lid was installed it was sealed by the IAEA safeguards inspectors who verified the nuclear material during the loading. Finally, the Health Physicist made a radiation survey of the shipping cask to complete the shipping document.

All the associated equipment used in the operation was packed back in designated boxes, according to the same original configuration.

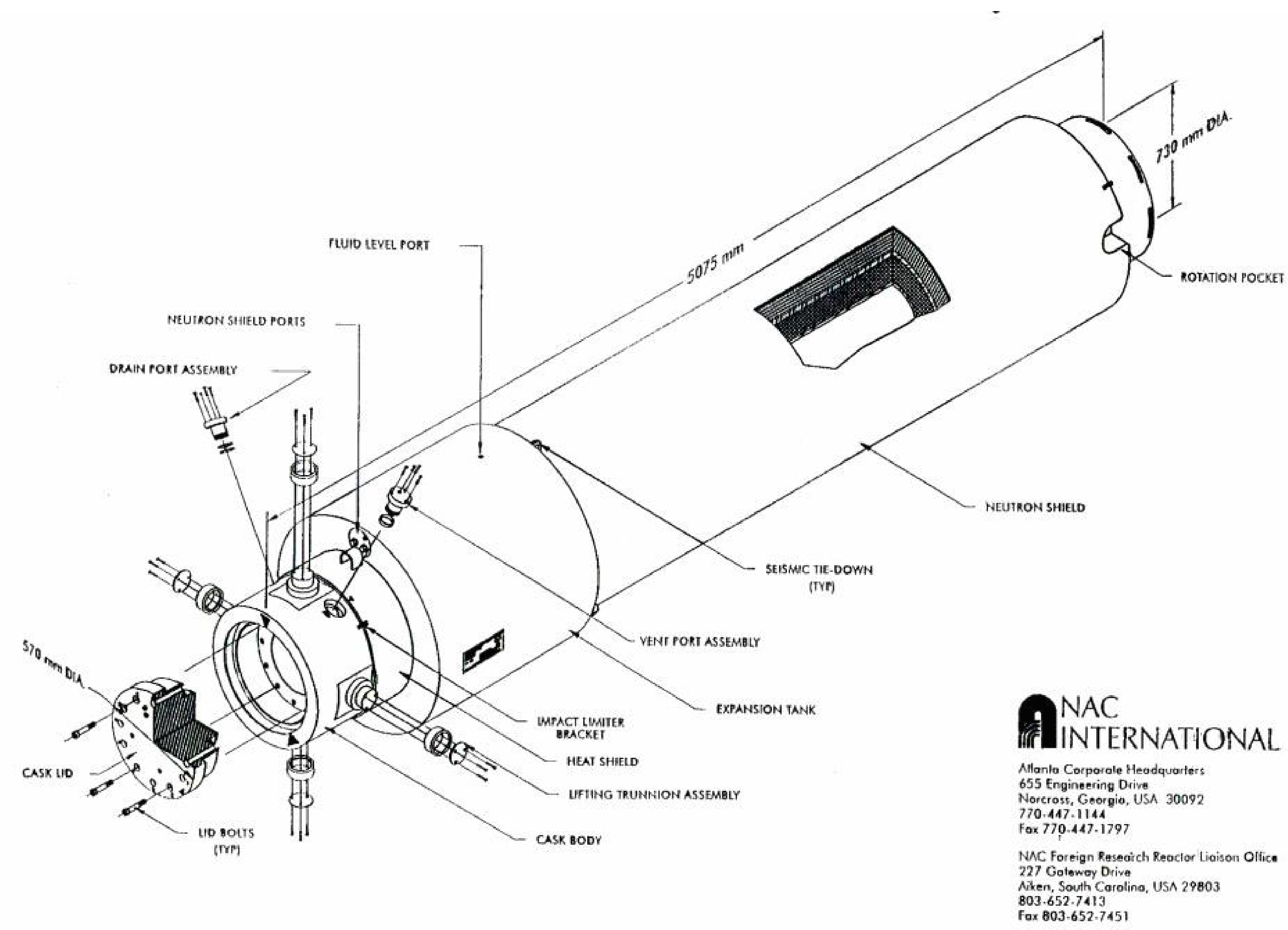

FIG. 4. NAC-LWT spent fuel transport cask body.

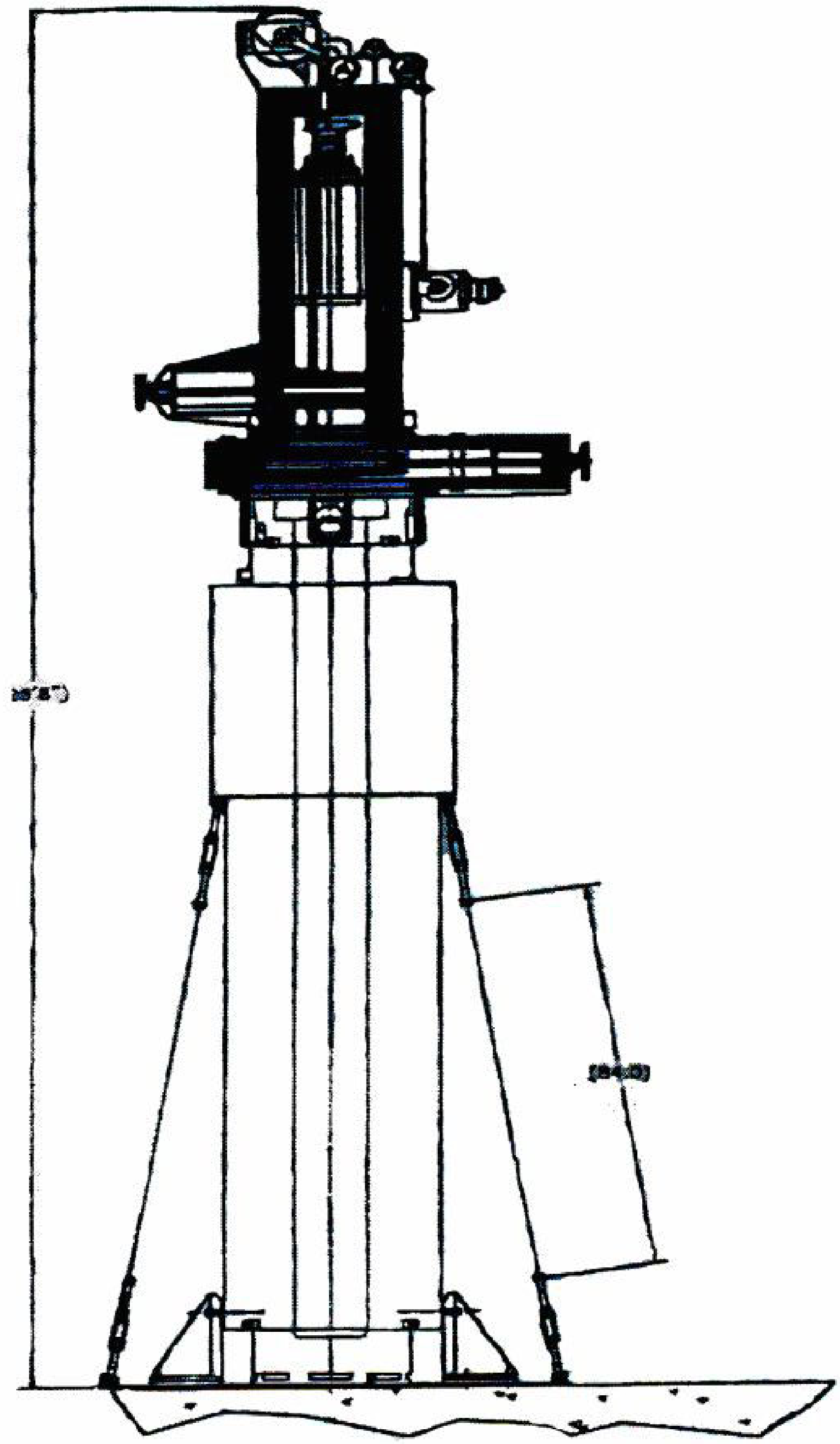

FIG. 5. MTR dry transfer system.

3.4. Transport to the Lam chabang deep seaport

After the 31 TRR-1 SNF were loaded into the NAC LWT shiping cask, sampling and measurements were required for this contract. All the measurements needed to be completed prior to departure of cask from the reactor site. They comprised:

(1) Measurement of Cs-137 activity,
(2) Sampling for microbiological contamination of the reactor site storage pool, and
(3) Conduct of a radiation survey for the cask.

Measurement of Cs-137 activity were performed by adding demineralized water to the cask and then measuring the concentration of Cs-137 in a sample of water drawn from the cask after 16 hours. In order to sampling for microbiological contamination of the reactor site storage pool, two(2) bottles containing 250 ml of water from the storage pool were taken and sent to WSRC within 24 hours of the departure of the cask from the reactor site. To conduct a measurement of a radiation survey for the cask, 0.05-0.2 mR/hr were the registered values at the cask contact.

The last operation was to transport the shipping cask from OAP to the Lam chabang deep sea port of Chonburi province, about 130 Km from Bangkok. The transport was done by using the highway from Bangkok to Lam chabang. The route was selected by the physical protection and risk prevention group among other alternatives. The convoy consisted of the container-truck, crime suppression police, NAC's security Group, vans with OAP and NAC personnel. The convoy was additionally protected by Local police Group.

On 28 of February 1999, the convoy left OAP at 08.45 am towards Lam chabang deep sea port, reaching the port after a three hours journey. After arrival at the port, the container-truck proceeded immediately to the pier where a chartered vessel was waiting to pick up the container. The ship left Lam chabang on 1 March 1999.

On 28 of April 1999, USDOE-SRS informed that the SNF packages from Asia had arrived safely at the Savannah River Site.

4. Conclusions

The whole operation was well organized and performed quite successfully although the delay of the signed contract and inspite some problem which appeared during the opearation, mainly because of the lack of any previous experience. All problem were solved in the best way possible, especially the control rod elements cropping operation, which was scheduled accordingly. In spite of this, the operation was performed without any accident (no injury or personnel overexposure). The NAC personnel showed to be very well trained and experienced..All they worked hard to finish the job on time. Finally, OAP would like to express a deep and sincere gratitude to the USDOE for the SNF take back program, and NAC International and Schenker International for the highest quality services, as well to the OAP staff who collaborated to the success of the operation.

REFERENCES

[1] SAFETY ANALYSIS REPORT FOR THE NAC LEGAL WEIGHT TRUCK CASK, Docket No.9225 T-8804 Rev.20, Nuclear Assurance Company :Atlanta,1996 .(The Company Handbook).

[2] SINDELAR, R.L., and HOWELL, J.P., Criteria for Acceptance of Off-site Spent Nuclear Fuel at SRS(U),); WSRC-TR-96-0401(U).Westinhouse Savannah River Co.: Savannah River Site, 1996.pp. 4-6.(Document prepared in connection with work done under contract No. DE-AC09-96SR 18500, U.S Department of Energy).

Venezuelan experience on return of MTR spent fuel from the RV-1 research reactor to the United States of America

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Abstract. The RV-1 Research Reactor of Instituto Venezolano de Investigaciones Científicas (IVIC) is a pool type reactor of General Electric design operating from 1961 to 1991, at 3 MW of thermal power. In its life, 76 MTR fuel assemblies composed the total inventory. The enrichment of the fuel assembly was around 19.9 %, and a total of 56 fuel assemblies were eligible for the U.S. Research Reactor Spent Nuclear Fuel (RRSNF) acceptance program. They were returned to the U.S. in two shipments. In 1998, 54 fuel assemblies were shipped in two casks. The shipment of the last two fuel assemblies had to postponed due to concerns about the cask certifications for fuel with identified cladding defects. They were shipped in 1999. Actually, in the facility remain 20 fuel assemblies of U.K. uranium origin, with an average 1% of burn up. They are kept in storage, waiting for a final disposal. In the first shipment two IU-04 casks were used, suitable for shipment of MTR fuel but not certificated to transport fuel assembly with cladding defects. In the second shipment it was used the GE-2000 to transport the two fuel assemblies with cladding defects. Both shipments were performed together with spent fuel from other research reactors in South America, which complicated formal procedures in Puerto Cabello Port. The tight time schedule also put a lot of stress on the transporters, IVIC authorities, and on the reactor operators. After overcoming some difficulties in the facility, the two shipments were arranged and they arrived safely at the Savannah River Site.

1. Introduction

Venezuela signed the US Atoms for Peace Program in June 1955. The construction of the research reactor building started in 1958, with a 1.5 million US Dollars total cost including a load of 36 fuel assemblies. These fuel assemblies were manufactured by General Electric Co. with U.S. enriched Uranium. The RV-1 was a Material Test Reactor (MTR). The first criticality occurred on 12 July 1960 and it was designed to operate at 3 MW. Since its beginning, until its shut down in 1991, it was operated 2 300 times, with a 30% of utilization factor. In these 31 years of operation the RV-1 was used to perform research in nuclear and solid state physics, radiochemistry, production of some radioisotopes and as neutron source to the scientific community.

In 1973, the Venezuelan Government decided to buy 20 fuel assemblies more. These fuel assemblies were manufactured in Spain by the Junta Nuclear of Spain (J.E.N.) using U.S. enriched Uranium. The last batch of 20 fuel assemblies was bought from United Kingdom Atomic Energy with United Kingdom Uranium origin.

The reactor was last operated in March 1991, at which time all fuel was removed from the reactor core. The decision to permanently shutdown the RV-1 was made in 1997.

The total spent nuclear fuel inventory of the reactor comprised 76 MTR fuel assemblies. Of this total, 56 were eligible for the U.S. RRSNF acceptance program and were returned in two shipments. The first one occurred on 25 September 1998, when 54 fuel assemblies were shipped to the U.S. and the second shippment, with the last two assemblies, occurred on 11 November 1999. Actually, 20 fuel assemblies remain in the facility, 18 in wet storage and 2 in dry storage. All they have U.K. uranium origin.

The Venezuelan government approved in 2000, the conversion of the RV-1 Research Reactor into an industrial gamma irradiator plant. On September 2004, IVIC received the authorization to operate the industrial irradiation plant with a 1 MCi capacity.

1.1. General description of the facility

The RV-1 research reactor was a 3 MW light water moderated and cooled reactor of the pool type. The basic core arrangement shown in Figure 1 consisted of a 9 x 7 array of 3 inch square core elements subdivided into three sections by two one inch gaps for the four control blades. The core contained a 6 by 5 array of aluminum alloy fuel assemblies surrounded by 3 inches of graphite reflector elements on all sides. The remaining row contained five radiation baskets, completing the core configuration.

One core position contained the servo-control element designed to compensate for small changes in reactivity. The reactor was contained in a stainless steel lined pool nine feet in diameter and thirty-four and one half feet deep. The core was supported, twenty-four feet below the water level, by an aluminium suspension frame. This water provided the necessary shielding above the reactor during operation and also for transfer of the fuel elements, control rods, and vertical experiments from the reactor to the storage pool via a fuel transfer canal. The reactor was designed by General Electric Company to operate with forced circulation.

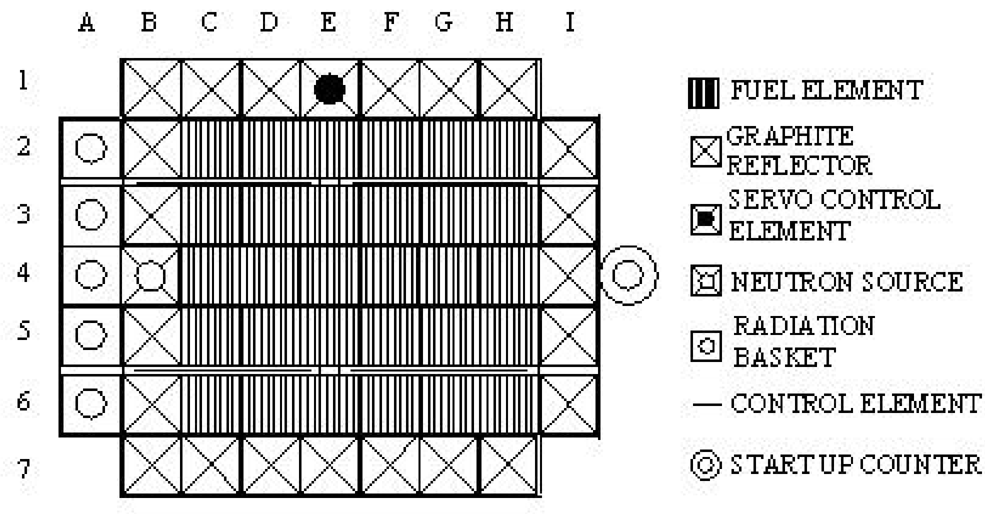

FIG. 1. RV 1 Core arrangement.

1.2. Fresh fuel assemblies

The fuel assembly consists of 10 equispaced fuel plates held vertically between two aluminium structural side plates. The fuel plates are of the flat sandwich type, assembled using picture frame techniques. The fuel meat is an uranium-aluminum alloy of about 28 w/o uranium, 20% enriched. Tables 1 and 2 summarize the description of the fresh fuel elements (plates) and assemblies that were used in the reactor. Table 3 summarizes the pre-irradiation history of any batch of fuel assemblies shipped to the USA, and Table 4 summarizes the post-irradiation history of the batches of fuel assemblies shipped to the USA.

TABLE 1. FUEL ELEMENT DESCRIPTION [1]

Fuel element type (curved or flat plate, disc, rod, tube, etc.)	Plate
Nominal dimensions of element (include clad and bond, cm)	63.50 x 7.046 x 0.2515
Nominal total weight of fuel element (g)	356.05
Nominal dimensions of fuel meat (cm)	61.0 x 6.08 x 0.175
Nominal total weight of fuel meat (g)	227.47
Chemical form of fuel meat	U-Al_x-alloy
Weight of total U (g ± g uncertainty)	68.18 (+ 0.29, -0.55)
Weight of U-235 (g ± g uncertainty)	13.58 (+ 0.06, -0.11)
Alloy or compound material, weight (g)	Aluminum, 136.1
Dispersing material, weight (g)	Aluminum, 91.37
Cladding material & method of sealing	Aluminum
Clad thickness (cm), total clad weight (g)	0.038, 128.58

TABLE 2: FUEL 'ASSEMBLY' DESCRIPTION [1]

Number of elements per assembly	10 plates
Over-all dimensions (cm)	100.33 x 7.73 x 7.73
Over-all weight (g)	5596.5
Total weight of U (g ± g uncertainty)	681.76 (+ 2.99, -5.55)
Total weight of U^{235} (g ± g uncertainty)	135.81 (+ 0.59, -1.11)
Chemical form of fuel meat	U-Al_x-alloy
Enrichment (% ± % uncertainty)	19.92 ± 0.02
Canning material	N/A
Canning dimensions (cm), weight (g)	N/A
Method of can sealing (screw, weld, etc.)	N/A
Side plate material	Aluminum
Side plate - dimensions (cm), weight per plate (g)	2, 76.2 x 7.73 x 0.48, 644.0
Spacer material	N/A
Spacer - dimensions (cm), weight per plate (g)	N/A
End box or fitting material	Aluminum Alloy
End box or fitting dimensions (cm), weight (g)	2, 14.56 x 6.6 x 6.6, ~374
Other structural material in assembly e.g. dummy plates, thermocouples, etc. (include quantity, dimensions, and weight (g))	N/A

TABLE 3. PRE-IRRADIATION HISTORY OF THE FUEL ASSEMBLIES [1]

			Pre-Irradiation	
Shipment No.	Batch No.	Total fuel assembly	Total U grams	Total U-235 grams
I	1	34	23 177.96	4 617.0
	2	20	15 163.23	2 700.0
II	3	2	1 358.6	270.63

TABLE 4. POST-IRRADIATION HISTORY OF THE FUEL ASSEMBLIES [1]

			Post- Irradiation					
Shipment No.	Batch No.	Total fuel assembly	U grams	U-235 grams	Pu-239 grams	Burnup average %	Total Decay Heat (watts)as of (date) 10/31/97	Cooling time (days) average as of: (date) 10/31/97
I	1	34	22 322.88	3 613.54	77.40	21.73	24.51	5 278.89
	2	20	14 954.08	2 454.44	19.94	9.1	8.59	2 940.05
II	3	2	1 328.75	235.38	2.74	14.77	0.80	4 483.00

1.3. *Description of fuel storage facility and facilities for cask loading and shipping preparations*

Figure 2 is a perspective view of the reactor building showing the reactor core, the containment shell, the arrangement of the reactor with its experimental facilities, and the fuel transfer canal. The crane capacity is 20 tonnes (short tonne) and the lift is 15 feet over the top of the reactor. The floor capacity was enough to withstand the fuel cask. The reactor building has a truck entrance door to make easier to work with heavy things. The spent nuclear fuel is stored in aluminium storage racks in the transfer canal. The transfer canal is designed for temporary wet storage for the RV-1 fuel assembly irradiated, and it can accomodate up to 180 fuel assemblies in 5 racks. Fig. 3 shows, another view of the fuel transfer canal and one of the storage racks.

1.4. *Facilities for cask loading and shipping preparations*

On 20-21 November 1997, a working team from the U.S Department of Energy Savannah River Operation Office, Sandia National Laboratories, Westinghouse Savannah River Company and U.S. Embassy Point of Contact visited IVIC to initiate discussions and to gather technical information on the RV-1 reactor spent nuclear fuel (SNF) eligible for shipment to the United State. All SNF and fresh material eligible for shipment were visually inspected.

On the transfer canal there were 56 assemblies, 2 fresh fuel plates, and 6 coupons cut from the 2 fresh plates available for shipment. An additional 18 SNF assemblies of United Kingdom origin were also stored in the same transfer canal, but in separate storage racks

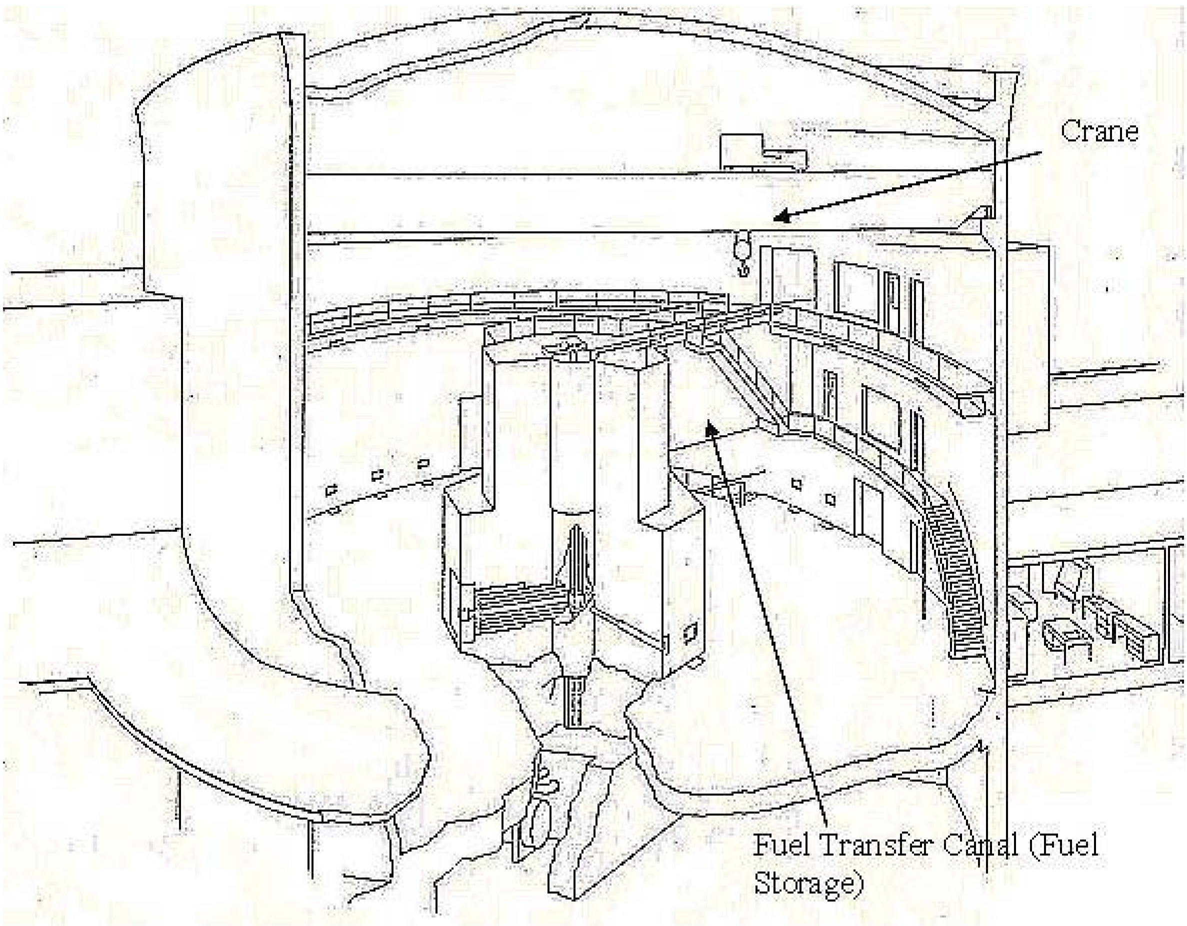

FIG. 2. Perspective of reactor building.

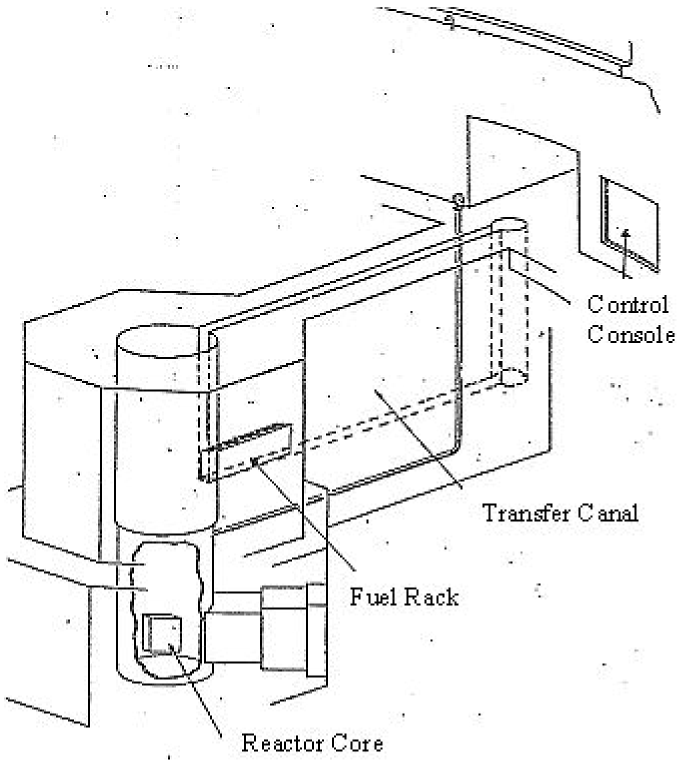

FIG. 3.The transfer canal of RV 1.

Tha facility has a 20 tonne polar bridge crane in the reactor room that could be used for all necessary loading activities. An additional 1 tonne hoist lift is available for use over the transfer canal. The transfer canal is 5 meters deep, 0.3 meters wide, and 9.7 meters long. The storage racks in the canal are designed to store the assemblies two across. At the end of the transfer canal there is a 1 meter diameter transfer well. The head room from the top of the transfer canal to the bottom of the bridge crane hook is about 4 meters. The hook does not extend directly over the transfer well and a transfer cask would need to be placed on a wheeled truck and moved over the well. Immediately above the outer edge of the transfer well there is a ceiling beam at a height of about 2.3 m. In the reactor pool there is a ledge about 0.75m wide that might hold a basket for loading. The ledge may hold a weight of about 100 kg. On the floor of the reactor room there is ample room for bringing in a transport cask and conducting loading operations. The outside access door to the reactor is about 4m wide and 5 m high. Outside of the reactor building there is ample room for equipment and vehicle staging. There are no overhead obstructions.

The transfer well also has a transfer tube that runs to a storage pool located outside of the reactor building, several meters from the outside access door to the reactor. The outdoor storage pool is located below ground level. The storage pool is approximately 2.5 m x 3.5 m with a 6 m depth. The transfer tube allows wet transfer of the fuel out of the transfer canal. There are two valves in the transfer tube, located at about 4m deep of the outdoor storage pool, to permit transfer between the transfer well and the pool. The two valves provide separation in flow between the pool and the transfer well for fuel transfer. Both valves are remotely operated and have not been used in 36 years. IVIC personnel expressed great concern about the operation of these valves. The concern is that the valves will either break while opening or not seal completely when closing. The outdoor pool is located about 5 m below the bottom of the transfer canal, and a leak in the valves could result in the transfer canal and reactor pool wall completely draining and overflowing outside.

The pool can be accessed from the front and from one of the sides. Other than the height restrictions, there is adequate space around the storage pool for final loading into an ISO transportation container.

IVIC has also the instruments and personnel necessary to support the fuel loading operation.

1.5. Assessment of fuel conditions

The U.S. DOE Trip Report Conclusions fulfilled on 20-21 November 1997 were the following:

- The reactor pool and transfer canal water chemistry are well maintained.
- Corrosion in the form of blisters was noted on two assemblies (FR36 and 6).
- All other SNF assemblies appeared to be in good condition.
- The 2 fresh plates and the 6 coupons are in dry storage.

2. National legislation

In Venezuela, the Direction of Nuclear Affairs was the Regulatory Body and determined that we would have to comply with the following regulations:

(a) **Decree 2.210: Technical Regulations and Procedures for the Use of Radioactive Material – April 23, 1992, and;**
(b) **Regulations for the Safe Transport of Radioactive Material, Regulations and Guidance, IAEA Safety Series No. 6. Venezuelan COVENIN rule 2026-87.**

On July 07, 1998 the Regulatory Body received the entire document requested to deal with the Fuel Exportation Permit. On July 17, 1998 IVIC received the Fuel Exportation Permit.

3. Institutional and political aspects

In September 1996, the Director of IVIC received a letter from the First Secretary of the Embassy of the United States of America, with the purpose to visit the reactor facility. In September 1996 the First Secretary visited the reactor facility and sent to Mr. Paolo Traversa a new letter dated 29 November 1996, informing that the twenty fuel assemblies purchased from Spain in May 1973 had been determined to be U.S. obligated, and were therefore eligible for return to the United States under the Department of Energy's Spent Fuel acceptance program.

On 30 September 1997, a fax was received proposing a U.S. DOE delegation to visit the RV-1 research reactor, with the purpose to examine and evaluate the conditions of the fuel elements. The visit was accepted and, it took place on 20 -21 November.

On 3 October 1997, the Director of IVIC received the Appendix A, Spent Nuclear Fuel Acceptance Criteria that was filled out by thetechnical staff of the reactor, and sent back for last comments on 20 January 1998.

On 15 December 1997, the Director of IVIC received a draft of the contract to be signed by DOE and IVIC, for the acceptance of Spent Nuclear Fuel.

On 29 December 1997, the Director of IVIC received the notification that Edlow International Company had been selected as DOE Transportation Contractor to coordinate the shipment from the RV-1 reactor facility to the Savannah River Site.

On 30 January 1998, Edlow International Company informed Mr. Traversa that Transnucleaire was subcontracted for supply the transportation casks and the draft schedule for the shipment plus fuel loading of two IU-04 cask with 56 assemblies.

On 2-6 February 1998, the staff of Transnucleaire made a technical visit to the RV-1 facility.

On 12 June 1998, the Government Security Board integrated by a different Ministry approved the exportation of the 56 SNF to the U.S.A.

On 19 June 1998, the Director of IVIC signed the Contract DE-AC09-98SR18935 between DOE and IVIC.

On 7 July 1998, the Director of IVIC requested to the Regulatory Body the export license for the 56 SNF.

In July 1998, Mr. Traversa, with all the information and the complexity of the task, prepared a detailed procedure to assure that all the steps in the process were performed safely and properly. This procedures included preparation of the facility and the fuel assemblies for shipment, personnel training, receipt and testing of the shipping cask, movement of the cask within the facility, loading of fuel into the cask, closure, sealing and testing of the cask, preparation for shipment, transport operations from RV-1 reactor facility to Puerto Cabello Port, radiation protection activities during the entire process and emergency plans.

On 19 July 1998, Transnucleaire sent the two IU-04 empty casks from France on a regular line vessel.

On 23 July 1998, the two IU-04 casks were validated in Venezuela, after being validated in the United States of America on the 17th of July 1998.

On 24 July 1998, the Director of IVIC received a notification from DOE about the temporary postponement of the shipment of SNF from Venezuela due to circumstances beyond the control of the U.S. DOE, related with inability to obtain transportation package certificates for all packages used in the shipment.

On 8 August 1998, the two IU-04 empty casks arrived at Puerto Cabello Port.

On 10 August 1998, the Director of IVIC received from DOE the evaluation of the two damaged assemblies, FR36 and 6, and the possibility to use the IU-04 cask to transport these assemblies at that time. DOE considered different options and finally decided to conduct the shipment without the two assemblies (FR36 and 6), which were left to be picked up at a later date, when transportation issues were resolved. These two additional assemblies ewre picked up one year later.

On 28 August 1998, the Director of IVIC received from DOE the authorization to ship the SNF to Savannah River Site (SRS).

On 26 September 1998, the MV “Blue Sky” picked up the two containers with the 54 irradiated fuel assemblies loaded into two IU-04 casks.

3.1. Economical aspects

It is important to explain that Venezuela is a developing country, considered as an Upper Middle Income Economy based on World Bank report, 1994, and independently of this category the U.S. DOE covered all the expenses of the shipment operations.

4. Safeguards

Based on the safeguards agreement between the Republic Bolivarian of Venezuela and the International Atomic Energy (IAEA), and considering that the nuclear material was under IAEA’s safeguard during its entire utilization and decay period at the RV-1 research reactor, the Director of IVIC requested an IAEA inspector to make the Physical Inventory Verification (PIV) prior to loading the spent fuel into the shipping container at IVIC.

On 18 September 1998, previous to start the load of the fuel in the shipping casks the IAEA inspector verified the physical inventory and all the numbers of the fuel assemblies that would be sent to United States of America.

After the shipment, the reactor operator completed the Physical Inventory Listing (PIL), the Material Balance Report (MBR) and the Inventory Change report (ICR), and these documents were sent to the IAEA Safeguards Department by the Venezuelan Authority.

5. Public acceptance

One important aspect during the operation was the IVIC's ability to handle the media with respect to public reaction to the transportation of nuclear material. The public in Venezuela is not negative to nuclear energy, but the IVIC Authority decided to maintain a low profile during the entire operation, and considering that the media could publish some information about the operation, the IVIC Authority planned to call the mass media to inform the public and politicians about the importance and benefits of this initiative; as well as of all the safety and security measurements being taken for the handling and transportation of the spent nuclear fuel. The reaction during all the operation and transportation was practically null. At the transportation date, only a regional newspaper published a note, which was supplied by Puerto Cabello Captain Port. The Puerto Cabello Port is the only port in Venezuela authorized to accept dangerous material in transit, and it is located approximately 210 kilometers (131,25 mi) from IVIC.

To maintain the privacy, instead of using the commercial pier of Puerto Cabello Port; it was decided to use the Venezuelan navy pier.

6. Nuclear safety

IVIC received from Transnucleaire the Safety Analysis Report of the IU-04 transport cask, shown in Fig. 4, to demonstrate its sub-criticality with all different types of baskets used to load the irradiated MTR flat plate fuel elements.

For the shipment Transnucleaire used two IU-04 casks. The first cask used the AA-267 basket an the second cask used the TN-9083 basket. The AA-267 basket has 44 compartments and the TN-9083 basket has 36 compartments.

The criticality study was based on the calculation of the effective multiplication factor K_{eff} of an individual package, with regular moderation and reflection conditions. Table 5 summarizes the result of the criticality study for the two types of baskets. The various media constituting the packaging and the basket were assumed to have the minimum densities specified in charters 1 and 2 [2]. The fuel elements were described as homogeneous fissile media equivalent to an infinite array of flat plates immersed in water.

The composition of the fissile medium was defined by ignoring the uranium 238 and the volume occupied by the uranium 235 (i.e. by increasing the aluminium density) in the U-Al core of the fuel elements plate.

The calculation method used by Transnucleaire Cask Engineering Department was initially based on determining cross-sections of the fissile medium which were representative of the assembly, using APOLLO 1 code [2] The cross-sections were then input into the MORET III code [2] to determine the effective multiplication factor K_{eff} The maximum value obtained for $K_{eff} + 3\sigma$ was equal to **0.948** in the most unfavourable case for the AA-267 basket and **0.940** for the TN-9083 basket [2].

Table 5 shows the RV-1 fuel elements characteristics loaded in the IU-04 casks compared with the safety-criticality study fulfilled by Transnucleaire. It is possible to see that all RV-1 fuel elements values are lower than the most unfavourable case estimated in the criticality study and the real number of assemblies loaded in any cask was far below the maximum allowable number.

Figures 5 and 6 show a description of baskets AA-267 and TN-9083 and the real arrangement of fuel assemblies on each one.

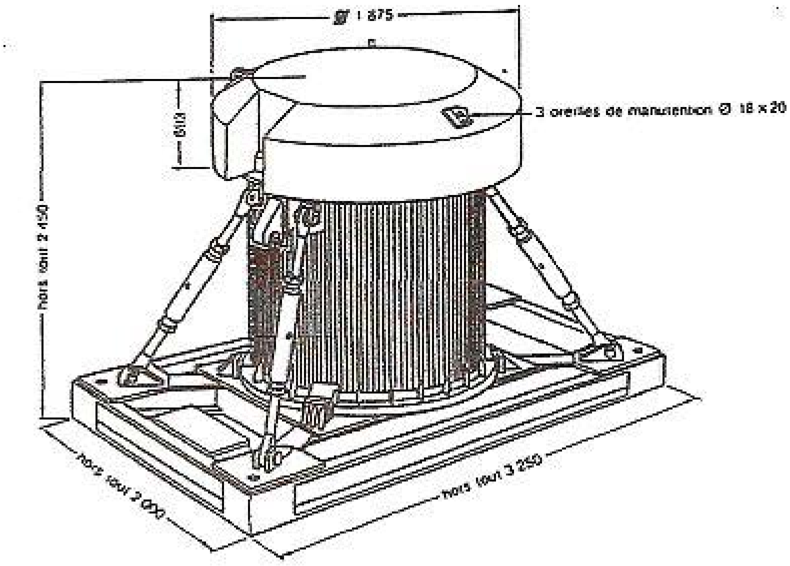

FIG. 4. :The IU-04 transport cask.

TABLE 5. RESULTS OF THE SAFETY-CRITICALITY STUDY OF THE IU-04 CASKS WITH ITS BASKET COMPARED WITH THE CHARACTERISTICS OF THE RV-1 FUEL ASSEMBLIES AND FUEL LOADED.

	Conclusions of Criticality Study Basket Type [2]		RV-1 Fuel Assemblies Loaded Basket Type	
Fuel element characteristics	AA-267	TN-9083	AA-267	TN-9083
Concentration of U-235 in the alloy (g/cm^3)	≤ 0.40	≤ 0.40	0.10	0.10
U-235 enrichment (%)	≤ 100	≤ 100	19.92	19.92
Thickness of a plate fuel core (mm)	≤ 2.1	≤ 2.1	1.75	1.75
Thickness of a plate cladding (mm)	≥ 0.2	≥ 0.2	0.765	0.765
Irradiated fuel assembly per cask	≤ 40 U-Al	≤ 36 U-Al	26	28
Maximum residual power per assembly	80 watts	130 watt	0.82 watts	0.95 watts
Materials Basket Composition	Boronated Aluminium at 2% of Boron at least	Stainless steel 304L with 0.5% of Boron	Boronated Aluminium at 2% of Boron at least	Stainless steel 304L with 0.5% of Boron

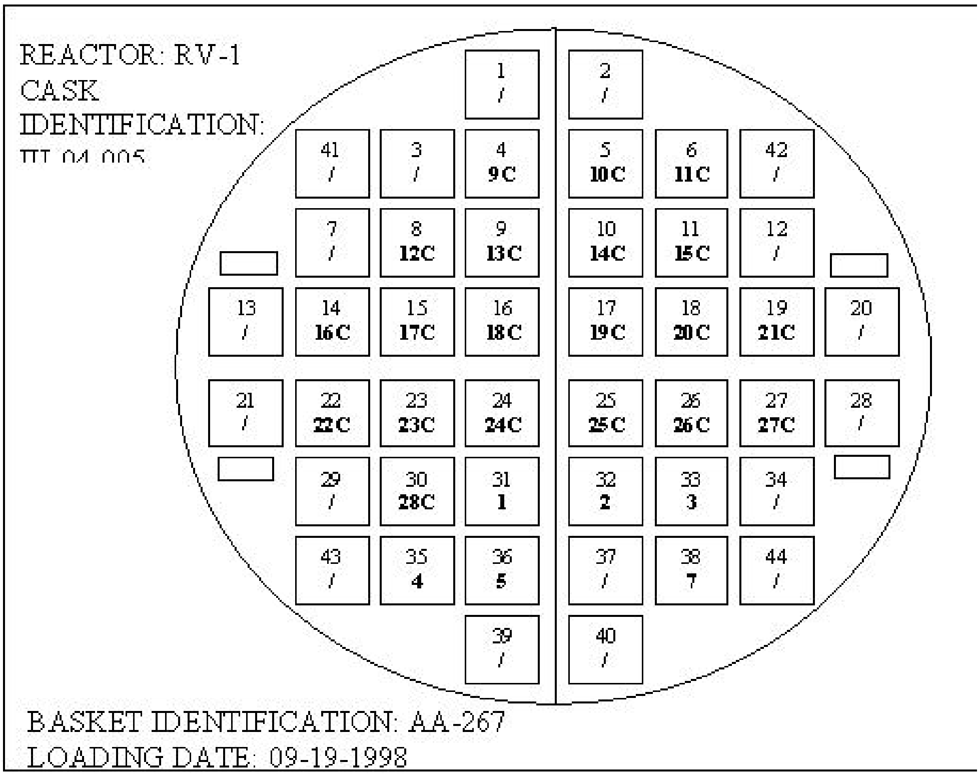

FIG. 5. Fuel arrangement within basket AA-267.

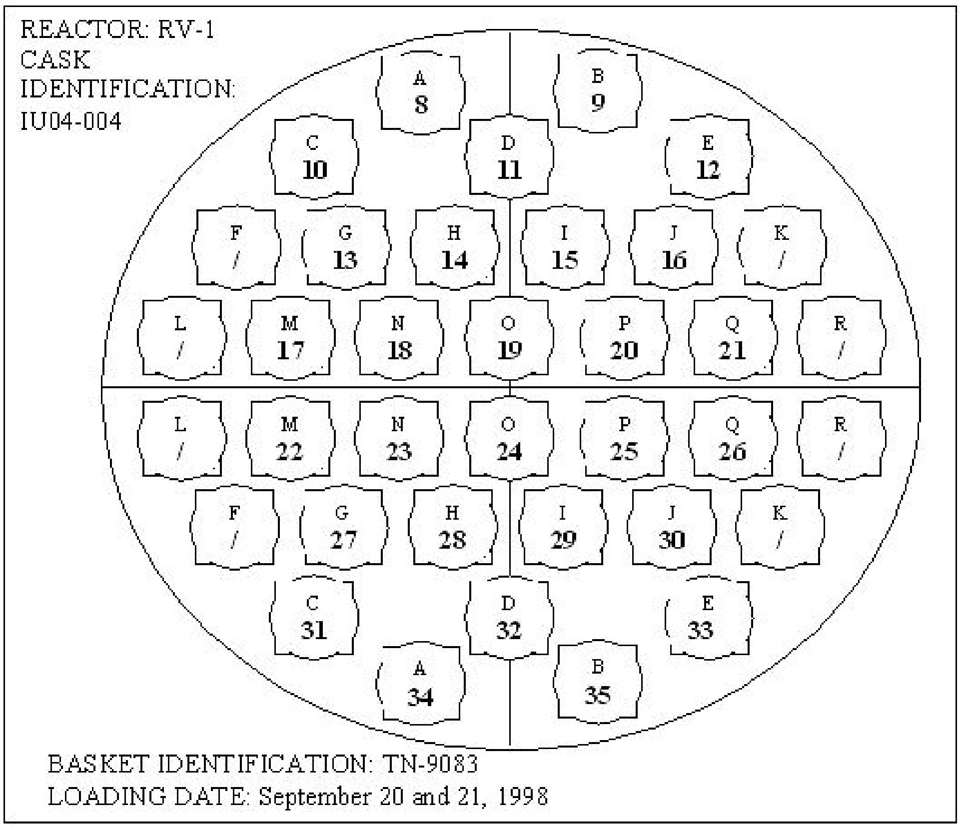

FIG. 6. Fuel arrangement within basket TN-9083.

7. Quality assurance

There was no requirement that all activities to return the spent fuel to its country of origin should be conducted according to an approved QA plan. The Reactor Manager was designated by the Director of IVIC as responsible for the preparation of all documents and procedures necessary to obtain the approval from the Nuclear Safety Commission and Radiological Protection Officer.This procedures describe the organization, preliminaries testing, preparation of the fuel for shipment, customs formalities required to obtain the release when the casks arrive at the port, inspections during receipt, transport to the facility, loading fuel baskets in the casks, preparations for shipment, loading shipping cask on the trailer, transport to the port and customs formalities. Other documents and procedures were prepared by the Radiological Protection Officer related the health physic protection of all personnel, equipment acceptance and receiving survey, and contamination checks at the receiving and pre-shipment testing of the loaded casks.

The activities under the jurisdiction of each supervisor were specified in every document taking into account the documents written by Transnucleaire and the draft of the IAEA-TECDOC “Guidelines Document on Technical and Administrative Preparations Required for Shipment of Research Reactor Spent Fuel to Its Country of Origin” of January 1997, as well as other documents received at the Interregional Training Course on Technical and Administrative Preparations Required for Shipment of Research Reactor Spent Fuel to Its Country of Origin, in January 1997, organized by Argonne National Laboratories, the IAEA and the Government of the United States of America. The IVIC organizational chart to carry out the return of the spent fuel to U.S.A. is shown in Fig. 7.

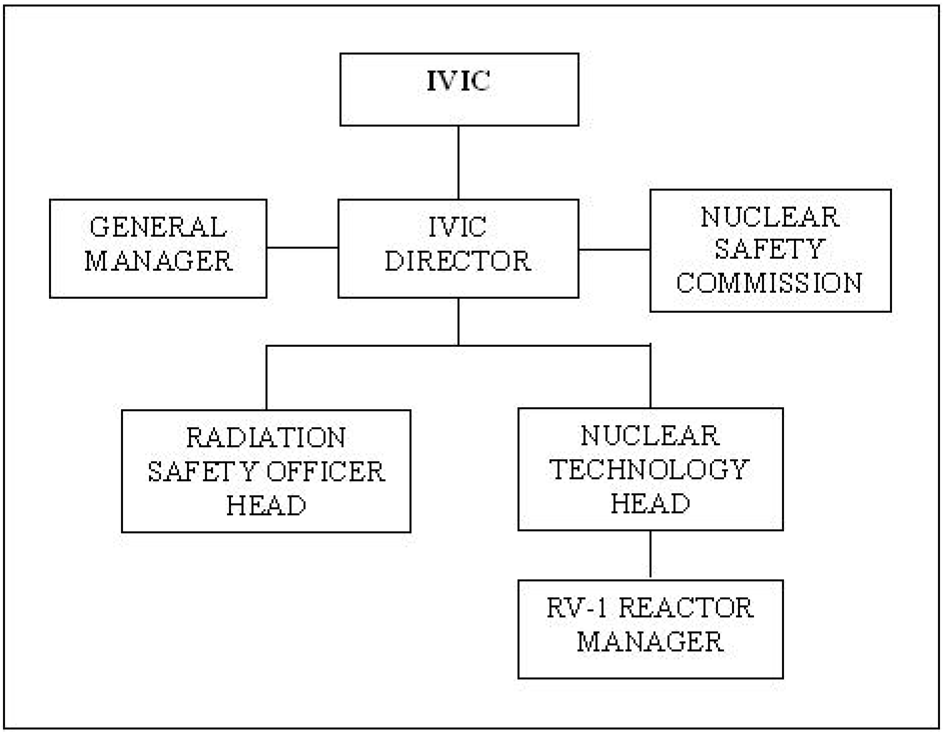

FIG. 7. IVIC organization chart.

8. Loading procedures

8.1. Pre-shipment activities documentation

From 13 to 24 January 1997, a reactor operator was trained at Argonne National Laboratory, Illinois, USA and this operator was in charge to complete the "Appendix A Agreement: Spent Nuclear Fuel Acceptance Criteria" with the important support from the Savannah River Site counterparts. When DOE accepted the Appendix A, a written Authorization to ship was issued to IVIC, and a tentative shipping date was established.

The IVIC Authorities submitted to the Venezuela Government the request to approve the return of the 56 spent fuel assemblies of USA origin, which was given on June of 1998. Then DOE and the Director of IVIC signed the Contract to transfer the 56 fuel assemblies from IVIC to Savannah River Site.

At this moment the IVIC authority submitted to the Nuclear Affairs Direction (Regulator) of the Ministry of Energy and Mine all the necessary documents in order to get the Export License for Nuclear Material. These documents were: fuel assemblies specific data with the irradiation history, a complete set of drawings of the cask, safety analysis report of the IU-04 package (shielding and criticality calculations, practices and procedures of loading, characteristics of the cask, thermal analysis, study of IU-04 package health physics), valid license of the cask from US NRC and US DOT provided by Transnucleaire and Edlow International Company. It is importante to remember that Edlow International Company was selected by DOE for the transportation management services and the casks were provided, in the first shipment by Transnucleaire, for the transport of 54 fuel assemblies on September, 1998 and, for the second shipment by General Electric, for the transport of 2 fuel assemblies with corrosion in the form of blisters in November 1999. These two damaged fuel assemblies were shipped without encapsulation.

The Nuclear Affairs Direction was in charge to issue permission for the ship to enter the Venezuelan territory with the nuclear cargo from Uruguay in the first shipment and from Brazil in the second shipment.

8.2. Loading of the casks shipment

Due to limited dimensions in the transfer canal to place the IU-04 cask, Transnucleaire designed and manufactured a specific transfer system in order to handle the fuel assembly from the storage rack and to load them into the IU-04 cask. This transfer system was composed of two main equipments: a shielded bell and a cask loading equipment.

The shielded bell was used to handle each fuel assembly. It is composed of an external shielding protection, made of steel and lead, and a handle device to catch the top of the fuel assembly. Each fuel element was transferred under water from the storage rack placed in the transfer canal down to a well (5 meters deep) where the fuel element was placed on a support. The shielded bell was handled in the water over the support and placed on it. Then, the handled device was operated manually, to grab the fuel element and lift it inside of the shielded bell. The bottom of the latter was closed and then the fuel assembly was safely transferred to the loading area.

The cask loading equipment was used to protect the operators against radiation, while loading the fuel elements into the IU-04 cask. It was composed mainly of a thick stainless steel shell, surrounded by lead, placed and bolted over the top of the cask. On the bottom part, there was an elastomer gasket to prevent from water leak. Then, the cavity and cask loading equipment was filled with water. The cask was placed in a configuration for loading, similar to a loading configuration in a large pool. Inside this equipment, a transfer device allowed to support the fuel assembly.The shielded bell was handled in the cask loading equipment, full of water, over the transfer device. The door of the shielded bell was opened and the fuel assembly was placed on the transfer element. After taking out the shielded bell, the fuel assembly was manually handled from the transfer device to a previous designated place into the IU-04 cask.

What foolows is a description of the procedures used to load the two IU04 casks. They are based on the loading procedures written by Transnucleaire, and includ a clear definition of the responsibilities assigned for Transnucleaire (TN) and IVIC staff, in all the tasks,.

8.2.1. Preliminary operations before the arrival of the container

8.2.1.1. Check the reception area of the equipment container, packaging container and the IU04 packaging in the reactor building. Task responsibility: IVIC – TN.

8.2.2. Unloading of the transport vehicle. Responsibility for these operations: IVIC – TN.

8.2.2.1. Read the transport file. Transfer the equipment container into the building reactor.
8.2.2.2. Open the top and doors of the packaging container.
8.2.2.3. Remove the shock absorber (1500 Kg) of the cask and transfer it to the wood block placed in the building reactor. An overview of the RV-1 Reactor Building is given in Fig. 8.

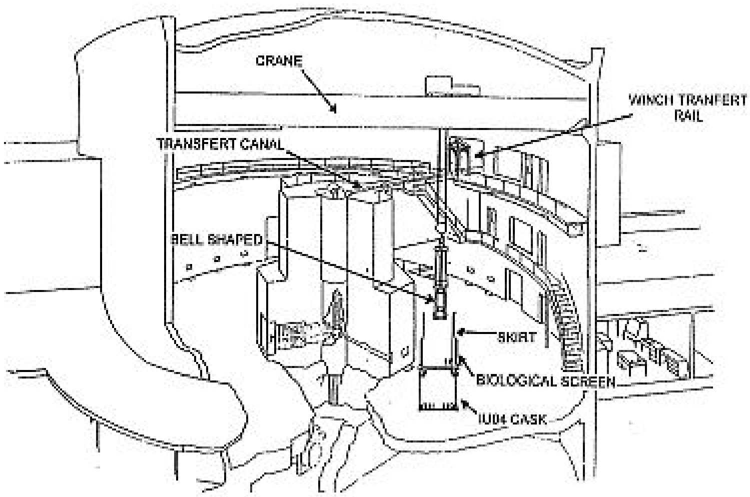

FIG. 8. . IU04 package in the RV-1 reactor building [3].

8.2.2.4. Transfer of the casks (19,4 Tonnes each) to the reactor ground, as illustrated in Fig. 9. This task could not be performed because the reactor crane didn't have enough capacity to lift up the cask. The it was decided to leave the casks on the respective containers and trailers.
8.2.2.5. Unload all interfaces and tools.
8.2.2.6. Inspect the casks and the accompanying material.
8.2.2.7. Perform an inventory of the content.

8.2.3. Setting of the TN winch and transfer fuel stool

8.2.3.1. Set the duck boards across the canal, following as shown on Fig. 10. Task responsibility: IVIC – TN.
8.2.3.2. Assemble the winch transfer rails. Task responsibility: IVIC – TN.
8.2.3.3. Assemble the winch on the reactor ground. Task responsibility: TN.

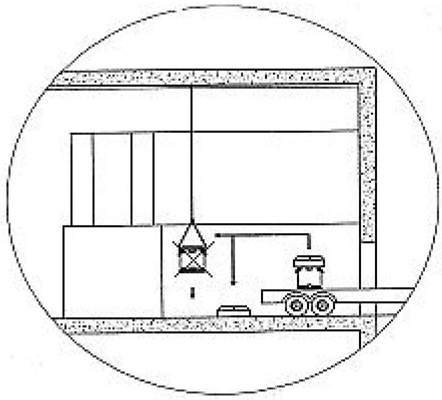

FIG. 9. Illustration to handle theIU-04 cask inside the reactor building.

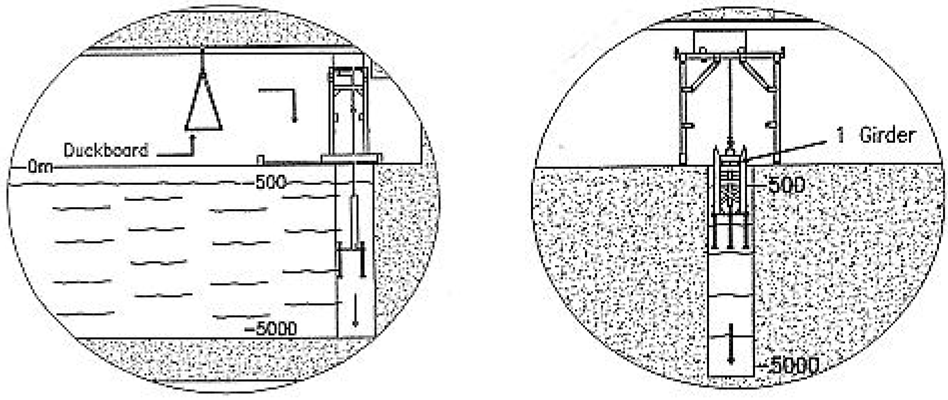

FIG. 10. Illustration to set the duckboard across the canal.

8.2.3.4. Transfer the winch on their rails above the well. Task responsibility: IVIC – TN.
8.2.3.5. Transfer the stool above the well. Task responsibility: IVIC.
8.2.3.6. Set one girder across the well. Task responsibility: IVIC – TN.
8.2.3.7. Set the stool on the girder. Task responsibility: IVIC – TN.
8.2.3.8. Set a long sling on the stool beam. Task responsibility: TN.
8.2.3.9. Set the fuel transfer stool at the bottom of the well. Task responsibility: IVIC.
Note: Maximum weight of each item: 3 Tonnes.

8.2.4. Preparing the packaging

8.2.4.1. Remove the clamping disk (250 Kg). Task responsibility: IVIC – TN.
8.2.4.2. Store the clamping disk on the block. Task responsibility: TN.
8.2.4.3. Fill the cavity of the first cask with water. See Fig. 11-a) for reference. Task responsibility: IVIC.
8.2.4.4. Assemble the skirt (2 Tonnes) with biological screen on IU-04, using Fig. 11-b) as reference. Task responsibility: IVIC – TN.
8.2.4.5. Lift the lid (2 Tonnes) along a few millimeters. Task responsibility: IVIC – TN.

8.2.4.6. Fill the loading skirt with water using Fig. 12a) as reference. Task responsibility: IVIC.
8.2.4.7. Remove, clean and storage the lid.. Task responsibility: IVIC – TN.
8.2.4.8. Set the fuel transfer cell in the skirt, as shown in Fig. 12b)..

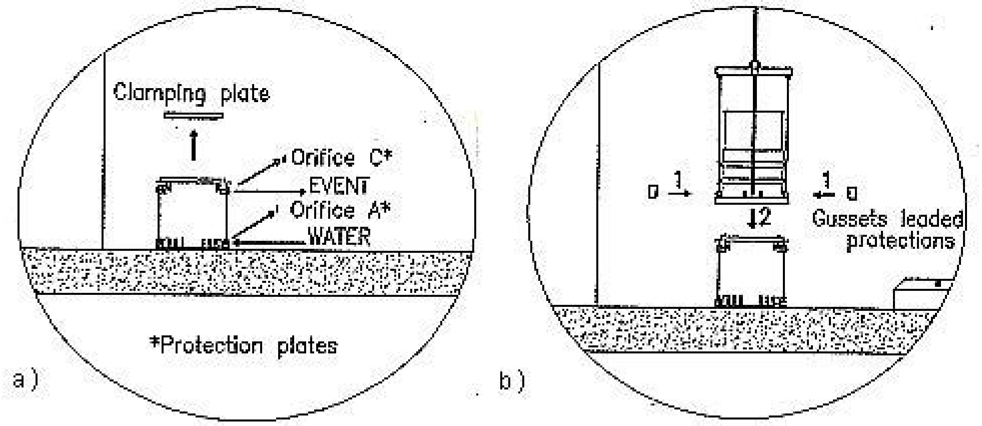

FIG. 11. The transport cask IU-40 (a) and the loading skirt (b).

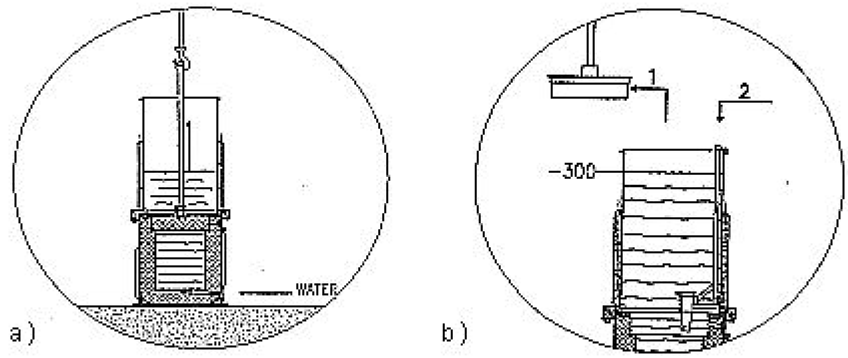

FIG. 12. Setting the cask loading arrangement.

8.2.5. Loading of a packaging

8.2.5.1. Transfer of a fuel element to the transfer well, as shown in Fig. 13a) Task responsibility: IVIC.
8.2.5.2. Place the shielded bell (3 Tonnes) in the duckboard across the canal, as shown in Fig. 13b) Task responsibility: TN.
8.2.5.3. Set the short sling on the handling beam of the shielded bell. Task responsibility: TN.
8.2.5.4. Transfer the shielded bell above the well using the winch, as illustrated on Fig. 14. Task responsibility: TN.
8.2.5.5. Set the two girders across the well. Task responsibility: TN.
8.2.5.6. Set the bell shape item on the two girders. Task responsibility: TN.

8.2.5.7. Transfer the shielded bell t the two girders, using Fig. 15 as reference. Task responsibility: TN.
8.2.5.8. Open the shielded bell hatch, (see Fig. 15). Task responsibility: TN.
8.2.5.9. Go down the hook and hanging of the fuel element. Task responsibility: IVIC – TN.

8.2.5.10. Go up of the hook with the fuel element in the shielded bell. Task responsibility: IVIC – TN.

8.2.5.11. Close the bell shape item hatch. Task responsibility: TN.

8.2.5.12. Move up the shielded bell to the two girders. Task responsibility: IVIC – TN.

8.2.5.13. Drain, clean and dry the shielded bell surface. Task responsibility: IVIC – TN.

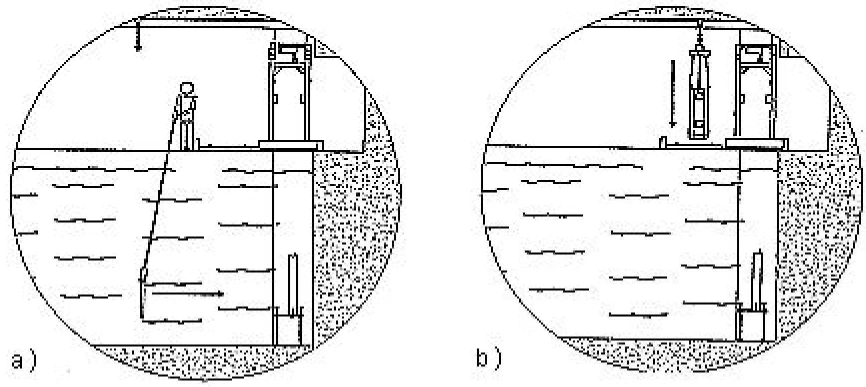

FIG. 13. Loading the fuel assembly into the shielded bell.

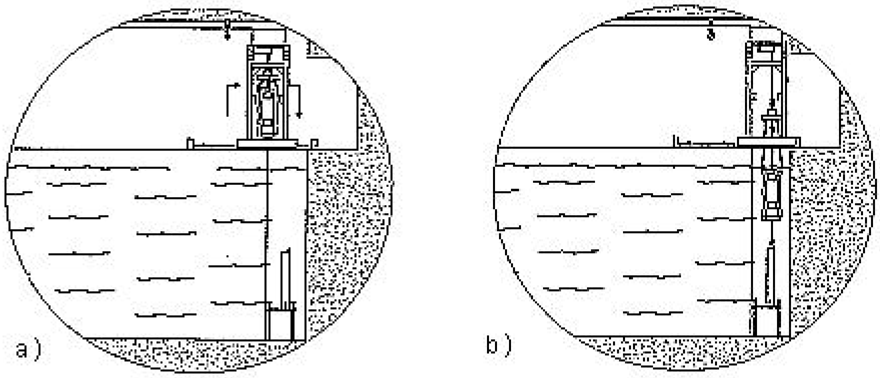

FIG. 14. Placing the shielded bell in the transfer well.

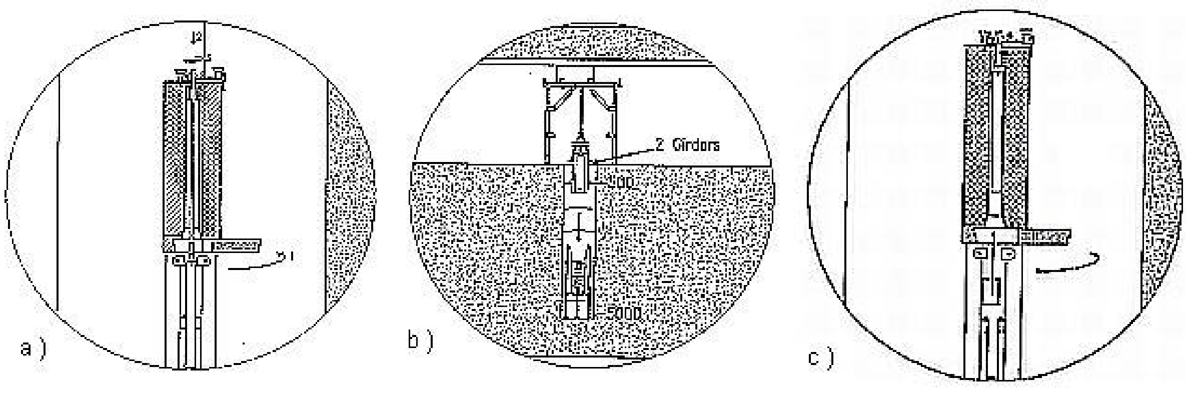

FIG. 15. Adjusting and loading the shielded bell in the transfer well.

8.2.5.14. Remove the upper part of the shielded bell's hook and the long slings, using Fig. 16 as reference. Task responsibility: TN.
8.2.5.15. Set the dripping pan on the shielded bell and transfer it to the duckboard, (See Fig. 16 c). Task responsibility: TN.

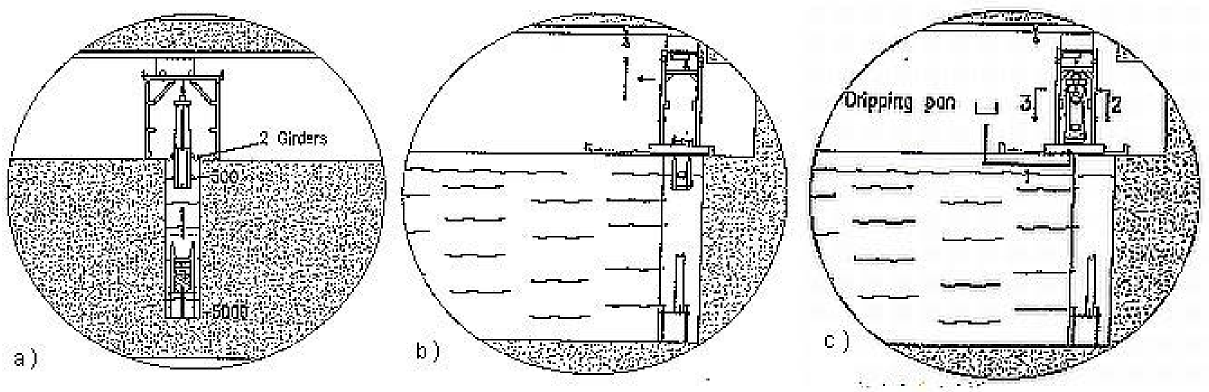

FIG. 16. Removing the shielded bell from the transfer well.

8.2.5.16. Set the upper part of the shielded bell's hook, set the long slings. Task responsibility: TN.
8.2.5.17. Transfer the shileded bell above the loading skirt, as shown in Fig. 17. Task responsibility: IVIC – TN.

8.2.5.18. Remove the dripping pan and the shielded bell positioned above the loading skirt. Task responsibility: IVIC – TN.
8.2.5.19. Immerse the shileded bell; using Fig. 18 a) as reference. Task responsibility: IVIC.
8.2.5.20. Open the shielded bell and lower the hook with the fuel element in the transfer cell, as shown in Fig. 18 b).
8.2.5.21. Lift the shielded bell with its hook in down position. Task responsibility: TN.
8.2.5.22. Drain, clean and dry theshielded bell
8.2.5.23. Set the dripping pan (see Fig. 18 c) on the shielded bell and transfer it to the duckboard,. Task responsibility: IVIC.

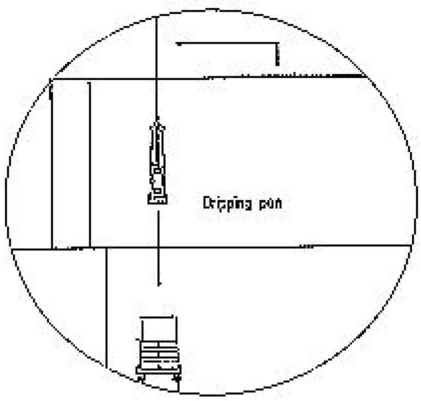

FIG. 17. Adjusting the shielded bell to the transport cask.

8.2.5.24. Manually load of the fuel element in the respective position of the cask, as shown in Fig. 18 d), Task responsibility: TN. Weights: Fuel element: 6 kg.
8.2.5.25. Repeat all the operations for each fuel element. Task responsibility: IVIC – TN.

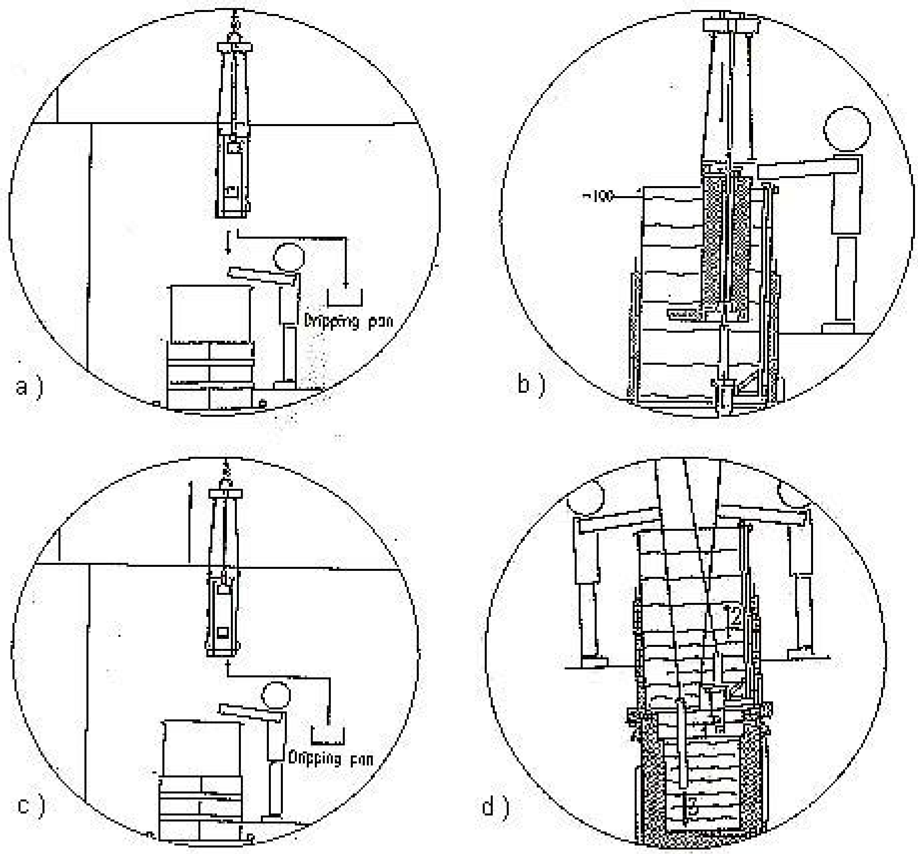

FIG. 18. Loading the fuel element into the transport cask.

8.2.6. Close the transport cask

8.2.6.1. With the IU04 loaded remove the fuel transfer cell and set the lid in the loading skirt above the cask, as shown in Fig. 19 a). Task responsibility: IVIC – TN.
8.2.6.2. Empty the loading skirt with the lid just above the cask and set the lid, as shown in Fig. 19 b). Task responsibility: IVIC – TN.
8.2.6.3. Remove the loading skirt with biological screen; as shown in Fig. 19 c). Task responsibility: IVIC – TN.
8.2.6.4. Fit the clamping disk and tighten the 18 nuts of the clamping disk (tightening torque: 1000 Nm); using Fig. 20 a) as reference. Task responsibility: TN.
8.2.6.5. Perform liquid sample, drain and dry by vacuum the cavity. Task responsibility: IVIC – TN.

8.2.7. Leaktighness test

8.2.7.1. Perform a leak tightness test of the cavity and inter-seal space according to the instruction manual. Task responsibility: TN.

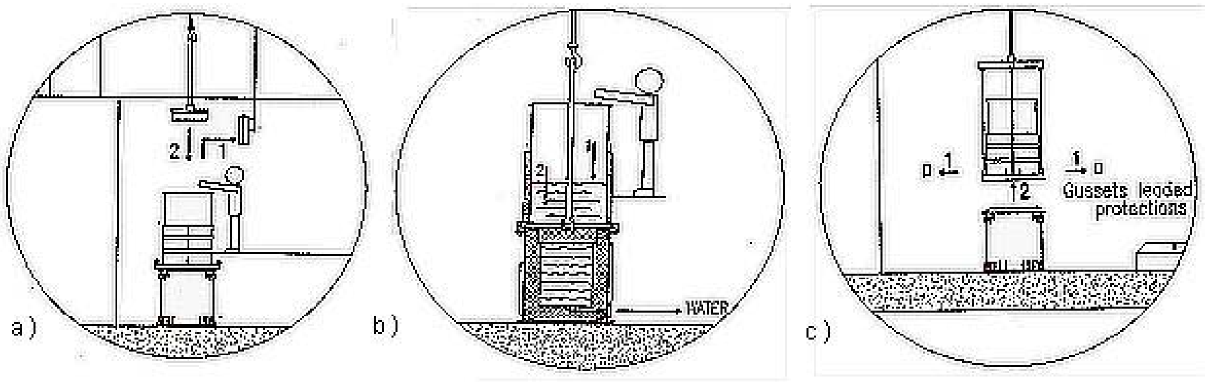

FIG. 19. Closing the Transfer cask.

8.2.8. Preparing packaging before shipment

8.2.8.1. Set the protection plates; as shown in Fig. 20 b). Task responsibility: TN.
8.2.8.2. Fit the shock absorbing cover. Task responsibility: IVIC.
8.2.8.3. Put the security seals. Task responsibility: TN.

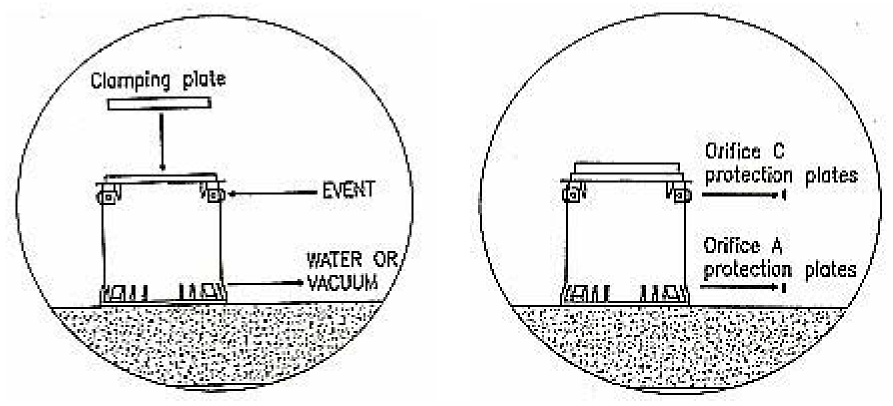

FIG. 20. Installation of a) clamping disk; and b) protection plates.

8.2.9. Inspections before shipment

8.2.9.1. Perform and inventory of the content of the accompanying equipment. Task responsibility: IVIC – TN.
8.2.9.2. Perform the radiological control of the equipment according the transport regulations. Task responsibility: IVIC.
8.2.9.3. Transfer the equipment into the container. Task responsibility: IVIC – TN.

8.2.9.4. Perform the radiological control of the cask and the container according to the transport regulations. Task responsibility: IVIC.
8.2.9.5. Transfer the equipment container to the trailer. Task responsibility: IVIC.

9. Logistic and supporting infra structure

9.1. Radiation protection

The health physics staff was in charge of all the procedures preparation related to radiation protection activities during the entire process. Their activities covered all health physics actions needed for the RRSNF shipment, from the arrival of the casks at the La Guaira port until the ship steams left the Puerto Cabello port.

The main participation of the health physics staff took part during the loading of the casks in two aspects: the control of the dose rates to protect the personnel, and in the maintenance of a communications control room emergency communications with ambulance personnel, fire department and the national security, if necessary.

Moreover, the health physics staff carried out: the radionuclide sampling test, and the package radiation & contamination levels certificates in accordance with the Appendix B "Transportation Package Acceptance Criteria" of the DOE Contract.

9.2. Facilities requirements for the loading periods

Before of the reception of the casks at the facility it was necessary to check the weight resistance of the reactor floor, so as to enable storage of the cask plus the transfer systems. This study permitted to determine that the floor couldl withstand the cask weight but it was impossible to prove the crane capacity with a similar cask weight. This was the only problem detected. It was considered a minor problem and caused no delay in the schedule of the operation.

At the reactor facility it was necessary to supply: liquid nitrogen, demineralized water for different uses, vinyl film for lid storage, scaffoldings for assistants to access on top of the transfer skirt and to manufacture a stainless steel dripping pan to collect liquid effluents.

For the transport period it was necessary to contract a truck with a short trailer to achieve the reactor main door.

9.3. Responsible entity for the equipments to loading operations

Transnucleaire in its interface procedure [4] defined the responsibilities for each entity, as shown in Table 6. During the loading operations the IVIC and Transnucleaire personal worked in two shifts

TABLE 6. DEFINITION OF RESPONSIBILITIES TO SUPPLY THE NECESSARY EQUIPMENT FOR THE LOADING OPERATION.

Equipments	
IVIC	**TRANSNUCLEAIRE**
Lifting tool	Sling
Specific key	Lifting beam
Specific lifting beam	Short sling
IVIC crane	TN winch
Pipe	Specific tool
Long sling	Hook tool
Sample bottle	Leak tightness equipment
Vacuum pump	Labels
Radiological tools	

10. RRSNF preparation for shipment

The casks cavity was flooded with demineralised water to carry out the Radionuclide Sampling Test in accordance with the specifications and requirements provided in Appendix B, of the DOE Contract. On 24 September 1998, the two casks were sampled.

The initial water sample was measured at time "0 hours", the second water sample was taken six hours after the initial sample and the last water sample was taken twelve hours after the initial water sample. In all the samples, it was analyzed the Cs-137 activity in 1 liter of water drained from the cask and determined the increase in the Cs-137 activity, between the initial sample and the final sample. The increase in the activity level was 1,62 dpm/ml for the first cask and 1,7 dpm/ml for the second cask. The acceptable increase established by DOE-SRS for the cask type IU-04 was less than 726 dpm/ml, therefore it was concluded that the fuels in the two casks were considered as "not failled".

To remove the water from the shipping cask, pressurized air was blown into the cavity followed by a vacuum dried process. Then, the cask cavity was filed with helium and the closure lid was leak tested. The shipping casks were sampled to determine non fixed external radioactive contamination on surfaces carried out by IVIC Health Physic Department.

Other pre-shipment inspections were carried out as: dose rate at the contact of the packaging, at 1 m of the surface of the packaging, surface temperatures, seals and labels and the shipping documents.

11. Description of transport operations

The last operation was to transport the shipping casks from the IVIC to Puerto Cabello Port.

11.1. Previous considerations

The reactor is located on the outskirts of the city of Caracas. For the shipment there were two options. One was the port of La Guaira which is to the north of Caracas, about 50 km from the reactor. There is basically only one route from the reactor to La Guaira and this route has had some problems with the use of a mayor bridge. This route was ruled out.

The other option, the selected one, was is Puerto Cabello, located west of Caracas and about 210 km from the reactor. There are several choices of route and major roads to Puerto Cabello from the reactor. IVIC personnel considered Puerto Cabello the preferred choice. Another advantage of Puerto Cabello is that it has a naval facility which would enhance security during the ship loading.

11.2. Transport operations

With the approval by DOE, the transport was done by using the Panamerican road for 11 Km and following the Autopista Del Centro highway from Caracas to Puerto Cabello for about 200 km. The road was selected by the Physical Protection and Risk Prevention. The convoy consisting of the two containers-trucks with the shipping casks and one extra truck for any emergency, Military National Guard cars, vans with IVIC and Edlow personnel, Health Physics Service, Regulator Personnel and the Miranda Emergency team. The convoy was protected by Military National Guard special team in the entire road. The pier of the navy base was used for docking the ship rented for the operation, which arrived with some RRSNF from Uruguay. Fig. 21 shows a resume of the timeline with the main events related to the shipment of the first 54 spent fuels..

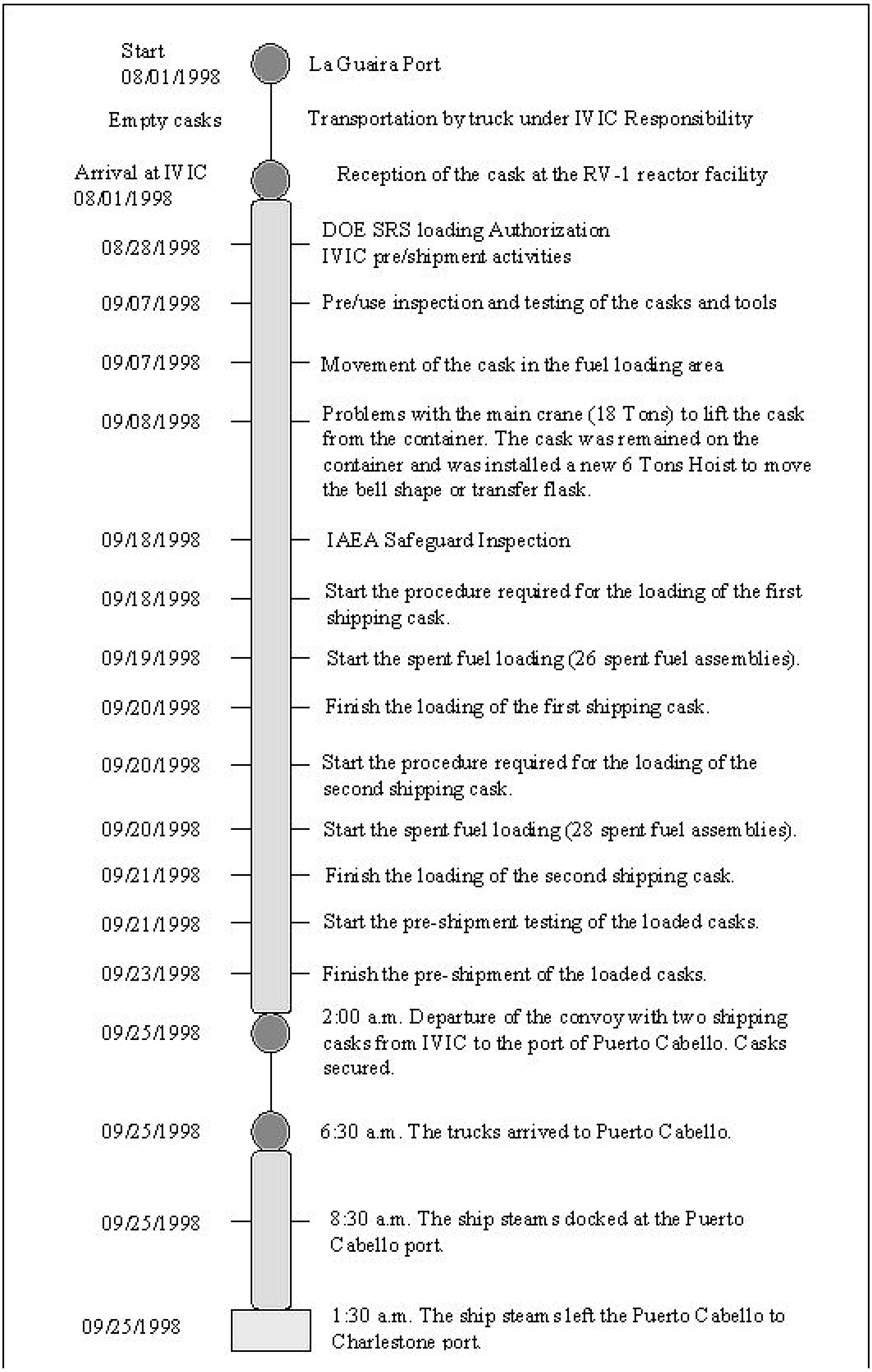

FIG. 21. Timeline of main events related to the first shipment of RRSNF from Venezuela.

11.3. IVIC Certification

At the end of the loading operations of the two Transnucleaire IU-04 transport casks IVIC certifies:

(a) The physical condition of the spent nuclear fuel complies with all applicable Transport Package certification requirements.
(b) The spent nuclear fuel is structurally sound such that it will not change shape during handling.
(c) The spent nuclear fuel is not bent or deformed such that could cause interference with a cask basket cask insert surfaces.
(d) The increase in radioactivity measured by the leakage tests prescribed by Section C of Appendix B is within specifications for both casks.
(e) The spent nuclear fuel was removed from the reactor core without any cladding failure and without any other failure of material condition that may require special handling or packaging for transportation or storage.

This document was signed by Mr. Paolo Traversa, Reactor Manager of IVIC.

12. Conclusions

We can say that the main conclusions of the two shippping operations are:

- The reduction of the fuel assemblies inventory in the transfer canal permitted to reduce the water contamination from 7 Bq/L to 1 Bq/L of Cs-137 with water recirculation. This reduction permitted to reduce the contamination levels in the deionizer resins and the contamination in the liquid effluent.

- After thirty years maintaining the spent fuel assemblies in wet storage, with a good condition of the water, we found the beginning of the fuel plate's corrosion. The return of the old fuel assemblies permitted to reduce important contaminations problems and its costs for canning or to consider other solutions.

- The Interregional Training Course on "Technical and Administrative Preparations Required for Shipment of Research Reactor Spent Fuel to Its Country of Origen" organized by Argonne National Laboratories with a Cooperative Program of the IAEA and the Government of the United State of America was essential to help the reactor personal to prepare all the documents required for the shipping operation, including the Appendix "A" and the Appendix "B" of the Agreement.

- The very heavy equipment involved in the spent nuclear fuel shipment program needs special consideration because most research reactors do not have the capability to handle such heavt loads. Moreover, the majority of the research reactor facilities are more than thirty years old, and some parts of the crane has aged and if attempted to be used close to the limit of the capacity it probably wont work.

- Another important factor that contributed to the success of the operation was the established communication stragtegy, which allowed to rapidly solve any problems between the different actors. This permitted to make easy all interface activities including::

  (a) US-DOE-SR Contract Management Division with IVIC Director supported by IVIC Legal Department to study and to sign the Contract for the acceptance of the Spent Nuclear Fuel;

  (b) US-DOE-SR Staff with the Eng. Paolo Traversa, RV-1 Reactor Head for all technical and operational aspects of the shipment;

  (c) Eng. Paolo Traversa, RV-1 Reactor Head with Edlow International Company for all the arrangements for the transportation plan;

(d) Eng. Paolo Traversa, RV-1 Reactor Head with Transnucleaire and General Electric Co. for the pre-shipment, cask loading and preparation of the transort cask.before shipment, for the.first and second shipments, respectively;

(e) IVIC and Nuclear Affairs Direction (the regulatory body of Venezuela) for the exportation permissions and casks validation.

(f) IVIC and Ministry of Defence for planning and providing security on the route would be used for the road shipment.

(g) Eng. Paolo Traversa, RV-1 Reactor Head and the IVIC Purchase Department for all the Custom formalities and tax exemption.

(h) IVIC and the truck company for the transportation of the containers from and to the port.

(i) The support of the Venezuelan Navy for the permit to use the pier of the navy base.

(j) Miranda Emergency was the organization in charged for any medical emergency during the loading procedures and the transportation to the Navy Port.

(k) The IVIC Health Physics Department, in charge of the all aspects of the radiation protection for all the Spent Nuclear Fuel Shipment activities.

(l) The IAEA Department of Safeguards for the Physical Inventory Verification (PIV) of the fuel assemblies during the Nuclear Material Transaction between Venezuela and USA Government.

- The reduction of fuel inventory in the RV-1 building was a key step to avoid a nuclear criticality event. This permitted to convert the facility into and industrial gamma irradiation plant with a radiation shielding to 3 million Curies. This new use of the reactor facility was licensed by the Regulatory Body on September, 2004. Figure 22 shows the physical location of all protective features of the Cobalt-60 Irradiator located in the RV-1 reactor basement and the general layout of the irradiator [5].

- The financial and logistic support of DOE, was of fundamental importance to cary out both shipping operations. Withour it would be impossible to return the spent nuclear to the USA.

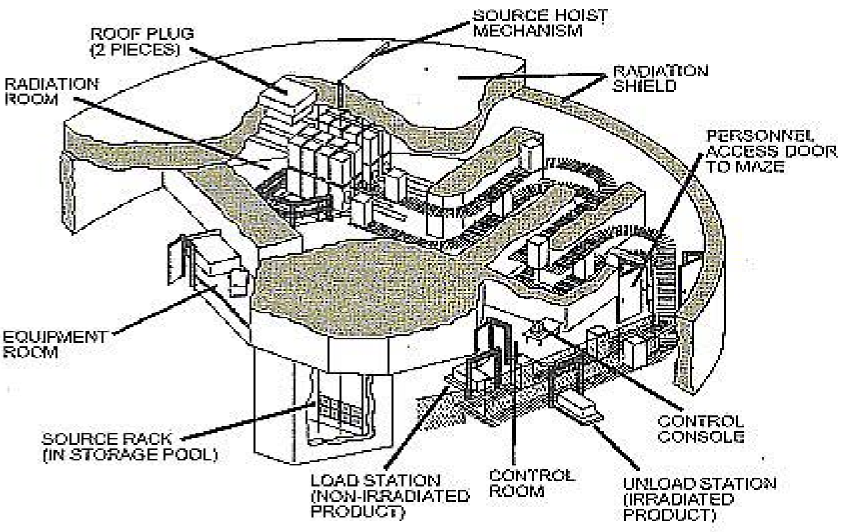

FIG. 22. General irradiator layout at the reactor basement.

REFERENCES

[1] US DEPARTMENT OF ENERGY, Appendix A, Spent Nuclear Fuel Acceptance Criteria, document DOE SRAAF-98-013 (1998).

[2] COGEMA, Safety Analysis Report IU-04 Packaging, 9686-Z Rev.6, Paris (1998).

[3] KORNISCH, O., ANNE C., MTR Transport Experiences of Transnucleaire in South America and Status on the New TN-MTR Packaging, Presented at: The 1998 International Reduced enrichment for Test Reactor Conference, Sao Paulo, Brazil (October 1998).

[4] TRANSNUCLEAIRE, Interface Procedure Loading in the IVIC Place, Ref. 10904-P-02-E, Paris (1998).

[5] MDS NORDION, Cobalt-60 Irradiator IR-216 IVIC, Operator’s Manual, Kanata (2002).

Foreign research reactor spent nuclear fuel shipment: Typical approach and methodology

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Abstract. Since the re-start of the U.S. Department of Energy (DOE) Foreign Research Reactor (FRR) Program in 1996, NAC International has been involved in a large number of Spent Nuclear Fuel (SNF) shipments from various countries. NAC has returned about 1,500 MTR and 1,200 TRIGA fuel assemblies from 20 countries.

The organization of a SNF shipment is a significant, time-consuming and unique project for a research reactor:

- Such reactors do not ship SNF on a regular basis and consequently do not have expertise in safety and security regulations
- Often the infrastructure of the facility has limitations which require special solutions
- International shipments are more complex as many more specific regulations may apply (country of origin, transit countries, country of the flag of the vessel, destination country)

Based on NAC's unique experience under the FRR program, this paper gives an overview of the activities to be implemented for the safe and secure completion of a SNF shipment. It describes a typical transport methodology, including the description of the pre-shipment planning and coordination activities which are key to the success of the shipment, and also describes the main activities related to the packaging and shipment performance.

NAC's role is to assist the shipper in understanding and meeting its obligations in the performance of a SNF shipment in conformance with all of the applicable regulations. This presentation describes NAC's capabilities and concludes with lessons learned from the performance of many safe and secure shipments of spent fuel.

The paper also provides useful recommendations and guidance for prospective future shippers of spent fuel.

1. Introduction

A typical SNF shipment requires a variety of skills and a good working knowledge of transport safety and security regulations. The shipping facility is ultimately responsible for the compliance of the transport to these regulations. For this reason, a good understanding of the tasks to be undertaken is a critical element for the success of a SNF shipping project.

After more than 10 years of implementation, the FRR program is now running very smoothly. NAC has been involved in the return shipments since the beginning of the program working in 20 different countries. NAC is now very familiar with the process which is described below. This paper describes the activities that are implemented for the performance of a shipment. It differentiates the tasks performed well in advance (called hereafter the pre-shipment activities) and the tasks accomplished prior to the shipment performance (called hereafter the shipment performance activities).

To conclude the paper, we hare very beneficial lessons learned over the years.

2. Pre- shipment activities

The main pre-shipment activities are:

- Licensing
  - Cask
  - Transport
- Cask interface with the shipping facility
- Transport logistics

2.1. Licensing

Cask licensing starts with the collection and evaluation of detailed fuel data. Each spent fuel cask design is licensed for a specific content with detailed parameters based on criticality, structural, thermal and radiological analysis (dimensions, U content, cladding thickness, decay heat, etc). Each deviation to the authorized contents parameter implies a required amendment of the cask Certificate of Compliance (CoC). Considering that for international shipments, multiple authorities are involved in the licensing of the cask, it can take up to fifteen (15) months to prepare and obtain the amended certificate complying with the fuel to be shipped. Furthermore, under the FRR program, DOE has developed a specific form entitled "Appendix A" to be filled out by the shipping facilities. The Appendix A describes the list of fuel elements to be shipped with detailed parameters. DOE will perform a fuel inspection, review the Appendix A submitted by the reactor operator, and provide approval.

Transport licensing is typically specific to the regulations of each country. Public acceptance is a major factor in the determination of the requirements. Some countries might require the preparation of an Environmental Impact Statement (EIS). The purpose of the EIS is to demonstrate that the SNF shipment will have no radiological impact on the workers and population along the shipment route. Preparation and publication of an EIS can be a time-consuming task. Other countries might limit their requirement to shorter notification periods. However, especially recently, the security surrounding a particular SNF shipment is an element to be evaluated well in advance of the shipment. Most countries have based their own regulations on the IAEA INFCIRC/225 Revision 4. However, depending on the environment and potential threats in the areas where the SNF cask(s) will transit, the shipment might be subject to additional, temporary over-regulatory requirements. Consequently, it is very important to know well in advance the security requirements which will apply for the shipment. Regular updates with competent authorities are necessary to make sure that the requirements remain unchanged at the time of the shipment performance.

2.2. Cask interface with the shipping facility

The cask interface with the shipping facility needs to be evaluated well in advance of the shipment (minimum of 6 months). A technical site assessment is performed by the cask provider to look at various site specific characteristics, such as roads, building access, crane capacity, pool dimensions, work space, support services, special site requirements, etc, and determine with the site personnel the most efficient way to perform the cask loading operations. As a result of the assessment, additional specially designed equipment may or may not be necessary for the loading operations.

Under the FRR program, two DOE facilities have been dseignated to receive the spent fuel elements: DOE Savannah River site for the MTR type fuels and DOE Idaho for the Triga type. These facilities have already assessed and included in their safety analysis the use of a list of casks including the NAC-LWT cask.

2.3. Transport logistics

Transport logistics are more and more challenging due to the reluctance of transport carriers to accept radioactive cargo and the ever-changing security requirements. For these reasons the performance of a feasibility study to confirm available modes of transportation and approved routes for the empty and loaded casks is more and more vital for the success of the project. The feasibility study involves multiple contacts with the shipping site, the competent authorities, the ports, the carriers, etc. It is also a good opportunity to re-educate the shipper on his responsibilities in regard to the SNF transport regulations. As the owner of the fuel, it is the shipper's responsibility to ensure that the transport is performed in compliance with all applicable regulations. Upon completion of the cask loading, little time is left for preparation of the cask and the shipping documentation. Informing the shipper of his responsibilities well in advance and providing a template of the shipping documents will be beneficial when it becomes time to release the shipment from the site.

3. Shipment performance activities

The main activities can be divided as follows:

- Empty cask(s) shipment from the storage facility to the research reactor
- Cask loading operations
- Loaded cask(s) shipment preparation
- Loaded cask(s) shipment execution
- Cask(s) unloading operations
- Return shipment of empty cask(s) to their storage facility.

3.1. Empty cask(s) shipment from their storage facility to the research reactor

The transport of the empty cask(s) is typically performed by truck to a sea port, then by the use of an ocean-going vessel to a port located near the shipping facility. The empty cask(s) cannot be shipped as non-contaminated since there is limited residue of radioactive material inside the cask. In spite of the fact that the quantity of radioactive material is not significant, the access to shipping lines is more and more difficult, especially if the cask is shipped in a location where radioactive shipments are not performed on a regular basis. As an example, on a recent DOE FFR spent fuel shipment originating from Greece, NAC shipped the empty casks by sea to Germany and then performed a truck shipment transiting through ten (10) different countries prior to reaching Athens. Many ships are sailing on a daily basis from the US to the Mediterranean Sea, but none of them accept the lowest level of radioactive cargo (empty cask(s)). The ultimate, but very expensive, option is the use of a chartered vessel (potentially sharing of a vessel).

3.2. Cask loading operations

Cask loading operations is a team effort performed by the reactor personnel and the cask vendor personnel who are providing technical guidance. The performance of a preliminary site assessment, as described above, has proven to be very beneficial for the smooth performance of cask loading operations by helping to avoid any unexpected events or issues.

For the NAC-LWT cask(s), it takes usually three (3) days for the loading of the first cask, including a dry run, and then two (2) days per additional cask.

3.3. Loaded cask(s) shipment preparation

Upon cask loading completion, the preparation for shipment includes the cask testing (conformance with leak tightness criteria defined in the safety analysis report) and the contamination and radiation surveys. The DOE Appendix B form includes information to be provided by the reactor prior to release for shipment. In particular, Appendix B describes the radiological requirements prior departure. A copy of the surveys is faxed to DOE prior to departure of the cask.

Per DOE requirements and US regulations, transportation and security plans will be issued at least 10 days prior to the departure of the shipment. Additional notifications will be made to the relevant authorities involved in the shipment. The shipment dates are “Safeguard Information” and cannot be disclosed except on a “need to know” basis.

Also, the shipping declaration is issued and signed by the shipper. The shipping declaration describes information in conformance with the applicable regulations, such as the shipment classification, the UN number, the activity of each cask, the transport index, the emergency contact information and the labeling of the cask. Finally, the cask and ISO containers are labeled and marked in compliance with the regulations.

Typically, NAC assists the shipper in filling out the forms and providing, as necessary, the appropriate labels and marking.

3.4. Loaded cask(s) shipment execution

The transport of the loaded cask(s) is typically performed by truck, then by the use of a chartered Irradiated Nuclear Fuel (INF) vessel and finally, in the US, by truck or train depending on the number of casks. The US port of entry for the FRR shipments is the Naval Weapons Station (NWS) in Charleston South Carolina. Triga fuel shipments are also transiting via the Savannah River Site prior to being trucked to Idaho. DOE encourages combined maritime shipments in order to limit the number of shipments transiting via the NWS in Charleston. A combined shipment presents an economical interest, as well, by sharing the cost of the vessel. Recent experience has shown that the premium for Nuclear Liability Insurance coverage might increase in the case of a combined shipment. However, except for a small minority of nuclear pools, a joint shipment will remain more economical for each individual shipper.

The shipments of the loaded cask(s) from the shipping facility to the port are typically escorted by local law enforcement or military forces provided by the shipping country. Once onboard the ship, the security requirements of the country of the flag of the vessel apply. The captain will be responsible for the implementation of the security requirements. Emergency plans are in place in case of an abnormal event. Also, at a minimum, twice every 24 hours, a tracking report of the ship is provided to the DOE. Within the US, all the SNF shipments are escorted by law enforcement.

3.5. Cask unloading operations

Once delivered to the Savannah River Site or Idaho, the cask(s) are subsequently unloaded. The DOE is now well experienced with cask unloading operations. It takes about 5 working days to unload each cask. DOE prepares the cask for return shipment including performing the radiological surveys, the preparation of the shipping documents, etc.

3.6. Return shipment of empty cask(s) to their storage facility

The last transportation phase consists of shipping the empty cask(s) back to the vendor’s storage facility.

4. Conclusion

During all these years, we have faced different issues and have improved the process to be applied to future shipments. The two main lessons learned are anticipation and responsiveness.

Anticipation is a key factor, since it allows to identify potential difficulties early enough to overcome them without significant hurdles (licensing, transport route, etc.). For this reason we strongly recommend initiating the pre-shipment activities at least 16 months in advance of the shipment schedule. More time might even be required, depending on the specific requirements of the shipping countries

Responsiveness is the second main lesson learned. Our world has been changing since 9/11. We need to be reactive and capable of adjusting quickly to new potential security requirements. When sensitivity or risk is identified, planning a shipment with back-up options is necessary to avoid significant impact on schedule.

The FRR program has been, and will continue to be, a very successful non-proliferation program. NAC is proud to contribute to this program and we stand ready to assist all research reactor managers in preparing and performing RRSNF shipment.

The U.S. Department of Energy/Idaho National Laboratory's Research Reactor Spent Nuclear Fuel acceptance programme

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1. Programme overview

The Department of Energy’s Idaho Operations Office (DOE-ID) has successfully implemented a management program that is responsible for the safe and cost – effective transportation and storage of TRIGA spent nuclear fuel at the Idaho National Laboratory (INL).

In May 1995, a Record of Decision (ROD) on the Environmental Impact Statement (EIS) for Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory (now known as the Idaho National Laboratory) Environmental Restoration and Waste Management Program was published [1]. Based on that Record of Decision, the United States Department of Energy, in consultation with the Department of Navy, adopted a policy regarding the management of existing and reasonably foreseeable inventories of spent nuclear fuel through the year 2035. The spent nuclear fuel inventory covered by this policy is generated from many different sources: DOE reactors, other government agency and university research reactors, and foreign research reactors. The policy consisted of a Department-wide decision to regionalize spent nuclear fuel management by fuel type at three DOE sites, with the INL being responsible for several spent fuel inventories, including all TRIGA research reactor spent fuel. The timing of the transport of the spent fuel between the respective sites is prioritized and scheduled based on the needs of the shipping site, fuel condition, facility availability, safety, safeguards and security concerns, budget and cost considerations, and transport logistics.

This Record of Decision was amended in late February 1996 to reflect requirements identified within an October 16, 1995 Settlement Agreement among DOE, the State of Idaho and the Department of Navy pertaining to spent nuclear fuel shipments into and out of the State of Idaho [2]. In essence, shipments of spent nuclear fuel into the State of Idaho are restricted, and tied to completion of various INL environmental restoration and radioactive waste management activities that are important to the State of Idaho.

Specific to foreign research reactor spent nuclear fuel, and in support of the 1995 Programmatic Spent Nuclear Fuel ROD, DOE then published a Record of Decision in May 2006 to implement a new foreign research reactor (FRR) spent fuel acceptance policy as identified within the Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel Environmental Impact Statement [3]. This ROD supported the DOE regionalized spent fuel management policy while providing additional information regarding the shipping of FRR spent nuclear fuel containing uranium enriched in the United States back to the United States for spent fuel management. This ROD specified that the FRR facilities were required to stop irradiating their fuels by May, 2006 and ship it to a U.S. DOE facility by

May, 2009. However, the Record of Decision was amended in 2005 to extend the Foreign Research Reactor Spent Fuel Acceptance Program to 2016, and 2019 respectively [4].

Table 1 provides a listing of all of the facilities included within these programs, while Fig. 1 provides a map identifying the countries with TRIGA foreign research reactor facilities that are eligible to participate.

TABLE 1. INL POTENTIAL SHIPPERS LIST

Domestic Shippers:
University Shippers
Cornell University*
Kansas State University (KSU)*
North Carolina State University (NCSU)
Oregon State University (OSU)
Pennsylvania State University (PSU)
Reed College
University of Arizona (UA)
University at Buffalo, State University of New York (SUNY)*
University of California-Davis (UC-Davis), formerly McClellan Air Force Base reactor)
University of California-Irvine (UC-Irvine)
University of Illinois (UI)*
University of Maryland (UM)
University of Texas (UT) at Austin
University of Texas A&M*
University of Utah (UU)
University of Wisconsin (UW)
Washington State University (WSU)
Non-University Shippers
Aerotest, Aerotest Research & Radiobiology TRIGA Reactor (ARRR)
Armed Forces Radiobiology Research Institute (AFRRI)
Argonne National Laboratory-East (ANL-E)
Argonne National Laboratory-West (ANL-W) (now known as the Materials and Fuels Complex)
Babcock & Wilcox (B&W), Lynchburg, North Carolina
DOW Chemical
General Atomics (GA)*
Hanford (HR)
Fort. St. Vrain*
Oak Ridge (OR)*
Sandia National Laboratory (SNL)
Savannah River Site (SRS)
Veterans Administration (VA)
United States Geological Service (USGS)
West Valley (WV)*
International Shippers (Foreign Research Reactors):
High-income-economy countries
Austria, Germany*, Japan*, Taiwan, Finland, Italy*, Slovenia*, United Kingdom (England)*
Other-than-high income economycountries
Bangladesh, Indonesia*, Mexico, South Korea*, Brazil, Malaysia, Philippines, Thailand, Democratic Rep. of Congo, Romania*, Turkey

* Identifies facilities that have made shipments to the INL

FIG. 1. Countries with spent nuclear fuel eligible for shipment to the INL.

2. INL research reactor spent fuel receipt preparations

During the first ten years of the Department's Spent Nuclear Fuel (SNF) Acceptance Program, DOE-ID has supported the DOE FRR program with 6 shipping campaigns, involving 8 different countries, with 15 casks containing approximately 1 500 TRIGA spent fuel assemblies; and the U.S domestic program with 7 shipping campaigns, involving 7 different research reactor facilities, with 14 casks containing approximately 700 spent fuel assemblies. All shipments have been safely received and stored at the Idaho National Laboratory (INL). Each shipment, and fuel type received, has gone through a rigorous pre-shipment preparation process that includes fuel characterization and cask shipping data, in support of criticality and facility specific safety reviews, culminating in an "authorization to ship" from the DOE-ID [5].

2.1. Receipt preparations

DOE-ID, in conjunction with the INL's Environmental Management contractor, CWI, has developed a disciplined process for completing the required activities to approve the safe receipt and storage of SNF at the INL [6]. Figure 2 depicts the flow of the basic process for prospective program participants. Two years in advance of the planned fuel shipment to the INL, an agreement on the terms of the shipment is reached. The process then proceeds in two parallel paths, an administrative path and a technical path. The administrative path is represented by the activities outlined in black on the left side of Fig. 2 and involves the formalizing of agreements, schedules and specific terms of the shipment. The areas outlined in red, on the right hand side of the figure, deal with the specific INL activities that provide the technical bases to support the safe receipt and storage of the SNF.

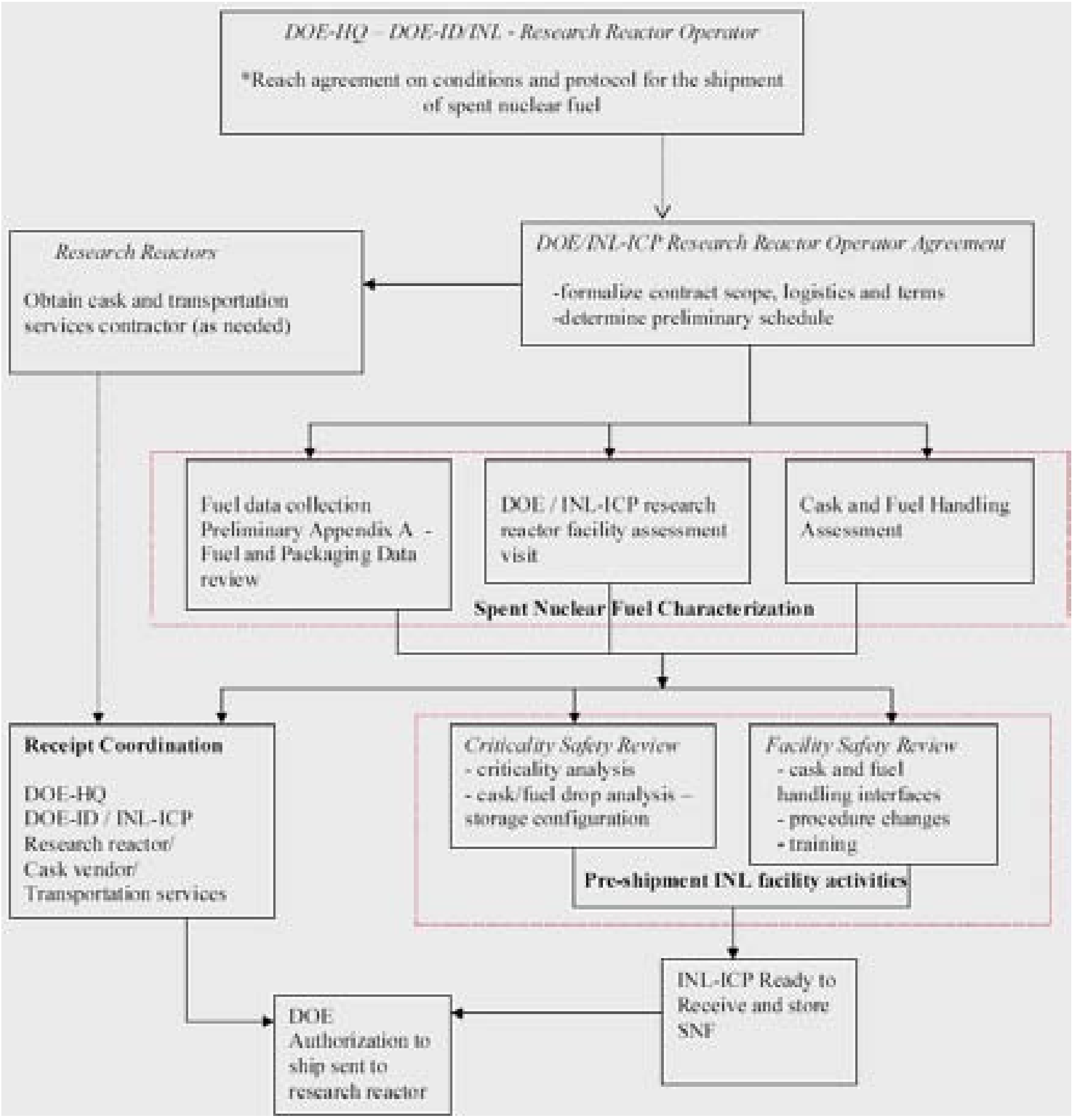

FIG. 2. INL SNF receipt prepatation flow diagram.

2.1.1. Spent nuclear fuel characterization

The first, and most important, step in the receipt preparations process is to secure an accurate characterization of the fuel. Characterization activities are segregated into three groups, which, ideally, are worked in parallel and are completed at least nine months prior to fuel handling at the INL.

- First, the characterization of fuel data required within the Appendix A attachment to the contract; the data form is known as the Fuel - Required Shippers Data (Fuel RSD). Copies of these data forms are attached to the end of this paper.

- Second, a team of DOE/INL personnel may visit the reactor facility to assess the fuel and reactor operating history and operating condition.

- And third, the final fuel characterization activity involves a review of the cask design/certification parameters, and how the fuel is to be handled. This information and data is required within the Appendix A the contract and is known as the Packaging – Required Shippers Data (Packaging RSD). Copies of these data forms are attached to the end of this paper.

The following paragraphs provide a more detailed description of these tasks.

Data Collection

Fuel data is collected and documented per the guidance provided within the Appendix A, which is, by contract, a part of DOE's agreement with the reactor facility. The reactor operator provides the Appendix A information to the INL for use in validating compliance with INL facility operations safety and authorization basis. The INL uses the reference documents, such as drawings, fuel fabrication reports, reactor operating logs, facility safety analysis reports, and others, to review the submitted data. The Appendix A is approved when all of the INL review comments have been resolved in the comment resolution cycle. Accuracy and timeliness are important factors during this cycle and are essential for the success and cost effective execution of each shipment.

Thoroughness and accuracy in preparation of the Appendix A is important for several reasons. First, the technical information provided, including drawings and other reference material, is used by the INL as the basis for safety and operational reviews to ensure safe receipt and storage of the fuel in the existing dry storage facility. Secondly, this fuel data provides the basis for the cask vendor to verify and/or modify the cask license certificate for the transport of a particular fuel. Inaccurate data may delay the cask certification process, with the potential for adverse schedule impacts. Finally, thorough and accurate Appendix A data ensures that any fuels will be properly characterized for ultimate disposition in a future permanent repository.

Timely submittal of the Appendix A document is also very important. Ideally, the final Appendix A should be approved at least 6 months prior to scheduled fuel loading. Historical trends indicate that about 3 to 5 months are required for the initial INL review of the Appendix A fuel data, and involves critical site resources and communications with the research reactor operators. The initial document should therefore be submitted approximately one year in advance of the scheduled fuel loading. Early finalization of the Appendix A will allow ample time for the INL to complete its safety bases and operational reviews and implement any new facility modifications, process changes, or special training of fuel handling personnel that may be required to safely receive, unload, and store the fuel. If cask license reviews and revisions are required for transport of a particular fuel, additional time may be required. Cask vendors, foreign government competent authority representatives, and the U. S. Nuclear Regulatory Commission (NRC) have taken the position that cask license reviews may not begin until the Appendix A document has been finalized. Depending on the extent of the evaluations needed to review license submittals, the U. S. NRC and Department of Transportation approval process could range from 8 weeks to 12 months. Therefore, late submittals of Appendix A's have the potential to result in significant delays or cancellation of shipments because of licensing issues.

Inspection/Assessment Visits

A team of DOE and INL representatives may visit the reactor facility for those facilities that are preparing a shipment. These visits are scheduled to occur 12 to 18 months in advance of the intended INL receipt date in order to initiate the exchange of technical information and to identify and resolve early concerns. Contracts between DOE-ID and the reactor facility are finalized, clear understanding of all INL receipt requirements is ensured, and preliminary fuel shipment logistics are identified during these visits. If necessary, the INL will also inspect the fuel at this time for structural integrity, evidence of corrosion, ease of handling, fuel cropping or canning needs, and any other indicators that could possible affect receipt, handling and storage at the INL dry storage facility. The facility assessments also cover a review of the radiological and/or industrial work activities to help ensure a safe work environment. The visits provide an excellent opportunity for the reactor operator and the INL representatives to discuss the Appendix A Fuel and Packaging RSD forms and review the reactor operating history to support the timely resolution of issues.

Cask and Fuel Handling Assessment

Once the cask to be used for the shipment is chosen, the INL will initiate an independent review of the various documents that describe the cask, its licensed contents, and its handling. The cask's physical dimensions and handling methods are reviewed against the capabilities of the INL receipt facility. Areas of concern are either resolved by modifying the INL equipment, or are brought to the attention of the research reactor and cask vendor for mutual discussion and resolution. The fuel data compiled in the Appendix A document is compared to the licensed contents specified within the cask Certificate of Compliance to determine if any license revision is required. Ongoing communications provide the feedback mechanism to discuss any potential discrepancies. The cask vendor is also very much involved in working through any problems.

The fuel handling assessment includes: the internal cask "basket" or "shipping can", which will contain the fuel within the cask, the cask and basket loading configuration, and any specific cask or fuel handling tools that will be used during cask unloading and storage activities at the INL. Often, assistance from the reactor facility is needed to properly determine the correct handling tools. Any equipment that will be used for handling or storage at the INL will also require design and fabrication reviews by INL quality assurance personnel to ensure safe handling of the fuel within the INL receipt facility.

2.1.2. Pre-shipment INL facility activities

Once the Appendix A fuel data is finalized, fuel inspections are complete, and the cask/fuel loading and shipping configurations have been determined, the information is passed on to the INL receipt facility safety analysis and operations staffs. The facility safety personnel perform the necessary evaluations to ensure that the fuel can be received, unloaded and stored without the possibility of a criticality incident or an "un-reviewed safety question". The operations teams ensure all fuel handling facilities, procedures and training have been adapted to the specific fuel receipt and that the fuel storage location has been properly designated.

Upon completion of the INL criticality and facility safety analysis evaluations, (which are conducted in parallel) the facility will identify that it is ready to receive and store the shipment of TRIGA spent nuclear fuel and that:

- The facility criticality and safety bases will not be compromised;

- Cask handling issues, including any facility modifications, have been resolved and implemented.

- If a specific cask does not provide them, a set of spare tools is staged to minimize delays in the unloading if the SNF; and,

- All receipt, unloading, and storing procedures are implemented, and all facility operators and supervisors are trained on these procedures.

2.2. Authorization to ship

All of the information collected and reviewed during the fuel characterization phases, and the subsequent INL pre-shipment activities provides the technical basis for DOE-ID to provide to the research reactor facility an "Authorization to Ship" letter, allowing the shipment process (loading and transporting) to commence. The INL process to achieve this technical justification was established to ensure the safe and cost effective receipt and storage of spent nuclear fuel.

3. Conclusion

This receipt preparation process provides a good foundation for the success of a shipment, for both the research reactor facility and for the DOE Spent Nuclear Fuel Acceptance Program. The preparations for the safe and cost effective shipment of spent nuclear fuel to the INL start well in advance of the actual receipt date. Much effort at the INL is spent executing the technical and operational reviews, analyses, and evaluations that support the INL receipt process. For these reasons it is important to maintain a disciplined approach and schedule to ensure all pre-shipment preparation activities are initiated and completed in a timely manner, with accurate data.

REFERENCES

[1] Record of Decision, U. S. Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement (EIS), May 30, 1995.

[2] Amended Record of Decision, U. S. Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement (EIS), February, 1996

[3] Record of Decision, U. S. Department of Energy Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel Environmental Impact Statement (EIS), May 13, 1996.

[4] Amended Record of Decision, U. S. Department of Energy Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel Environmental Impact Statement (EIS), May, 2006.

[5] THOMAS, J.E., BICKLEY D.W., CONASTER, E. R., "Pre-shipment Preparations at the Savannah River Site WSRC's Technical Basis to Support DOE's Approval to Ship", Proceedings of the 2000 International Meeting on Reduced Enrichment for Research and Test Reactors, ANL/TD/TM01-12, July 2001.

[6] BECHTEL BWXT IDAHO (BBWI), Standard for Receipt of Spent Nuclear Fuel, STD-1120, Revision 0, August 31, 2001.

Appendix A.
Fuel and Packaging RSD Form

Revision________________

Description	Shipper Summary **NOTE:** If items do not apply, mark as N/A.	References (list reference number from Form 434.28B)
I. **SHIPPER AND SHIPMENT IDENTIFICATION INFORMATION**		
1. Shipper Information		
a. Shipper Reporting Identification Symbol (RIS):		N/A
b. Shipping Facility Name:		N/A
c. Shipping Facility Address:		N/A
d. Shipping Facility Point of Contact (name of authorized person for shipper):		N/A
e. Shipping Facility Point of Contact Telephone Number:		N/A
2. License number (foreign receipts only):	☐ N/A	N/A
3. Transfer authority – contract, NM draft or order number:		N/A
4. U.S. port of entry (foreign receipts only):	☐ N/A	N/A
5. IAEA batch identification number (International Atomic Energy Agency [IAEA] protocol or signatory facilities or domestic facilities subject to IAEA inspection and inventory requirements only):	☐ N/A	N/A
6. Ownership of accountable nuclear material (foreign receipts only):		N/A
7. Shipping Agent Information (entity ensuring fuel arrives at the INL)		
a. Shipping agent name:		N/A
b. Shipping agent address:		N/A
c. Name of authorized person for shipping agent:		N/A

Description	Shipper Summary **NOTE:** If items do not apply, mark as N/A.	References (list reference number from Form 434.28B)
II. **REACTOR DATA**		
1. Reactor Name:		N/A
2. Core identification or designation (e.g., MK-I, MK-II, Seed I, Prototype D1G-1A, etc.):		N/A
3. Initial criticality date for the reactor:	☐ N/A	
4. Shutdown date for the reactor:	☐ N/A	
5. Typical number of elements or assemblies in the core (or provide range):		
6. Core rated power:		
7. Significant events in its operating history (including core changeouts, configuration changes etc.):		
III. **SPENT NUCLEAR FUEL DATA**	**NOTE:** Data should be consistent with information provided on individual fuel units on Form 434.28A, "Fuel Unit RSD." **NOTE:** Packaging data (e.g., cans, baskets, buckets) is covered separately in Section IV.	
1. Fuel Name:	☐ TRIGA ☐ Buffalo Pulstar ☐ Other (provide name)	N/A
2. Total number fuel units (rod, plate, assembly, etc.) to be shipped broken down by fuel type and cladding:		N/A

Description	Shipper Summary **NOTE:** If items do not apply, mark as N/A.	References (list reference number from Form 434.28B)
3. Description of how and where the individual fuel units are identified (i.e., state if serial numbers are engraved, stamped, etched etc., provide size of serial numbers, and provide location of serial numbers on fuel units):		
4. Description of fissile material for the as-shipped fuel unit at beginning-of-life (BOL) over the length of the fuel unit, Include information for any fuel units that have been modified. **NOTE:** This information is used to support criticality safety evaluations at the receiving facility.		
5. Provide the general fuel description for the fuel units to be shipped. Include applicable fuel description reports and drawing numbers, with the revisions, and dates. If providing drawing equivalent information it needs to be sufficient for analysis, modeling, fuel handling, and should include a list of materials. The following information shall be obtainable from the drawings or provide for items 5a through 5i below:	Fuel description report: ☐ TRIGA Fuel Summary Report, ICP/INL-05-00817, R. Smith ☐ Buffalo Pulstar Fuel Summary Report, INEEL/INT-97-01050 ☐ Other (provide report name) Drawing numbers, revisions, and dates: Other information:	
a. Physical description of each type of fuel unit:	☐ Information on drawings	
b. Total length:	☐ Information on drawings	
c. Length of fueled portion:	☐ Information on drawings	
d. Position of fueled portion with respect to a permanent reference point on the fuel unit:	☐ Information on drawings	
e. Cross-sectional dimensions:	☐ Information on drawings	
f. Shape (plates, rod, etc.):	☐ Information on drawings	

Description	Shipper Summary **NOTE:** If items do not apply, mark as N/A.	References (list reference number from Form 434.28B)
g. Plenum spacers or springs:	☐ Information on drawings	
h Fuel particle size and composition:	☐ Information on drawings	
i Fuel matrix composition:	☐ Information on drawings	
6. Provide the general description for changes to the fuel unit (e.g., cuts made to prepare for shipping, disassembly). Include applicable reports and drawing numbers, with the revisions, and dates. If providing drawing equivalent information it needs to be sufficient for analysis, modeling, fuel handling, and should include a list of materials. Provide the as shipped dimensions including the following:	Drawing numbers, revisions, and dates: Other information:	
a. Cut length:		
b. Total length:		
c. Cross sectional dimensions:		
7. Composition of material in the fuel unit (as shipped)	**Note:** Composition of packaging (e.g., cans, baskets, buckets) is covered separately in section IV.	
a. Total weight of fuel unit:		
b. Fuel (chemical form of uranium and plutonium):		
c. Alloy or diluent in the matrix:		
d. Cladding:		
e. Any external coatings applied to the cladding:		
f. Thermal transfer material (e.g., sodium):		
g. Organic materials:		

Description	Shipper Summary **NOTE:** If items do not apply, mark as N/A.	References (list reference number from Form 434.28B)
h. Special additives:		
i. Chemically reactive materials (e.g., sodium):		
j. Neutron poisons, fixed or burnable:		
k. Total water (free and chemically or physically bound):		
l. Other (specify):		
8. Fuel unit condition and cladding integrity		
a. State how the fuel condition and cladding integrity was determined. Reference an inspection report if available.		
b. Describe any fuel cladding degradation (including blisters, pinholes, cracks, tears, creases, dents, deep scratches, corrosion deposits, crud etc.), including type and extent of damage, where and how it occurred.		
c. Describe any fuel unit degradation (including elongation, bowing, bulging, mechanical damage) including where and how it occurred.		
d. Is there any reason to suspect the fuel unit/cladding has been damaged to the extent that it would no longer retain fission products? If so, state reasons.		
e. Reactor and storage history. Note only conditions that might affect storage or future fuel unit/cladding integrity.		
f. Describe any fuel unit exposure to chemical contaminants (Hg, halides, etc.)		
i. Was the contamination removed?		

Description	Shipper Summary **NOTE:** If items do not apply, mark as N/A.	References (list reference number from Form 434.28B)
ii. How was the contamination removed?		
g. Describe measured or projected external radioactive contamination to the fuel unit (particularly alpha contamination levels).		
i. Was the contamination removed?		
ii. How was the contamination removed?		
h. If the fuel unit(s) have been modified or damaged, describe how it was determined there was not a loss of material. If a loss of material was determined to exist, describe what measurement method was used to calculate the end of life shipper's values reported as required on Form 434.28A Fuel Unit RSD.		
IV. SPENT NUCLEAR FUEL PACKAGING DATA	**Note:** This section applies only to SNF packaging (e.g., can, basket, bucket). Fuel unit data is covered separately in section III. Transportation package (shipping cask) is covered separately in section V.	
1. Specify the fuel handling unit (FHU) (e.g., can, basket, bucket) to be shipped. If there is more than one type of FHU, complete the sections to follow for each type of containment (e.g if elements are in a can, and the can is in a bucket/basket, complete the items for both the can and bucket/basket).		
2. Describe how and where the FHU(s) (e.g. can, basket, bucket) are identified (i.e., state if serial numbers are engraved, stamped, etched etc., provide size of serial numbers, and provide location of serial numbers on the FHU(s):		
NOTE: Subsections 3 through 6 do not need to be completed for cans (cans, tubes, containers, capsules, etc.). Information on cans is provided in subsections 8 through 16.		

Description	Shipper Summary **NOTE:** If items do not apply, mark as N/A.	References (list reference number from Form 434.28B)
3. Provide detailed drawings for each type of SNF packaging (fuel storage/handling devices) and additional components (e.g., plugs, caps, dividers, spacers). Include applicable reports and drawing numbers, with the revisions, and dates. Describe any deviations from the drawings. If drawings are not available, provide drawing equivalent information. Information needs to be sufficient for analysis, modeling, fuel handling, and should include a list of materials. The following information shall be obtainable from the drawings or provided below for subsections 3a through 3c.	Drawing numbers, revisions, and dates: Other information:	
a. Physical description of each type of SNF packaging:	☐ Information on drawings	
b. Materials of construction:	☐ Information on drawings	
c. Dimensions:		
i. Total length:	☐ Information on drawings	
ii. Cross-section dimensions:	☐ Information on drawings	
d. Weight:		
i. Empty:		
ii. Maximum loaded:		

Description	Shipper Summary **NOTE:** If items do not apply, mark as N/A.	References (list reference number from Form 434.28B)
e. Reference the INL/ICP approval for design of SNF packaging (fuel storage/handling devices) including any additional components (e.g., plugs, caps, dividers, spacers) provided by the shipper. **Note:** This approval is required for shipper supplied packaging and components only. Include only the components that will remain with the fuel storage/handling device (e.g., bucket, spacer in a bucket, lid on bucket, etc.) at the INL/ICP.		
4. List the extraneous material and its mass in grams associated with the SNF packaging or its contents:		
a. Pyrophoric or reactive material		
b. Inert materials		
c. Organic materials		
d. Total water (free and chemically or physically bound)		
e. Other (specify)		
5. List any external coatings or special additives associated with the SNF packaging (e.g., paint, ink):		
6. Describe any SNF packaging exposure to chemical contaminants (Hg, halides, etc.).		
a. Was the contamination removed?		
b. How was the contamination removed?		

Description	Shipper Summary **NOTE:** If items do not apply, mark as N/A.	References (list reference number from Form 434.28B)
7. Describe measured or projected external radioactive contamination to the SNF packaging. (Particularly alpha contamination levels.).		
a. Was the contamination removed?		
b. How was the contamination removed?		
NOTE: Subsections 8 through 16 are applicable to cans (cans, tubes, containers, capsules, etc.) only. If canned fuel is stored in an overpack can (e.g., can in a can, or sealed can, and non-sealed can or screened can), this section must be completed separately for each type of can.		
8. Provide detailed drawings for each type of can. Include applicable reports and drawing numbers, with the revisions, and dates. Describe any deviations from the drawings. If drawings are not available, provide drawing equivalent information. Information needs to be sufficient for analysis, modeling, fuel handling, and should include a list of materials. The following information shall be obtainable from the drawings or provided below for subsections 8a through 8c.	Drawing numbers, revisions, and dates: Other information:	
a. Physical description of the can:	☐ Information on drawings	
b. Materials of construction:	☐ Information on drawings	
c. Dimensions:		
i. Total length:	☐ Information on drawings	
ii. Cross-section dimensions:	☐ Information on drawings	
d. Weight:		
i. Empty:		

Description	Shipper Summary **NOTE:** If items do not apply, mark as N/A.	References (list reference number from Form 434.28B)
ii. Maximum loaded:		
e. Reference the INL/ICP approval for design of cans provided by the shipper. **Note:** This approval is required for <u>shipper</u> supplied cans only.		
9. List the extraneous material and its mass in grams associated with the can or its contents:		
a. Pyrophoric or reactive materials:		
b. Inert materials:		
c. Organic materials:		
d. Total water (free and chemically or physically bound):		
e. Other (specify):		
10. List the extraneous material and its mass in grams associated with the can or its contents:		
11. Describe any SNF packaging exposure to chemical contaminants (Hg, halides, etc.).		
a. Was the contamination removed?		
b. How was the contamination removed?		
12. State the purpose and function of the can, tubes, containers, capsules etc		

Description	Shipper Summary **NOTE:** If items do not apply, mark as N/A.	References (list reference number from Form 434.28B)
a. Identify the purpose of all penetrations		
b. Is the can(s) sealed or vented? If it is sealed what is the maximum allowable pressure in a sealed can?		
c. Gas generation rates within the can		
13. State the conditions under which the contents is canned (e.g., canned wet, canned dry, etc.)		
14. If the can was subject to a wet environment and will be stored dry, describe the drying process to include the following:		
a. Drying process		
b. Dryness criteria		
c. Dryness criteria verification compliance method		
15. List the can back fill gas and pressure		
16. If material will reside in an original can and will not be transferred to a new can, describe the method used to verify the contents of the can and provide the data. The verification method shall be onsite inspection or approved alternate.		
V. TRANSPORTATION PACKAGE (SHIPPING CASK) DATA		
1. Identify the transportation package (cask), to be used for this shipment.	☐ NAC-LWT Cask ☐ GE-2000 Cask ☐ Other (provide name)	N/A

Description	Shipper Summary **NOTE:** If items do not apply, mark as N/A.	References (list reference number from Form 434.28B)
2. Provide the current Department of Transportation (DOT) Certificate of Competent Authority or the DOE/NRC Certificate of Compliance (C of C), or equivalent, as applicable.	Document number, revision, date:	
3. Provide the Safety Analysis Report For Packaging (SARP) or equivalent, including the Shipping Package Transport Plan if applicable.	Document number, revision, date:	
4. If other than the NAC-LWT cask or GE-2000 cask, provide detailed drawings of the transportation package (shipping cask), if requested. Include applicable reports and drawing numbers, with the revisions, and dates. If drawings are not available, provide drawing equivalent information. Information needs to be sufficient for analysis, modeling, fuel handling, and should include a list of materials. Describe any deviations from the drawings. The following information shall be obtainable from the drawings or provided below for items 4a through 4e.	☐ N/A (for NAC-LWT and GE-2000 Casks) Drawing numbers, revisions, and dates: Other information:	
a. Dimensions	☐ Information on drawings	
b. Weight (gross and net)	☐ Information on drawings	
c. Surface finish (e.g., roughness, painting, coating, etc.)	☐ Information on drawings	
d. Materials of construction	☐ Information on drawings	
e. Size and description of lid bolts	☐ Information on drawings	

Description	Shipper Summary **NOTE:** If items do not apply, mark as N/A.	References (list reference number from Form 434.28B)
5. Describe the poison, poison inserts, spacing insert, or dunnage necessary for shipping (including weights). Furnish certification of integrity. Provide drawings and/or equivalent documentation including weights.		
VI HOISTING AND RIGGING FIXTURES	**NOTE:** This section is applicable to rigging, slings, spreader bars, tools, yokes etc. This section is only applicable for hoisting and rigging fixtures designed by the shipper to be used at the INL/ICP.	
1. Provide detailed drawings for each type of hoisting and rigging. Include applicable reports and drawing numbers, with the revisions, and dates. Describe any deviations from the drawings. If drawings are not available, provide drawing equivalent information. Information needs to be sufficient for analysis, modeling, fuel handling, and should include a list of materials. The following information shall be obtainable from the drawings or equivalent information provided below for items 1a through 1c.	Drawing numbers, revisions, and dates: Other information:	
a. Dimensions:		
i. Total length		
ii. Cross-sectional dimensions		
b. Weight:		
c. Type of load bar (if applicable):		

Description	Shipper Summary **NOTE:** If items do not apply, mark as N/A.	References (list reference number from Form 434.28B)
2. Describe the use of each fixture. If any special tools are required, describe them in detail and provide drawings. Include precautions for use of the fixture and tools.		
3. Reference the INL/ICP approval for design of hoisting and rigging equipment provided by the shipper. **Note:** This approval is required for <u>shipper</u> supplied hoisting and rigging equipment for use at the INL/ICP only.		
VII. ENVIRONMENTAL SAFETY AND HEALTH		
1. Provide any documents covering criticality safety evaluations and calculations that determine the minimum critical number of pieces of the subject fuel and that evaluate the criticality safety of shipping, handling, and storing the fuel.	Document number, revision, date:	
2. Provide a copy of all existing NEPA documents for the storage and handling of this fuel. If none, state none.	Document number, revision, date:	
VIII. QUALITY ASSURANCE		
1. Reference attached documentation that the shipper's QA program has been approved by the INL/ICP. If approval of QA program is not necessary state why.		
IX. OTHERS		
1. Provide any information not provided above that may have an impact on the INL/ICP's SNF receipt and storage or operations.		

Description	Shipper Summary **NOTE:** If items do not apply, mark as N/A.	References (list reference number from Form 434.28B)

Prepared By:

Printed Name/Title	Signature	Date

Approved By:

(Shipper Management) Printed Name/Title	Signature	Date

The Y-12 National Security Complex's efforts supporting the foreign research reactor spent nuclear fuel programme

T. Andes

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Abstract. The transport of 3.7 kilograms of highly enriched uranium (HEU) from Argentina to the Y-12 National Security Complex (Y-12) was completed in July 2006. This project was a collaborative effort between the Comision Nacional Energia Atomica (CNEA) and Y-12 and supported the goals of the National Nuclear Security Administration's (NNSA) Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Return Program under the aegis of the Global Threat Reduction Initiative. The HEU, in the form of material test reactor (MTR) plates, originated from the shutdown RA-2 critical assembly. This paper describes both the programmatic and technical activities required to complete the mission. The programmatic activities include contractual requirements between the NNSA and CNEA, National Environmental Policy Act documentation, and safeguards agreements. The technical activities include inspection results, material packaging, container loading, and transportation.

1. Introduction

United States (U.S.)-origin HEU was acquired by Argentina in the 1960s in conjunction with the United States Atoms for Peace program. CNEA utilized the HEU to fabricate research reactor nuclear fuel elements for research reactors RA-3 and RA-6, as well as the RA-2 critical assembly. In July 2006, NNSA, CNEA, and Y-12 concluded a project that resulted in the shipment of 3.7 kg HEU from CNEA's Centro Atomico Constituyentes (CAC) complex to the Y-12 National Security Complex in Oak Ridge, Tennessee, USA. The HEU was in the form of 438 fresh and slightly irradiated research reactor plates (uranium-aluminum alloy core clad in aluminum) equivalent to 24 MTR fuel elements. The shipment was made by commercial air transport and was a culmination of over one and one-half year's effort.

2. Facility description

The RA-2 HEU was stored in fresh fuel storage facilities at the CAC complex located in the San Martin province of Buenos Aires. CAC is one of several CNEA sites which include Centro Atomico Ezeiza, located just outside Buenos Aires is the site of the RA-3 research reactor; and Centro Atomico Bariloche, located in southwest Argentina is the site of the RA-6 research reactor. The CAC consists of a number of facilities including those supporting fuel fabrication, the RA-1 research reactor (LEU fueled), an accelerator, laboratories, and other support facilities.

3. RA-2 Fuel Description

The RA-2 fuel plates and elements are standard MTR design and identical to those used in the RA-3 or RA-6 research reactors. The fuel elements were fabricated by CNEA. Figure 1 is a schematic of an intact standard fuel element. Details of the RA-2 fuel plates and elements are provided below:

Element Configuration:	Box-type assembly with 19 fuel plates for the standard types and 15 plates for the control type, lifting bail at the top and flow nozzle at the bottom

Element Dimensions: 7.62 cm × 8.4 cm cross-section × 88.0 cm in length (92.5 cm for the control assembly)

Plate Dimensions: 0.13 cm thick × 6.8 cm (curved) × 65.5 cm in length for interior plates and 75.5 cm in length for exterior plates

Active Length: 61.5 cm

Clad Thickness: 0.039 cm

Structural Materials: Aluminum side plates and end fittings

Fuel and Form: Uranium (90% ^{235}U) alloyed with aluminum and clad in 1100 grade aluminum

Fuel Loading: Approximately 7.8 g ^{235}U per plate and 148 g ^{235}U per standard element and 117 g ^{235}U per control element

Reactor Background: The RA-2 is a light water moderated and cooled critical assembly that achieved criticality in July 1966 and operated at a power of 0.03 kw (steady state thermal). The critical assembly and the fuel elements were designed and constructed by CNEA. The RA-2 operated until 1983 when it was shut down.

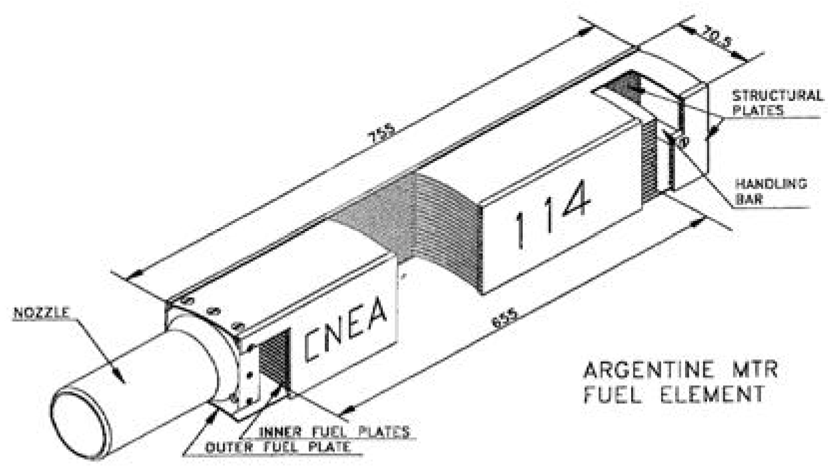

FIG. 1. Typical RA-2 intact fuel element.

4. Technical requirements

4.1. Inspection

An assessment team consisting of NNSA and Y-12 technical staff visited the CAC in November 2004 to inspect, characterize, and gather pertinent data on the RA-2 fuels plates and elements necessary to determine disposition pathways. The inspection and characterization process included measurements of gross weights, radiological doses, and enrichment (confirmatory) (Figs 2, 3, and 5). Weights were obtained using a Mettler Toledo scale (Fig. 4). An EFC NaI (Sodium-Iodine) Detector with MCA-166 spectrum analyzer and NaI GEM Software Version 1.5 were used for nondestructive assay. A standard FAG-type dose rate meter was used for measurements at contact and 30 cm (1 foot).

The inventory included 95 loose plates (plus 2 mini-plates) and 19 intact elements (14 standard and 5 control). The HEU inventory totaled approximately 3 kg of ^{235}U. The inspection results concluded that: (1) there was no damage that would require additional containment for shipping, (2) dose rate measurements were determined to meet Y-12 Acceptance Criteria for processing, and (3) the spectra acquired from the NaI detector confirmed that the plates and elements contained 90% enriched uranium. The spectra data was analyzed upon return to Y-12 with the NaI GEM calibrated with an enrichment standard and adjustments made for composition and thickness.

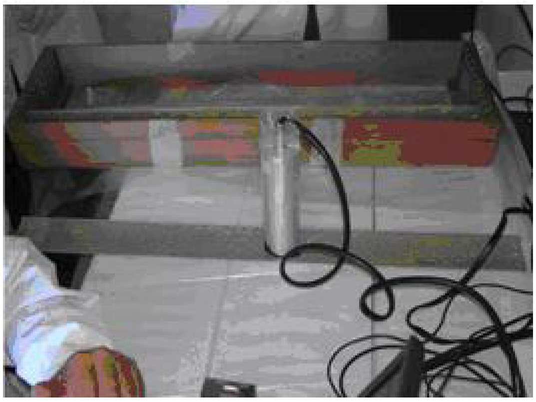

FIG. 2. Nondestructive analyses of plate.

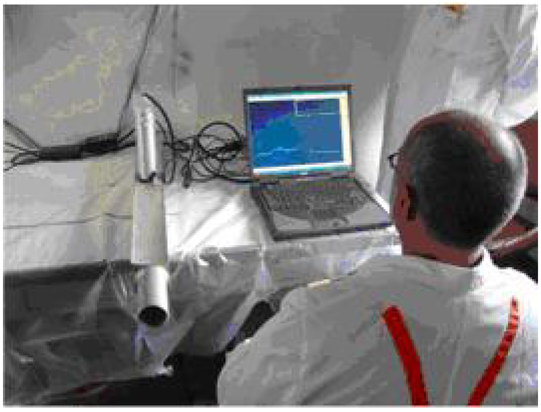

FIG. 3. NDA of intact assembly.

FIG. 4. Weight scale.

FIG. 5. Measuring dose rate of plate stack

Calculations using the SCALE5 collection of computer codes were used to determine the plutonium, fission product, and ^{236}U contents of the plates. The inputs for the codes were from the inspection data including dose rate measurements, plate dimensional and loading details, and assuming a conservative RA-2 operating history. The results indicated that the plates meet the definition of unirradiated as defined by 10CFR71 and 74 (compatible with IAEA TS-R-1).

In 2005, the CNEA staff dismantled the intact elements resulting in 341 additional plates. Dismantlement required removal of screws that attached the bail and flow nozzle, which allowed the side plates to be pulled in opposite directions to detach the fuel plates. The detachment was simpler than expected because only the exterior fuel plates were swaged to the side plates (this allowed variable fuel loadings of the critical assembly). The plates from the intact elements and the loose plates totaled 438 plates.

4.2. *Scrap declaration and package selection*

Once it was determined that Y-12 would be the receiving site for the RA-2 HEU, it was necessary to initiate technical documentation required to officially document acceptance of the material for receipt, storage, and disposition. That documentation consists of the Y-12 Scrap Declaration which describes the special nuclear material (SNM) as well as the shipping package that will be used to transport the SNM. The Scrap Declaration is used by Y-12 to perform assessments in areas of SNM accountability, nuclear criticality safety, radiological controls, facility operations, and disposition planning. The different sections of the Scrap Declaration include detailed descriptions of the shipping package and inner container (materials of construction, dimensions, weights, components, loading configuration, gross weights, and dose rates) and SNM (material form, dimensions, weights, identification numbers, dose rates, and contamination levels). The Scrap Declaration was initiated by the CNEA in 2005 with guidance from Y-12.

The shipping package utilized for the project was the CEA (Commissariat a L'Energie Atomique) TN-BGC1 that Y-12 had previously used in other enriched uranium transportation projects. The TN-BGC1 was selected because: (1) it was certified for material type of concern, (2) Y-12's familiarity with the package, and (3) it could be transported by air. The TN-BGC1 is a Type B package with a U.S. Department of Transportation Certificate of Competent Authority. Prior to use in Argentina, the Autoridad Regulatoria Nuclear (ARN) of Argentina issued a transportation certificate after evaluation and review of the TN-BGC1 Safety Analysis Report for Packaging (SARP).

The TN-BGC1 is generally described as an outer container (stainless steel pipe) surrounded by a protective rectangular box-shaped cage of aluminum tubing that is 600 mm square by 1 821 mm in height. The outer container can be sealed and leak tested. The inner container used for this project was the TN-90. The TN-90 is removable from the outer container and will hold the plates without any additional containment. The TN-90 has a screw-type lid with a top-end clamp that compresses an O-ring seal. The TN-BGC1 and TN-90 are shown in Fig. 6.

FIG. 6. TN-BGC1 and TN-90.

CNEA and Y-12 determined that a TN-BGC1 loading configuration that would most efficiently meet size limits and maintain nuclear safety was the banding of 25 plates in a stack with metal bands placed at strategic points around width and length of the stack to ensure integrity. The internal useable length of the TN-90 would allow for loading of two stacks, one on top of the other.

CNEA proceeded to complete the provisional Scrap Declaration with the plate stack configuration that resulted in 18 stacks loaded into 9 TN-BGC1 packages. CNEA staff also sorted the plates into their assigned stacks and obtained radiological data, including dose and contamination smears. The provisional Scrap Declaration was then provided to Y-12 for review and approval. The Scrap Declaration was completed just prior to shipment when the loaded packages were measured for external dose, weighed, identification numbers matched with contents, and security seals applied.

5. Programme requirements

Programmatic activities were performed in parallel to the technical activities described above to ensure the project met governmental requirements. These included contracts between the NNSA and CNEA, an environmental analysis, and determination of safeguards requirements.

The NNSA-CNEA contracts identified the precise inventory of RA-2 HEU that would be returned to the United States, as well as an equivalent quantity of LEU that would be provided to CNEA in exchange for the HEU. The LEU received by CNEA in the exchange would be used in the conversion of the RA-6 research reactor. A second contract, linked to the execution of the first, committed Argentina’s conversion of the RA-6 research reactor.

The environmental analysis, as required by the U.S. National Environmental Policy Act (NEPA), was issued as DOE/EA-1529, “Environmental Assessment for the Transportation of Unirradiated Uranium in Research Reactor Fuel from Argentina, Belgium, Japan, and the Republic of Korea to the Y-12 National Security Complex.” EA-1529 analyzed air transport over global commons, six ports of entry into the United States, commercial and non-commercial transport. The Department of Energy (DOE) determined that the activities described in the EA-1529 were not a major action significantly affecting the quality of the environment, and therefore issued a Finding of No Significant Impact. Existing Y-12 site specific environmental documentation was already in-place which permitted transportation activities within the U.S. borders and receipt of foreign source HEU.

The third activity was the safeguards requirements for the RA-2 material once it reached the United States. Argentina, which is a party to the ABACC (Agencia Brasileno-Argentina De Contabilidad Y Control De Materiales Nucleares) agreement (an agreement for accounting and control of nuclear material) required that when material such as the RA-2 HEU leaves Argentina for another country, it must be put under International Atomic Energy Agency (IAEA) safeguards. The only DOE facility capable of carrying out that role is the IAEA Vault at Y-12. Although the RA-2 plates would not be physically placed in the vault, it was agreed that an equivalent quantity of HEU would be substituted.

6. Project execution

Project execution includes activities performed that lead to the shipment of the material. Contracts were signed for package rental (CERCA) and transportation services (Edlow International) which placed 9 TN-BGC1 packages and support equipment in Argentina. Physically inside CAC, CNEA staff began banding of the 438 fuel plates into 18 stacks. During this process, the Y-12 observers recorded the plates’ unique identification numbers for accountability and CNEA attached banding around the width and length of the plates (Fig. 7). A loop of extra banding was included at one end of the stack to act as a lifting attachment during TN-90 loading and unloading. Indelible ink was applied to an exposed plate to record the identification number of the stack and list the stack’s gross, net, and tare weights (Fig. 8). Just prior to banding, approximately 45 longer (exterior) fuel plates (about 5 cm longer than internal plates) were sheared to the same length as the internal plates. The shearing removed only aluminum. This allowed the axial banding to fit snugly around all the plates in the stack.

FIG. 7. Attachment of banding.

FIG. 8. Stack identification.

Upon receipt of the final governmental permission (issuance of the Export License by the Argentine 603 Commission which included the ARN and Ministries of Defense and Economic Affairs), CNEA and Y-12 personnel proceeded to load, seal, and leak test the 9 TN-BGC1 packages. Figure 9 is a conceptual rendering of the RA-2 plates loaded in the TN-BGC1. Figure 10 shows the leak testing of the TN-BGC1. The loading operations were performed in a maintenance shop facility located in close proximity to a fresh fuel storage building. The shop included a large open bay area equipped with an overhead hoist, ladders, and catwalks that would serve to support TN-90 inner container handling and protective cover installation. .

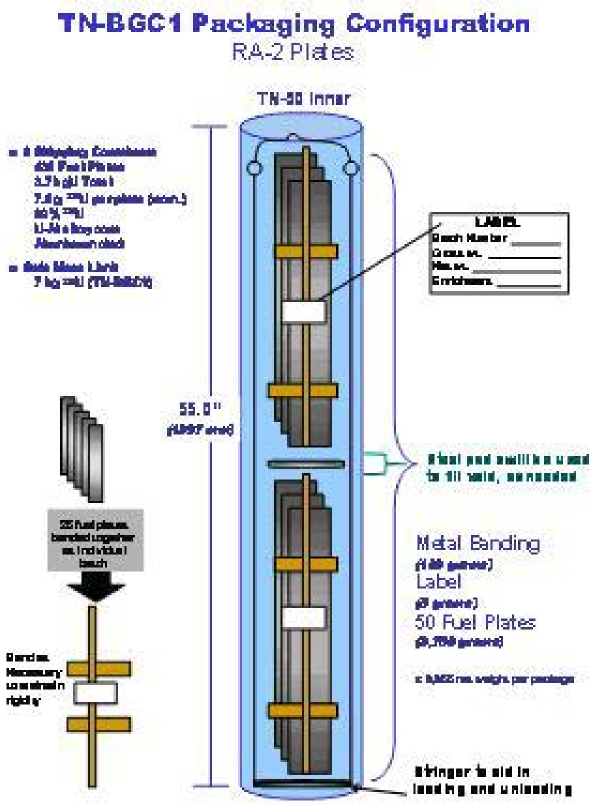

FIG. 9. TN-BGC1 loading arrangement.

Once the packages were successfully leak tested, the protective cover was attached and Y-12 personnel applied Tamper-Indicating Devices. As part of the ABACC agreement, inspectors from ABACC and the International Atomic Energy Agency observed loading operations and applied their seals to the TN-BGC1 packages. A pallet jack was used to move a loaded TN-BGC1 to a scale (fish-hook type suspended from the hoist) for weighing and then to an interim storage location within the shop. A loaded TN-BGC1 weighed about 340 kg.

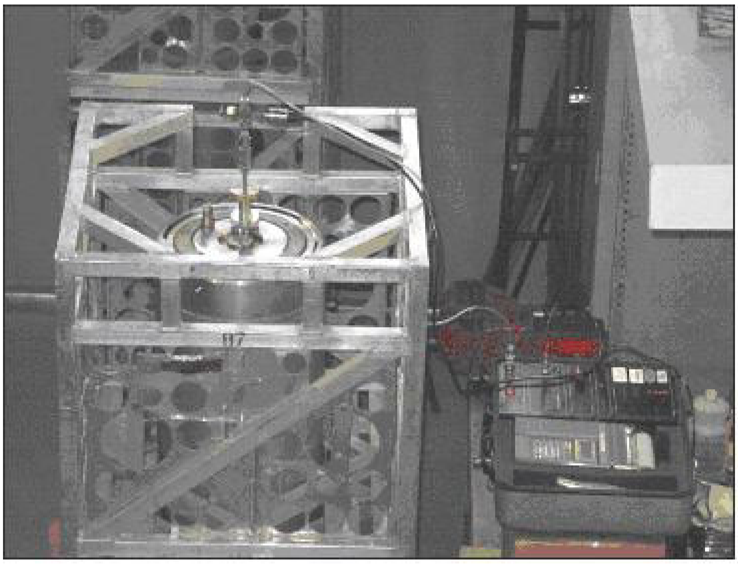

FIG. 10. TN-BGC1 leak testing.

The time required to complete the loading operations for the 9 TN-BGC1 packages was approximately a day and a half. Another day was required to finalize the Scrap Declaration and complete related shipping paperwork. CNEA then, working with Edlow International and their transportation team, proceeded to load the packages into transport conveyances for shipping to the port of exit (Fig. 11).

7. Summary

The project to repatriate HEU from Argentina in support of the FRRSNF Return Program was successfully completed in July 2006. The physical effort to transport the HEU from Argentina to the United States involved a number of organizations including CNEA, NNSA, Y-12, IAEA, ABACC, and Edlow. The team that performed tasks associated with loading operations consisted of approximately 6 CNEAstaff, 2 Y-12, 1 IAEA, and 1 ABACC. The success of the project is a clear demonstration of excellent collaboration between countries and organizations. This project was the first fresh fuel shipment for the FRRSNF Program to Y-12 and the experience gained will be very beneficial for future endeavors.

FIG. 11. . RA-2 HEU leaving CAC.

Approaches for expanding participation of developing countries in research reactor spent nuclear fuel return programme

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Edlow International Company, Washington, DC, United States of America

Abstract. The U.S. Department of Energy, with the support of the State Department, has changed policies in certain situations, which have resulted in enhanced participation by developing countries in the Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) program. This paper describes these changes, and offers suggestions for additional policy changes, which could further expand developing country participation in the program, with significant benefits for overall U.S. nuclear nonproliferation policy objectives.

1. Introduction

A willingness to develop creative and flexible approaches is essential to maximize the participation of developing countries in both the U.S. and Russian Research Reactor Spent Fuel Return programs. Over the past several years the U.S. Department of Energy (DOE) has demonstrated some flexibility in changing its policy in order to win developing country support for participation in the U.S. program. DOE's original position was not to help pay for the conversion of foreign research reactors from highly enriched uranium (HEU) to low enriched uranium (LEU). When it became clear that a number of developing country research reactor operators eligible to ship their partially irradiated fuel containing HEU in the reactor core to the United States would not do so because it would have resulted in the shutdown of their reactors, DOE changed course and decided to offer the operators compensation for the value of the remaining service life of the HEU in the core.

It is also useful to explore several early cases prior to the initiation of the U.S. Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) program in which reactor conversions took place without U.S. financial support. Most of these conversions were carried out in cooperation with Argonne National Laboratory (ANL), which started the Reduced Enrichment for Research and Test Reactors (RERTR) program in 1978.

2. Early cases of conversion to LEU

Taiwan

The earliest case was the THOR reactor in Taipei that began conversion from HEU MTR fuel to LEU Triga fuel in 1978. Conversion was completed in 1987 and was financed by the government of Taiwan.

The Philippines

Another early case was the Philippines Research Reactor (PRR-1), which was converted from 93 percent enriched uranium to 20 percent in 1987. The Philippines Government paid for the conversion with a lump sum grant in 1982. The PRR-1 was subsequently shut down and all of the spent fuel containing U.S. origin HEU was shipped to the United States in 1999.

Thailand

The TRR-1 reactor in Thailand was converted to LEU Triga fuel in 1977. After conversion, TRR-1 received support provided through an IAEA Technical Cooperation project which began in 1986 and was completed in 1992. The reactor is still operating. All of the spent fuel containing HEU has been shipped to the U.S.

Argentina and Brazil

The cases of the RA-3 research reactor in Argentina and the IEA-R1 reactor in Brazil have an interesting historical background. The operators of both reactors began working on research reactor conversion to LEU in the early 1980's. CNEA cooperated extensively with Argonne in developing LEU oxide dispersion fuel and on technical studies. After the LEU fuel was qualified, CNEA worked independently to convert the RA-3 reactor.

Brazil worked independently, and also cooperated with the IAEA and Argonne in manufacturing and qualifying LEU fuel for convesion of the IEA-R1 reactor.

In the 1980's, Argentina and Brazil had not adopted fullscope IAEA safeguards and had not joined the Nuclear Non-Proliferation Treaty or the Treaty of Tlateloco. As a consequence, neither country could obtain nuclear fuel exports from the United States of America. Moreover, after the Nuclear Suppliers Group adopted fullscope safeguards as a condition of supply in 1992, they would not have been able to obtain nuclear fuel from European suppliers. Nonetheless, Argentina and Brazil pressed ahead in developing and qualifying LEU fuels for reactor conversions.

The IEA R1 (Brazil) began conversion to LEU fuel in 1981. Conversion to LEU was completed in 1997, and Brazil paid for the conversion. The reactor was fueled with LEU supplied by the U.S. after Brazil adopted full-scope safeguards by concluding the Quadrapartite Safeguards Agreement with Argentina, the Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials (ABACC) and the IAEA [1], and had taken steps to complete the requirements for adherence to the Treaty of Tlateloco. The date of entry into force for Brazil was 1994.

The RA-3 (Argentina) was converted in 1990 using fuel elements manufactured in Argentina with LEU of Russian origin., CNEA paid for both the LEU maerial and for the manufacture of the fuel elements. This purchase was facilitated by the collapse of the Soviet Union. Before 1990, the USSR had not been known to supply research reactor fuel to countries outside the old Soviet Bloc except for a few countries in its sphere of influence, such as North Korea, Vietnam, and Libya. Also, the Nuclear Suppliers Group had not yet adopted fullscope safeguards as a condition of supply, so that Russia could supply the LEU.

The stories of the IEA R1 and the RA-3 had "happy endings" from the standpoint of nonproliferation policy. However, their reactor operators back in the 1980's did not have many options. They either had to work on their own or in cooperation with Argonne, and they presumably regarded willingness to convert to LEU as their best hope for continuing to operate their reactors. It seems unlikely that these examples will be duplicated elsewhere.

Colombia

The IAN-R1 reactor in Colombia also was converted to LEU fuel without U.S. financial assistance. However, the United States of America had considerable leverage with Colombia with respect to reactor conversion. The physical security at the reactor site, which was in downtown Bogota, did not meet IAEA recommended guidelines. The United States of America made it clear that they would not approve export licenses for replacement fuel or components for the IAN-R1 until Colombia agreed to ship back to the United States of America all of the irradiated fuel containing HEU as well as the fresh HEU fuel. After the shipment of this fuel to the United States of America in 1996, the U.S. granted export licenses for LEU fuel for the reactor. The reactor converted to LEU, was shut down temporarily, and is now back in operation..

Chile

Conversion of the La Reina research reactor was completed on May 10, 2006. CCHEN (Comision Chilena de Energia Nuclear), which operates the reactor, received extensive technical and financial assistance through IAEA Technical Cooperation projects for developing the fuel fabrication facility which supplies fuel for La Reina and for irradiation qualification for the fuel, which was done in cooperation the Nuclear Research and Consultancy Group (NRG) at the Petten reactor in The Netherlands. CCHEN paid for the LEU, which was purchased from Russia. However, the major expense was for the fuel fabrication facility, which was supported by the IAEA.

Although the La Reina conversion has been completed and the spent fuel containing U.S. origin HEU has been shipped to the United States of America under FRRSNF, there is still unfinished business in Chile in that there is still HEU in the country, ans issue that is further discussed later in section 5 of this paper.

3. Recent conversion cases resulting from DOE policy change

As noted previously, the rather unique circumstances surrounding the earlier reactor conversions discussed above do not apply elsewhere. After the Record of Decision was issued for the FRRSNF program in 1996, DOE from time to time approached reactor operators in “other than high income countries” [2] eligible for shipping spent fuel to the U.S. with most of the expenses paid by DOE, trying to interest them in shipping their U.S. origin spent fuel to the United States of America. However, it became abundantly clear that few if any of them — particularly those reactor operators who intended to continue operating their reactors — were interested in shipping their unirradiated HEU to the United States of America because the net effect would be that they would not be able to continue operating their reactors. They could not afford to pay for conversion.

DOE apparently has changed course and either has provided or may provide financial assistance for conversion in a number of cases. Included on this list are Mexico (Salazar), Portugal (RPI), Jamaica (Slowpoke), and Romania (SSR). There is a U.S. plan to convert Salazar but DOE had not concluded an agreement with the Mexican authorities. In addition, the GRR-1 reactor in Greece, in the context of a large amount of assistance from the U.S. for security at the 2004 Olympics, shipped all of their HEU to the US. The GRR-1 now operates with a LEU core.

4. Other opportunities for removal of HEU from research reactors

The Canadian-designed and built Slowpoke reactors and the Chinese-built Miniature Neutron Source Reactors (MNSR, 27 kw) are small reactors of similar design. Most of them have cores consisting of less than one kilogram of HEU (90 percent enrichment or greater).

Two of the Slowpokes in Canada operate on LEU — one was converted and the other was initially built with a LEU core. The Slowpoke in Jamaica is being converted to LEU, and as noted previously, the conversion will likely be paid for by DOE and the spent fuel containing HEU will likely be shipped to the United States of America. However, so far no work is being done currently on conversion of the reactor.

There are MNSRs in Ghana, Iran, Nigeria, Pakistan, and Syria, which present an excellent opportunity for conversion to LEU. A technical study by Argonne on the feasibility of use of LEU for this type of reactor showed promising results [3].

The IAEA hosted a technical meeting on conversion of Slowpoke and MNSRs in May 2005, which was attended by representatives from Canada, China, France, Ghana, Nigeria, Pakistan, Syria, and the U.S. The report of this meeting made a number of recommendations, including the creation of IAEA

Technical Cooperation projects that address conversion of Slowpoke and MNSRs, and a suggestion that donors provide extra-budgetary assistance for these projects [4]. The IAEA has set up a Cooperative Research Project involving China, the U.S., and most of the MNSRs. The first meeting is currently planned for December 2006. China has recently informed the IAEA of their willingness to accept for disposition the spent fuel from the MNSRs that were purchased with IAEA assistance.

5. Cases that may require comprehensive solutions

There are a number of cases that are relatively complex in that other issues beyond the straight-forward return of spent fuel under the FRRSNF program need to be addressed. It would be appropriate at this point to identify what these other issues are. These cases are probably best resolved through a comprehensive approach that addresses all of the issues, rather than trying to deal with them on a piecemeal basis.

A considerable amount of diplomatic effort will be needed to handle these cases, but DOE, working closely with the U.S. Department of State, has succeeded in resolving equally challenging situations in the past.

An example of past creative problem-solving was the decision to accept "Urgent Relief" spent fuel shipments in 1994 and 1995, which DOE, in cooperation with the State Department, undertook to address concerns about spent fuel at six reactors which had run out of storage space and had to be dealt with prior to the completion of the Environmental Impact Statement for FRRSNF and the announcement of the Record of Decision in 1996 [5]. A total of 252 fuel elements came to the U.S. under the "urgent relief" shipments.

More recently, DOE, with U.S. State Department assistance, was able to resolve disposition issues regarding U.S. origin HEU at several locations involving material in different forms by working with Argentina using a multi-faceted approach. Argentina (CNEA) has a contract with DOE for shipment of the RA-6 spent fuel containing U.S. origin HEU and for conversion of RA-6 to LEU. Linked to this issue was the need to find a disposition path for HEU fuel plates which had been manufactued years ago for the RA-2 reactor, which was shut down in 1983.

The DOE Y-12 facility is the only U.S. facility that can receive the RA-2 fuel plates. Y-12 is not on the list of eligible facilities for the U.S. "Voluntary Offer" to the IAEA for the application of IAEA safeguards on source and special nuclear material in facilities of "no direct national security significance" in the U.S. [6]. Under the Voluntary Offer, the U.S. has provided the IAEA with a list of eligible facilities for the application of safeguards. However, under the Quadrapartite Safeguards Agreement, all Argentine and Brazilian nuclear material must be located at a facility that is under IAEA safeguards [7]. It took many months to resolve the matter but in the end the State Department worked out a substitution arrangement whereby IAEA safeguards are applied to U.S. HEU in a safeguarded facility. The safeguards are attached to an amount of U-235 that is equal to the amount of U-235 in the RA-2 fuel plates being shipped to Y-12.

DOE also agreed to compensate CNEA for the value of the HEU in the RA-2 fuel plates in the form of LEU to be sent to Argentina to fabricate fuel for the RA-6 conversion.

In addition, the DOE-CNEA contract also contains a commitment by CNEA to recover and blend down HEU contained in various forms of scrap material in storage in Argentina.

The various shipments were delayed for months because it took a long time to resolve the safeguards issue, thus adding significantly to the costs incurred by DOE and the private firms handling the various shipments. Nevertheless, the Argentina case is a good example of DOE being willing to in effect agree to a "package deal" with CNEA that covered all of the outstanding issues pertaining to U.S. origin HEU in Argentina.

South Africa

The most important case involving several issues that may best be resolved through a "package solution" at present is South Africa. There are five Global Threat Reduction Initiative (GTRI) issues that pertain to South Africa:

- return to the U.S. of the U.S. origin HEU contained in spent fuel;
- shipment to the U.S. of Republic of South Africa (RSA)- origin HEU contained in spent fuel;
- conversion of the Safari research reactor to LEU;
- commitment by the Nuclear Energy Corporation of South Africa (NECSA) to use LEU targets when the technology can be deployed by a significant industrial producer;
- disposition of a large quantity of fresh HEU of RSA origin.

Significant quantities of HEU are involved:

- 50 U.S. origin spent HEU assemblies that initially contained approximately 10 KG of 93% HEU;
- 470 RSA origin HEU assemblies containing approximately 188 KG of 93% HEU;
- at least 175 KG fresh HEU – 93 % assay;
- an unknown quantity of 45% RSA origin HEU used for Safari between the mid-1970's and about 1995.

There is also target material containing HEU (45%) of RSA origin. Depending on whether the uranium is rcycled after Mo-99 has been removed, there may be a considerable inventory of this material.

The RSA has announced its intention to convert the Safari reactor to LEU and is taking appropriate steps in this endeavor. However, South Africa appears reluctant to agree to ship the U.S. origin spent fuel without a contractual commitment to remove the RSA research reactor spent fuel because it is afraid of being left without a disposition option for the RSA spent fuel at the end of the FRRSNF program in 2016. The RSA spent fuel and fresh HEU fuel was derived from South Africa's nuclear weapons program, which was officially abandoned in 1990, followed by adherence to the NPT and conclusion of a fullscope safeguards agreement with the IAEA in 1991. Some of the fresh fuel has been used for operating the Safari reactor and for targets for the production of Mo-99 for medical isotopes. The rest is in inventory and is under IAEA safeguards.

The RSA spent fuel is similar technically to the U.S. origin spent fuel. It may be possible to satisfy the requirements of the U.S. National Environmental Policy Act (NEPA) for the RSA fuel by amending the Record of Decision (ROD) of December 2004, which was issued by the Secretary of Energy to extend the time period for the FRRSNF program [8]. The first Record of Decision was issued by the Secretary of Energy with regard to the Foreign Research Reactor Spent Nuclear Fuel program after the completion of a multi-volume Environmental Impact Statement which was undertaken to satisfy NEPA requirements [9]. The ROD specifies that the Research Reactor Spent Fuel Return program is for U.S.-origin fuel. A strong case can be made that it is both appropriate and desirable for the U.S. to provide a disposition path for this fuel. In order to accomplish this, it should not be necessary to undertake a new Environmental Impact Statement because the RSA nuclear fuel is similar technically to the U.S.-origin fuel and would be sent to the same facilities in the U.S.

Chile

As noted previously, Chile completed the conversion of the La Reina reactor to LEU in May 2006. All of the spent fuel containing U.S. origin HEU has been shipped to the United States of America. However, there remains approximately 17 KG of U.K.- origin HEU in spent fuel at La Reina and approximately 4 KG of French- origin HEU in lightly-irradiated fuel at Lo Aguire (a reactor that is essentially mothballed). This case is similar to that of South Africa, in that the HEU fuels at La Reina and at Lo Aguire have technical characteristics similar to the U.S. origin material and therefore could be sent to Savannah River or Y-12, where similar U.S. origin spent fuel has been sent. Again, it may be worth exploring the possibility of satisfying U.S. NEPA requirements by amending the Record of Decision.

6. Cases involving reactors unlikely to operate

The TRICO II Triga reactor in the Congo is a case that very much needs to be addressed. There are 144 slightly irradiated LEU fuel assemblies (19 percent enrichment) and nine fresh LEU assemblies at the reactor site. This material is not a proliferation concern, but could be used to make a Radioactive Dispersal Device (RDD, or "dirty bomb). This reactor is eligible to ship its LEU spent fuel to the United States of America under the Foreign Research Reactor Fuel Return program. This has not been done because of the civil unrest and internal violence among militias. However, there is a possibility that the situation will stabilize after elections in 2006. When the U.S. Embassy indicates that the security situation will allow the spent fuel to be moved, DOE needs to be ready to seize the window of opportunity and act quickly. Circumstances in the Congo are such that DOE cannot be sure about how long the internal stability would last.

The research reactor operator has made it clear for some time that he expects to get something in return for agreeing to have the spent fuel shipped. U.S. assistance with technical projects could be explored in advance..

7. Conclusion

The Foreign Research Reactor Spent Nuclear Fuel program and the RERTR program are now both part of the Global Threat Reduction Initiative in NNSA. This, along with the recent reorganization in NNSA along regional lines (as opposed to the previous programmatic organization,) affords an opportunity to look at the broad picture with respect to nuclear materials that need to be removed from a country. This "big picture" approach, together with creative thinking, should allow DOE to achieve greater results than under a "piecemeal" approach.

In addition, with respect to to "other than high income countries" with fresh and partialy irradiated fuel, often in the reactor core, DOE needs to recognize that these countries are unlikely to ship this fuel to the U.S. unless they receive sufficient compensation to cover the cost of a LEU core. It is not reasonable to expect these countries to accept a situation in which they are worse off if they sent the HEU fuel in the core to the U.S. than they would be if they did not send it. Countries in these circumstances need to be made whole.

REFERENCES

[1] Agreement between the Republic of Argentina, the Federative Republic of Brazil, the Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials (ABACC) and the IAEA for the Application of Safeguards, INFCIRC/435, IAEA, Vienna (1994).

[2] Record of Decision for the Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel, Federal Register, Vol. 61, No. 97, 17 May 1996.

[3] J.E. MATOS, J.E., LELL, R.M., "Feasibility Study of Potential LEU Fuels for a Generic MNSR Reactor", Reduced Enrichment for Research and Test Reactors (Proc. Int. Mtg, 6-10 November 2005, Boston, Massachusetts). Available at: http://www.rertr.anl.gov/RERTR27/PDF/S10-2_Matos.pdf

[4] ADELFANG, P., GOLDMAN, I.N., "Latest IAEA Activities Related to Research Reactor Conversion and Fuel Returns Programmes", Reduced Enrichment for Research and Test Reactors (Proc. Int. Mtg, 6-10 November 2005, Boston, Massachusetts). Available at: http://www.rertr.anl.gov/RERTR27/PDF/S2-2_Adelfang.pdf

[5] "Environmental Assessment of Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel", DOE/EA-0912, USDOE, Washington DC. (1994).
[6] Agreement of 18 November 1977 Between the United States of America and the Agency for the Application of Safeguards in the United States of America, INFCIRC/288, IAEA, Vienna (1981).
[7] INFCIRC/435, op. cit.
[8] Revision of the Record of Decision for a Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel, Federal Register, Vol.69, No. 230, 1 December 2004.
[9] Final Environmental Impact Statement, Proposed Nuclear Weapons Non-Proliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel, DOE/EIS-0218F, USDOE, Washington DC. (1996).

NAC's packaging and transportation capabilities for non-proliferation take-back programmes

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Abstract. Over the past 30 years, NAC International has been supporting the U.S. Department of Energy (DOE) with non-proliferation initiatives addressing not only sensitive missions, such as the removal of the Iraqi, Republic of China and Georgian fuel, but also the long term take back effort, such as the Foreign Research Reactor (FRR) program. NAC is DOE's sole-source provider for Other Than High Income (OTHI) Country shipments performed under the FRR program.

NAC owns and operates a fleet of eight NAC-LWT casks, each of which can transport up to 42 MTR fuel assemblies. NAC has gained extensive experience with the back-end programs of research reactors by offering comprehensive services for packaging and transportation of the spent fuel. The services include cask licensing, fuel preparations, cask rental, technical support of cask loading and preparations for shipment, and the physical transport of the spent fuel casks from the shipping facility to the storage facility. The NAC-LWT cask has a unique foreign certification record for spent fuel. In addition to the 2 500 spent fuel assemblies packaged and transported within the United States, it has been validated in 25 countries to allow the removal of more than 3 000 spent fuel assemblies internationally.

The transport of spent fuel is regulated by complex and specific safety and security regulations which are constantly evolving to meet the ever-changing environment. Security requirements have been, and continue to be, revised with more and more country-specific measures.

NAC's role is to assist the shipper in understanding and meeting these obligations to prepare and ship the cask in conformance with all of the applicable regulations. This paper describes NAC's capabilities and concludes with the beneficial lessons learned from extensive experience for the continuation of safe and secure shipments of research reactor spent nuclear fuel.

1. Introduction

NAC International offers comprehensive services for removal of spent nuclear fuel from research reactors wherever they are located, in research institutes or in universities. These services include a various range of skills such as design, licensing, engineering, manufacturing, international transportation expertise, cask operations, communication, adaptability to different environments, and more.

The paper describes NAC's capabilities including a description of the NAC-LWT casks, transport capabilities and also provides some records of NAC's worldwide experience.

2. The NAC-LWT cask

The NAC-LWT Type B transportation cask was designed by NAC to be a state-of-the-art replacement for its aging legal weight truck casks, the NAC-1 and the NLI-1/2 models. Each of these casks was designed to accommodate a single PWR assembly or two BWR assemblies. However, the first use of the NAC-LWT was not for the transport of commercial reactor spent fuel but as a participant in a major U.S. Department of Energy non-proliferation campaign to return research reactor fuel from Taiwan to the United States. A total of 125 loaded cask shipments were required to transport the majority of the Taiwan Research Reactor fuel from the reactor and storage facilities to the Savannah River Site in Aiken, South Carolina, USA. The limited infrastructure in the facility, and the discovery of extensively degraded spent fuel dictated the design, certification, and fabrication of a host of specialized equipment with which to complete the project.

These capabilities proved essential several years later when NAC was requested to go into Iraq following Desert Storm, remove the irradiated nuclear fuel from the reactors destroyed by US cruise missiles during the conflict and transport it to Russia. Additional equipment was developed to remove the fuel from clandestine underground storage in the desert, discovered by the IAEA during the post-war inspection program. Since then, the international community and the DOE Foreign Research Reactor spent fuel repatriation program have made extensive use of the NAC-LWT cask fleet due to the size of the fleet (8 casks), the extensive support equipment, the diversity of fuels for which it is certified, and the outstanding site support for which NAC is noted. The fact that the NAC-LWT is certified to the most current U.S. and international standards is essential to its unilateral approval for international multi-country transport.

3. Cask general description

The NAC-LWT is a steel-encased, lead-shielded shipping cask. The main dimensions of the package are described in Table 1 below:

TABLE 1. MAIN CHARACTERISTICS OF NAC-LWT SHIPPING CASK

Dimensions		
Overall Length	199.80 in	5075 mm
Overall Diameter	44.20 in	1120 mm
Cavity Length	180.90 in	4600 mm
Cavity Diameter	13.375 in	340 mm
Weight		
Loaded	25.6 tonnes	24.0 metric tonnes
Empty	24 tonnes	22.4 metric tonnes

The cask has been fabricated with interior and exterior electro-polished surfaces to minimize contamination spread and simplify decontamination. The cask includes primary and secondary trunnions allowing for a variety of lift configurations and redundant lift for those facilities requiring it.

The NAC-LWT basket system developed for research reactor use is a 7-compartment arrangement with baskets stacked on top of each other to fill the cask cavity. Variations of this basic arrangement are available for plate type MTR fuel, DIDO type MTR fuel, TRIGA pins and TRIGA cluster assemblies. The baskets can be provided in various lengths for cropped or un-cropped fuel. Up to 42 MTR assemblies can be shipped cropped, 28 un-cropped. NAC owns and operate a fleet of 8 NAC-LWTs offering the opportunity to perform large quantity shipments (up to 336 MTR fuel elements within one shipment). A photo and a sketch of the NAC-LWT cask are shown in Fig. 1.

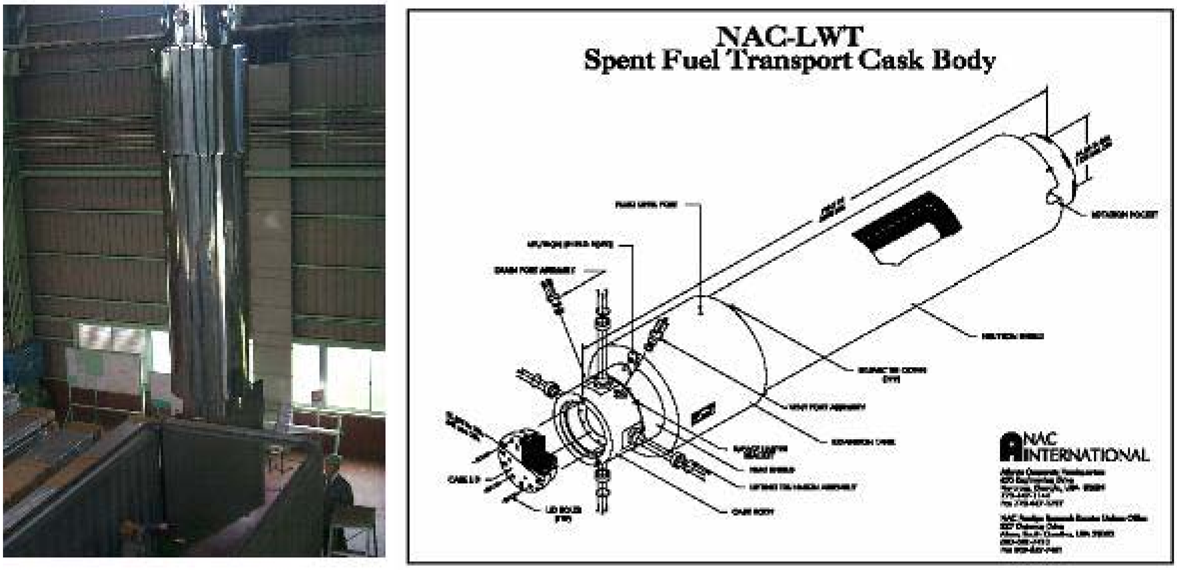

FIG. 1. A photo and a sketch of the NAC-LWT cask.

4. Cask certification

The NAC-LWT is currently designated as USA/9225/B(U)F-96, signifying its certification as a Type B fissile package to the most recent U.S. Regulations.

The other consideration associated with certification is the breadth of the current cask certification and the extraordinary ability of the cask to accommodate different and sometimes exotic fuels. Due to its design for long commercial reactor fuel assemblies, the cask cavity is lengthy but of narrow bore. This provides extremely favorable criticality characteristics and facilitates high decay heat accommodation. With the exception of research reactor fuels exceeding the cavity diameter, virtually all research reactor fuels can be accommodated in a high capacity loading configuration. The current certification envelopes a wide variety of plate and cylindrical MTR fuel forms and pin and cluster forms of TRIGA fuel. The Certificate of Compliance includes certification for damaged research reactor fuel, plates, partial plates, and rubble.

In addition to its US certification, the NAC-LWT cask has achieved a unique record of foreign validations as it has been certified in more than 25 countries.

5. Support equipment and facility compatibility

Another measure of the adaptability of the NAC-LWT cask system is the diversity of support equipment available to complement it and allow any loading and unloading operations in facilities with restricted capabilities (crane, pool, etc.). As noted before, the NAC-LWT's initial applications dictated that NAC engineer a unique assemblage of support equipment. These include several different basket designs, containers for varying forms of failed fuel, retrieval tools, grapples, and cask loading systems. The system selected by NAC for cask loading has become a precursor to that now being implemented at commercial utilities for their dry cask storage systems, namely a transfer cask system capable of handling one fully loaded basket or canister of fuel. The rationale for the design used by NAC and now adopted by others is the absence or restricted capability of a spent fuel pool capable of loading a transport cask under water. Generally speaking, in the research reactor environment the cask is handled outside the facility using a portable crane (so there is no challenge to the facility crane). A transfer cask with a series of shield gates is used to remove the fuel from its storage area and to transfer the contents to the transportation cask. All the transfer operations are accomplished in a fully shielded configuration. Additional equipment was developed by NAC in 1998 for the SNF from Tbilisi, Georgia due to pool size and minimal crane capacity. The newly designed equipment can be used to transfer the fuel element from the reactor pool to the transport cask using a 3 tonne crane in the fuel storage area. Fig. 2 shows some pictures of the transfer cask.

FIG. 2. The tranfer cask.

6. Shipment configuration

Although the cask was designed to allow for legal weight shipment, its normal shipment configuration is in a closed 20 foot ISO container. This has been done for a host of reasons; it facilitates inter-modal transport, it protects the cask from road or rail film and grime, it alleviates concerns about contamination spread during rain storms, and it allows for transport on a standard high load rating single drop trailer or rail car. With the exception of the Class 7 hazard postings, the ISO looks very much like many other motor freight shipments on the interstate highway system. This configuration is considered to have significant security advantages as well.

7. NAC technical support

NAC maintains a staff of qualified personnel with documented training and radiation worker certification to assist reactor operator personnel in a fuel handling evolution. Typically, two NAC engineers provide assistance to the site personnel during the cask loading operations. It takes usually 3 days to loading the first cask as a dry run is always performed and 2 days for the additional casks. NAC also provides cask and transfer system operating procedures.

8. Transportation

Transporting spent nuclear fuel is a complex activity especially when multiple countries are involved. It has become even more difficult with the evolution of our environment. With on-going revision of security requirements, common carrier are more and more reluctant to accept radioactive cargo, even for shipment of empty casks. Regardless of the difficulties, communication is a key element for the success of a project as many entities are involved in a shipment (shipper, receiver, carriers, competent authorities, etc). Furthermore, NAC's role is not limit to moving the material but it is also our responsibility to assist the shipper in preparing all the shipping declaration in accordance with all the applicable regulations. As the owner of the material, it is the shipper responsibility to perform the packing, testing of the cask and also issue the shipping declarations. For facilities which are shipping every 5 or 10 years, it is very challenging to understand and absorb all the shipping regulations. This is the reason why NAC offers its valuable knowledge and structure to ensure the safe and secure shipment of research reactor SNF. Some scenes of a transportation operation are shown on Fig. 3.

FIG. 3. Scenes of a transportation operation.

9. NAC's experience

NAC has unique records for shipments of Spent Nuclear Fuel from Research reactors:

- 3 700 fuel elements under the FRR Program (53 casks)
- 2 300 fuel elements under US domestic program (56 casks)
- 2 000 fuel elements under various other international program (130).

The world map shon in Fig. 4 indicates the countries in which the NAC-LWT transport cask has been used.

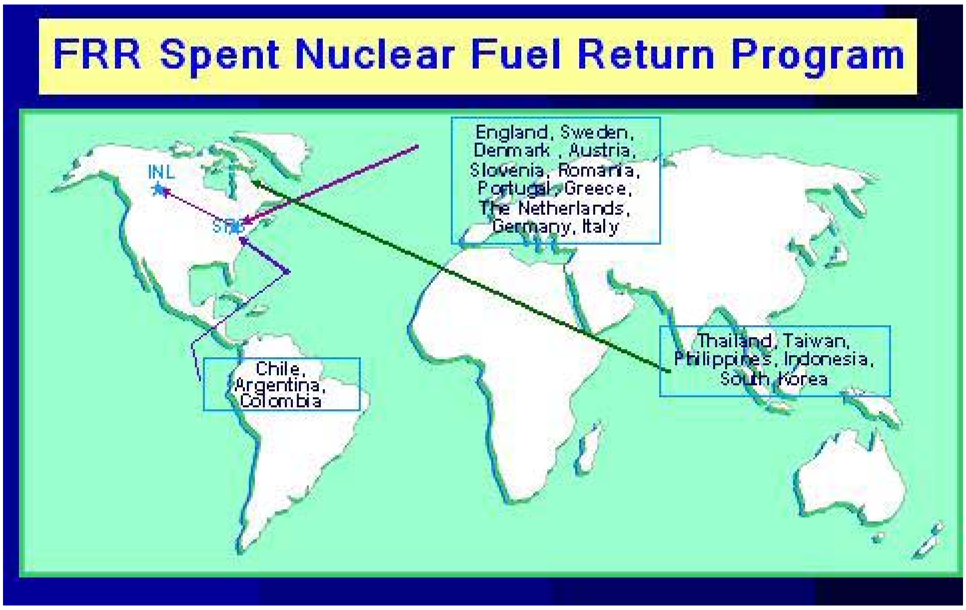

FIG. 4. Countries in which the NAC-LWT transport cask has been used.

10. Summary

Over the years NAC has accumulated unique lessons which are used to improve our services.

The NAC-LWT cask offers significant advantages:

- Multiple contents licensed
- Large capacity with up to 42 MTR fuel elements per cask
- Large fleet of casks with 8 NAC-LWTs
- Loading and unloading flexibility by the use of a dry transfer system

- Proven international experience with cask and dry transfer
- Highly qualified support services

Transportation will remain a complex activity in our new environment but responsiveness and professionalism are key elements to the performance of safe and secure shipments os researach reactor spent nuclear fuel. For all these reasons, NAC International is a transport solution to face the challenging tasks of a spent fuel shipment operation. NAC will be pleased to provide more details and information to anyone interested.

TN International research reactors spent nuclear fuel transport experiences and capabilities

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Abstract. For more than forty years, TN International supports the research reactors community in safely performing national or international transport of radioactive materials (fresh and spent fuel).

TN International is bringing its experience to the various actors of the Global Threat Reduction Initiative Program (GTRI) in particular the Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) acceptance programme and the Russian Research Reactor Fuel Return (RRRFR) programme.

TN International owns and operates a fleet of three TN™-MTR casks. Each cask can transport up to 68 MTR fuel elements. Specific baskets have been developed for transport spent fuel to the US as well as for the Mayak site.

During the last decades, TN International has contributed to non-proliferation initiatives by loading and transporting casks in more than 10 countries. With the support of the AREVA network, Transnuclear Inc. in particular, TN International provides transportation services, including cask and technical assistance on site, to different Research Reactors under US Department of Energy and IAEA frameworks.

This presentation will summarize our experience and futures activities for the coming years in the framework of non-proliferation initiatives.

1. Introduction

Mandated by the U.S. Secretary of Energy, the National Nuclear Security Administration manages, consolidates and speeds up the return of high risk materials in the framework of the Global Threat Reduction Initiative (GTRI).

The office of the Global Threat Reduction NNSA manages several programmes including the FRRSNF acceptance programme, the RRRFR programme, the RERTR (Reduced Enrichment for Research and Test Reactors) and the Emerging Threats and GAP Material programme.

In the framework of the FRRSNF and RRRFR programmes, a significant number of spent fuel shipments are to be organised from the countries eligible for the two programmes to the two countries where the uranium originated (the United States and Russia).

For the RERTR programme, several shipments of fresh fuel (LEU) to reactors which are to convert to LEU need to be organised.

TN International has for more than 40 years worked in the various phases of the research reactor cycle in organising and carrying out nuclear materials transportation all over the world. This paper will demonstrate TN International's capacity to support those involved in the GTRI programme.

2. FFRSNF Acceptance Programme

For many years TN International has worked in the transportation of spent fuel from research reactors, in particular in the framework of the FRRSNF acceptance programme.

This programme has enabled and still enables scores of reactors in around forty countries to send their spent fuel, whose uranium came from the U.S., back to the U.S. in particular to the Savannah River Site Laboratory. In November 2004, the U.S. Secretary of Energy announced the extension of the FRRSNF acceptance programme to May 2019 for fuel irradiated before May 2016.

When the programme started, three or four shipments were organised mobilising around six casks per shipment, but over the last years the number of shipments has been halved. In our view, this reduction in the number of transports is due to a combination of the reactors' efforts in the field of fuel management, the shutdown of certain reactors and the flexibility concerning the date of return of spent fuel to the U.S. caused in particular by the extension of the programme.

3. RRRFR Programme

The Russian Research Reactor Fuel Return Programme concerns around twenty countries which have research reactors which have used, and in some cases still use, Russian fuel. For these reactors the programme means their fuels can be sent back to Russia, in particular to the Mayak site.

The United States, Russia and the IAEA work together to make a success of the programme. The first objective has been the return of fresh fuel and over the last two years several shipments have been carried out (Czech Republic, Uzbekistan, etc.)

Now efforts will turn to spent fuel. The objective set is the return to Russia of around 15 000 fuel elements before 2010. It should be noted that most of these fuel elements to be returned are enriched in uranium 235 by more than 20%. The first spent fuel shipment organised in the framework of the RRRFR Programme concerns the Uzbekistan research reactor. This shipment will take two years to implement. This example shows how complex it is to organize a shipment which is to cross several countries.

There is one particular case, the Vincia research reactor in Serbia. It represents around 8 000 small size fuel elements. The IAEA treats the return of these fuel elements to Russia separately from the other reactors eligible for the RRRFR programme.

The existing Russian transport cask fleet (TUK 19) will not allow the objectives set to be reached on account of its low capacities (4 fuel elements) and even though, under the aegis of the IAEA, the manufacture of 10 casks of greater capacity has been launched, we feel it will be necessary to mobilize the existing world fleet of casks if we are to reach the return objective between now and 2010.

4. TN-MTR Cask and shipment

At the end of the 90s TN International developed a new cask, the TN^{TM}-MTR in accordance with the TSR-1 [1] regulation, to replace the IU04 cask.

The IU04 cask fleet was managed by TN International (whose name at the time was TRANSNUCLEAIRE). It was often lent for the FRRSNF programme and has been used in many countries, for example Denmark, Portugal, Italy, Venezuela, etc.

As IU04 no longer met the new international regulations, it was necessary to develop a new cask which did. With the TN^{TM}-MTR it is possible to transport up to 68 spent fuel elements. The TN^{TM}-MTR fleet numbers four.

The TN^{TM}-MTR is used to transport spent fuel to the La Hague plant in the framework of the reprocessing contracts of AREVA NC. The shipments are from French reactors (of the CEA, French Atomic Energy Commission, and ILL) as well as foreign ones (ANSTO, from Australia and BR2 from Belgium).

In order to comply with the different needs and the different spent fuels geometries, TN International has designed several types of baskets allowing to transport up to 68 spent fuel elements: MTR68 (transport up to 68 spent fuel elements); MTR52 (transport up to 52 spent fuel elements); MTR-52S and MTR-44 (transport up to 44 spent fuel elements). A specific type of basket (called TN-MTR52S) has been developed to satisfy the requirements of the American authorities. This basket can transport up to 52 fuel elements and was used in 2001 in the framework of the last shipment from the Risoe reactor (Denmark) to the U.S. The TN^{TM}-MTR was accepted and used with no problem in the Savannah River installations.

For the American part of the transport we can rely on another company in the AREVA group, TRANSNUCLEAR Inc., and the close links between our companies ensure satisfactory cooperation in the shipments to the U.S.

The TN^{TM}-MTR is suitable for all types of loading/unloading under water (in the pool) or dry. For the CEA, TN International has developed a system by which the TN^{TM}-MTR can be vertically connected to a hot cell.

The TN^{TM}-MTR is accredited for a large number of MTR and TRIGA fuels. Fuels of Russian origin (IRT, EK-36, EK-10, etc.) have been analysed and can easily be integrated into the TN^{TM}-MTR accreditation. The TN^{TM}-MTR accreditation was obtained in April 2002 for a period of 5 years. The extension for this accreditation will be submitted in mid 2006 to obtain a new five year accreditation.

We would like to point out that, working from information supplied by the IAEA and the Russian authorities, and with the collaboration of AREVA Moscow, unloading operations at Mayak have been studied. We may modify the MTR68 baskets to be ready, if necessary, to unload the basket loaded directly in the cell.

TN International makes available to small research reactors, which cannot receive the TN^{TM}-MTR, a transfer system for loading spent fuels. The system is composed of a radiological protection 2 metres high which is placed on the cask filled with water, and a transfer system allowing the fuel elements to be transferred in complete safety.

This transfer system has already been used with the IU04 (e.g. in Italy, Venezuela and France) and also with the TN^{TM}-MTR (in France). It will be used in France again in 2007.

TN International has several decades of experience in the international transport of spent fuels by road, rail and sea and can rely on the collaboration of companies in the AREVA group such as AREVA Moscow and TRANSNUCLEAR Inc. Consequently TN International can offer efficient, reliable and safe solutions for the FRRSNF and RRRFR programmes.

5. The RERTR programme and fresh fuel shipment

Following the initiative of the NNSA, several reactors throughout the world are converting to LEU fuel in the framework of the RERTR Programme.

Via CERCA, a subsidiary of AREVA, which is world leader in the supply of fuel to research reactors, TN International participates in supplying some of the reactors converting to LEU.

TN International regularly (5 to 10 times a year) performs international shipments of fresh MTR and TRIGA fuel, leaving the CERCA plant, using all means of road, rail and sea transport, and taking account of French and foreign Physical Protection Requirements.

TN International's latest work projects are as follows: Belgium, Romania, the United States of America and Morocco.

For most of these shipments, the cask used was the TNBGC-1, designed by TN International. It is known worldwide and has accreditation in France validated in the U.S., in Russia, and in different countries in Europe and the rest of the world.

Due to its small size and its not excessive weight, the TNBGC-1 is easy to handle and requires no special means. The TNBGC-1's accreditation covers a large number of MTR and TRIGA fuels and usually enables them to be transported by air.

6. Emerging Threats and Gap Material Programme

The objective of the Emerging Threats and Gap Material Programme is to address vulnerable, high-risk, nuclear and radiological materials that could be of terrorist concern throughout the world that are not currently being addressed under existing programmes. By creating an initiative that comprehensively addresses these materials, the Office of Global Threat Reduction will be able to quickly and more effectively respond to evolving threats requiring rapid removal of nuclear or radioactive materials worldwide.

Through AREVA's collaboration within NNSA, TN International can undertake the transport of different types of materials: uranium (HEU & LEU) and plutonium. It should also be noted that this programme could also concern spent fuels which are not covered by the FRRSNF and RRRFR programmes.

TN International can also propose several types of casks which all satisfy TSR-1 [1]:

- The TNBGC-1: this cask enables uranium (HEU & LEU) in all its forms to be transported by all means of road, rail, sea and air transport. The transport of powdered or metal plutonium is also authorized (except by air).
- The FS47: this cask allows for the transport of plutonium.
- The TNUO2: this cask is accredited for the transport of metallic uranium (LEU & HEU).

TN International has long experience of the international road, sea and air transport of uranium and plutonium in conformity with the international safety and physical protection regulations.

7. Conclusion

In the framework of the various GTRI programmes, we therefore expect an increase in the number of shipments, in particular with the start-up of RRRFR and the continuing FFRSNF programme.

TN international can mobilize a fleet of efficient casks for the different phases of the fuel cycle of research reactors, which can be used in the transportation of different types of nuclear materials (uranium, plutonium, fresh and spent fuels).

We intend to maintain close links with the various parties involved (reactors, NNSA, IAEA, etc.), in order to assess the demand and analyse the capacity of our casks to satisfy it. In this endeavour TN International can rely on the collaboration of the different entities of the AREVA group and in particular AREVA Moscow, AREVA NC Inc; and TRANSNUCLEAR Inc.

TN International has long experience of international transport in all the various modes (road, rail, sea and air) and of all types of materials all over the world. TN International permanently updates its competences by reviewing regulatory and technological developments, by training its teams and developing its casks.

REFERENCE

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Regulations for the Safe Transport of Radioactive Material, Safety Standard Series No. TS-R-1, IAEA, Vienna (1996).

NCS experience in the shipment of spent nuclear fuel to US DOE

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Abstract. On 13 May 1996, the US-Department of Energy issued a Record of Decision relating to a "Nuclear-Weapons Non-proliferation Policy concerning Foreign Research Reactor Spent Nuclear Fuel (FRRSNF)". The goal of the long-term policy was originally to recover enriched uranium exported from the United States by 12 May 2009. Last year, this time limit has been extended to 2019.

Within the scope of this program, NCS is one of the companies organizing the complete service, which includes cask rental, cask loading, and shipment by road, rail, or sea, so the program will be successful.

NCS has been involved in the management of the shipments from Europe, South America, and Australia. Each shipment has proven to be unique in some aspect, and the lessons learned from each of these shipments have been incorporated, so future shipments will run more smoothly.

The purpose of this paper is to describe the possibilities offered by NCS in order to assure that these transport operations will then be carried out successfully. We can further rely on our past experience, which goes back to 1977.

1. Introduction

1.1. NCS overview

NCS was founded in 1985. The common stock amounts to 10 million DM. The head office of the company is located in Hanau/Germany.

The objectives of the company are: To carry on the business of forwarding, shipping, commission agency, packaging services, insurance broker services and other operations, and especially services in the nuclear sector.

The NCS business site with offices, secured truck parking lot and three warehouses for the interim storage of contaminated containers and equipment, as well as maintenance facilities, are situated at Hanau, which provides the necessary infrastructure with security and medical services, as well as protection against radiation.

NCS has a staff of 130 persons with long-standing experience in the field of packaging and transport of dangerous goods, and particularly radioactive goods.

1.2. NCS international

To ensure a continuous supply of electric energy is a global challenge. Today and in the future, close international co-operation is and will be the key to achieve this task.

The international fuel cycle is highlighted by special standards and extremely high requirements, calling for reliable as well as qualified enterprises to solve the logistic problems of this "hard to please" industry.

NCS has the necessary know-how and up-to-date information in front-end as well as back-end areas, to find practical solutions for all transport problems.

In all major countries with a nuclear industry, NCS is represented by experienced and well known partners qualified to arrange nuclear transports smoothly and reliably, according to the laws and the regulations of the respective country. In addition to that, NCS has offices in Almaty/Kazakhstan, Amsterdam/Netherlands, Beijing/China, Burtonsville/USA, Cairo/Egypt, Montélimar/France, St.Petersburg/Russia, Budapest/Hungary and Belgrade/Serbia.

NCS offers comprehensive transport solutions, optimized as far as safety and costs are concerned.

When it was known that the US Department of Energy (DOE) was taking back fuel assemblies from research reactors containing fuel of American origin, it became clear for us that here was a challenge to the transport providers and the owners of transport packages, in order to assure the success of the 10 to 13 years programme. In this respect, NCS reached an agreement with GNS company - Gesellschaft für Nuklear-Service mbH (Company for Nuclear Services Ltd.), which specializes in the disposal and removal of waste and spent nuclear fuel, and especially in handling, obtaining of approvals and in handling transport packages, to work together within the scope of a consortium in the field of nuclear transports for research centers and research reactors (NCS/GNS Consortium). The objective is to provide all services from one hand and to manage projects without interface problems, to the advantage of the customers. For this purpose, the members of the consortium will pool their specific knowledge and use their respective packages and equipment in common.

2. Transport casks

The following transport casks, which are approved according to the 85/96 – International Atomic Energy Agency regulations, are available:

2.1. TN 7-2

This cask, of which 2 exist, has a maximum transport capacity of either 64 cut square section MTR fuel assemblies or 60 cut round section MTR fuel assemblies, the weight of the cask being 21 t. The U-235 enrichment varies from 20 to 93 % according to the type of fuel. The fuel assemblies are loaded under water into the transport baskets. These are then placed one above the other in the cask. Due to the weight of the cask, which is shown in Fig. 1, the latter can only be used in ponds which have sufficiently powerful cranes. A 20' Open Hard Top Container is used to transport the TN 7/2, so that transfer from road to rail or ship causes no problems.

2.2. GNS-11

The GNS-11 has a maximum transport capacity of 33 square section or 28 round section MTR fuel assemblies, a maximum of 90 TRIGA fuel assemblies or one 200 l-drum containing high radioactive waste, the maximum cask weight being 11.5 t. The U-235 enrichment again varies from 20 to 93 %, according to the type of fuel assemblies.

Due to its relatively small weight of 11.5 t, this cask can be used in a large number of facilities.

The GNS-11 transport casks, shown in Fig. 2, are also transported in a 20' Open Hard-Top Container. There are 2 of these casks available.

2.3. GNS-16

The GNS-16 cask, which has been in operation since May 1998, has the same capacity for MTR fuel assemblies as the GNS-11. The total weight of the cask including the shock absorber is about 15.3 t.

The cask is also approved for up to 90 TRIGA fuel assemblies with an enrichment of 21 % of U-235.

FIG. 1. The TN 7-2 transport cask.

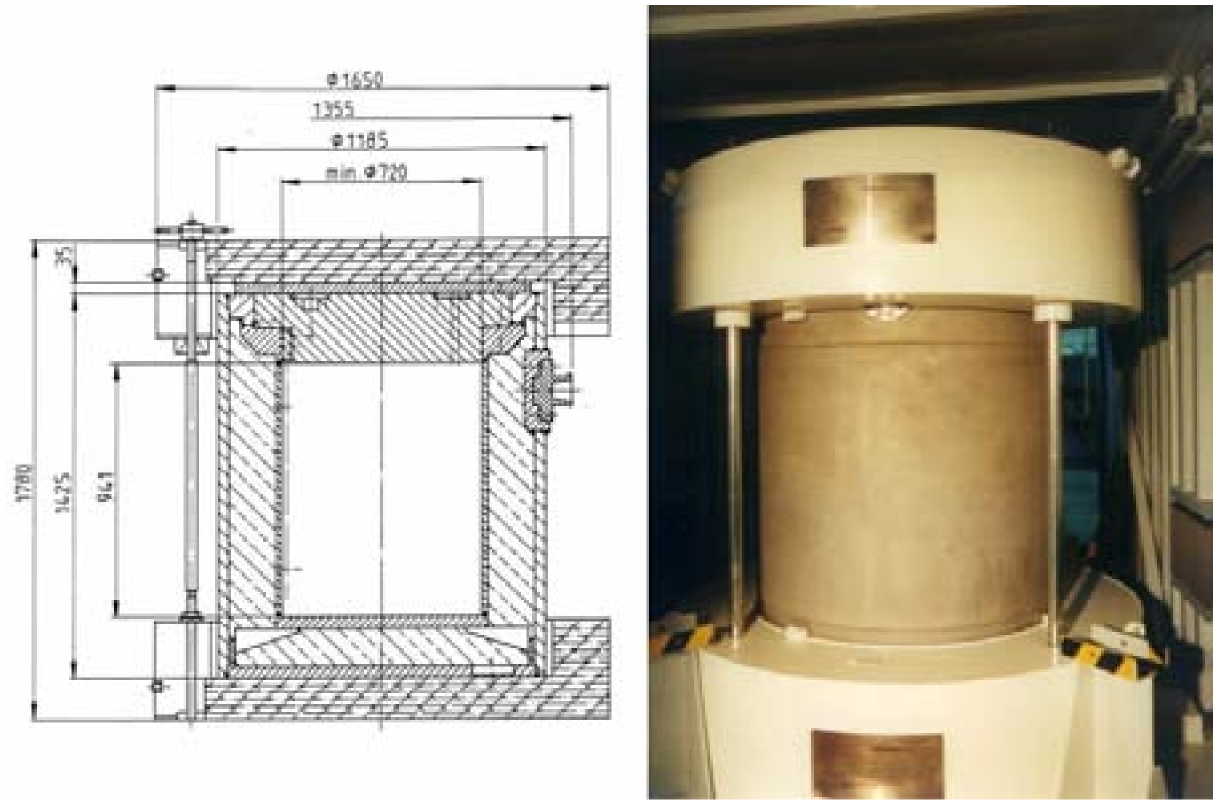

FIG. 2. The GNS 11 transport cask.

The loading of TRIGA fuel assemblies will be carried out by means of so-called transfer baskets which can take 3 to 6 assemblies each.

These transfer baskets are then loaded into the 15 transport positions.

The GNS-16, shown in Fig. 3, are also transported in a 20' Open Hard-Top Container. There are 2 of these casks available.

FIG. 3. The GNS 16 transport cask.

3. Transfer station

Considering that there are research facilities which cannot handle weights of 12 t or more, for they we have developed a transfer station, shown in Fig. 4, so as to assure the capability to remove the spent nuclear fuel too..

FIG. 4. The transfer station

The transfer station includes:

- the loading lock
- the adapter plate
- the water tank
- the transfer cask.

The fuel assemblies are loaded into the ready “transfer baskets” and transported to the loading lock, using the transfer cask. When MTR fuel assemblies must be loaded, the transfer cask can take 1 MTR fuel assembly at a time. If the transfer cask is in position on the loading lock, the “transfer basket” may be lowered into the basket shaft of the GNS 16 package by opening the slide door of the loading lock and the rotating lock of the transfer cask.

During the transfer procedure, the transfer basket or the MTR fuel assembly is attached to a suitable gripping device.

When all basket positions have been loaded, the water tank is set onto the adapter plate and filled with water, in order to allow for the removal of the loading lock. The shielding lid of the GNS 16 package is then set on, the water is removed by aspiration and the water tank is removed.

The GNS 16 is then emptied and dried, after which a leak check is performed.

The transfer station was used to remove the fuel assemblies from the research facilities of IPEN, Sao Paulo, FRM-Munich and DKFZ-Heidelberg.

4. Maritime transport

As far as maritime transport is concerned, the International Maritime Dangerous Goods (IMDG) Code was supplemented in early 1995 by the INF code, which laid down stringent requirements for ships carrying irradiated nuclear fuel, plutonium or high level radioactive waste.

For the transport of MTR or TRIGA fuel assemblies, one may assume that the activity limit of the total load for INF2, from 4 PBq to 1000 PBq, will be easily reached if transport combinations are foreseen (up to 16 casks per ship load).

The INF Code is implemented in numerous countries where the IMDG Code is considered as the basic regulation for international maritime transport This Code covers matters concerning ship's design, construction and equipment. The requirements of the INF code and the decision of the DOE to use military ports in the USA, de facto exclude the use of routine line shipping services.

Taking these facts into account, NCS has decided to qualify "MV ARNEB" as an INF2 ship together with a German shipping company.

The particularity of MV ARNEB is that the complete unit (vehicle + cask) can be driven into the cargo hold (Ro-Ro), thus keeping transfer times low. In ports which have no Ro-Ro ramps, the casks can of course be lowered with a crane into the cargo hold. In the meantime, NCS has transported fresh MOX fuel assemblies from BNFL/England to a German nuclear power plant, and MOX fuel pins from Hanau to Dounreay with MV ARNEB.

Within the scope of the US-DOE program, the ship MV Arneb has been used 7 times since 1997 under this name and 3 times more after having been sold to BNG in 2002, sailing till then under the name Atl. Osprey.

5. Conclusion

So far, NCS has transported 2606 MTR-fuel assemblies in 68 shipments to Savannah River Site or using the mentioned transport packages and without any trouble.

We have shipped 202 TRIGA fuel assemblies in 2 shipments to Idaho Falls Site. NCS not only provides advice for all transport and handling problems, NCS also carries out the transports.

Status of spent fuel in the 3MW BAEC MK-II research reactor facility of Bangladesh

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Abstract. Bangladesh has been operating a 3 MW TRIGA MARK II research reactor since 1986. The reactor is installed in the campus of the Atomic Energy Research Establishment (AERE) at Savar, which is located about 40 km northwest of Dhaka. It is one of the main nuclear research facilities in the country. The reactor uses TRIGA LEU fuel with uranium content of 20% by weight. The enrichment level of the fuel is 19.7%. So far the reactor has been operated for 5624 hours with a total cumulative burnup (BU) of 10 690 MWh (445 MWd). The main areas of use are: training of man-power for research reactor operation and applications, radioisotope (RI) production, neutron activation analysis (NAA), neutron radiography (NR) and neutron scattering. Radioisotopes produced to date are: I-131, Sc-46 and Tc-99m. Bangladesh is a peace loving country with a strong commitment towards nuclear nonproliferation. Accordingly, it has signed several multilateral and bilateral agreements, protocols, treaties, etc. prevailing in the International Nuclear Non-proliferation regime. Bangladesh has also signed a Nuclear Cooperation Agreement with the USA on 17 September 1981, which facilitated export of nuclear technology from the USA to Bangladesh. The research reactor was procured under the provisions of this agreement. In 2003, the tenure of the Agreement was extended up to 2012. At present, there does not exist any spent fuel element in the reactor facility. However, with the recently undertaken RI production enhancement program, it is expected that the reactor will start generating spent fuels from the year 2012. It is to be mentioned that Bangladesh is aware of the US DOE's 'Take Back Program' in connection with the research reactor spent fuel of US origin, and is very much interested to take part in this program. The paper presents the current status of handling and storage facilities available for spent fuel and strategy for the safe management spent fuel to be generated from the research reactor in near future.

1. Introduction

The Bangladesh Atomic Energy Commission (BAEC) TRIGA Mark-II research reactor was made critical at 50W for the first time on 14 September 1986 and was commissioned to steady state power of 3 MW in October 1986. Since then, it has been used for manpower training, radioisotope production and various R&D activities in the field of NAA, NR and neutron scattering. During the period, operation of the reactor was interrupted several times due to different incidents encountered mostly in the cooling system of the reactor. One of these incidents was a leakage the Exi-check valve of the primary cooling loop and for this incident the reactor operation was suspended for about 21 months. The most severe of these incidents was the "N-16 Decay Tank Leakage Incident" that took place in 1997 due to pitting corrosion on the bottom of the tank. It is to be mentioned that the corroded bottom portion of the tank was in direct contact with the concrete saddle. As a result of this incident, full power operation of the reactor remained suspended for several years. During that time, the reactor was, however, operated at 250 kW under natural convection cooling mode, so as to cater the needs of the reactor users (e.g., NAA and NR groups) who require lower neutron flux. Operation of the reactor at lower power level was made possible by establishing a temporary by pass connection across the decay tank using local technology. To take the reactor back to normal operation, BAEC implemented a government funded ADP (Annual Development Program) with a total project cost of about 0.8 million US dollar. Under the project, renovation and upgrading of the entire cooling system of the reactor were carried out. The renovated cooling system was successfully commissioned in June 2002 and through this, it was possible to restore the full power operation of the reactor after a long period of about five years. Since July 2004, the reactor is being used for production of I-131 on routine basis. At present 100% of the I-131 (in solution form) requirements of the country is being met by local production.

2. Features of the BAEC TRIGA reactor

The TRIGA Mark-II research reactor of BAEC is a light water cooled, graphite reflected reactor, designed for steady-state and square wave operation up to a power level of 3 MW (thermal) and for pulsing operation with a maximum pulse power of 852 MW [1]. The reactor core is located near the bottom of the reactor tank. The reactor tank is made of 6061-T6 aluminum alloy and has a length of about 8.23 m (27 ft) and a diameter of about 1.98m (6.5ft). It is filled up with about 24 865 liters (6 578 gallons) of demineralized water. The reactor core consists of a total of 100 fuel elements (including 5 fuel follower control rods and 2 instrumented fuel elements), 6 control rods, 18 graphite dummy elements, 1 Dry Central Thimble (DCT), 1 pneumatic transfer system irradiation terminus and 1 Am-Be neutron source (strength: 3Ci). The general characteristics of the reactor are summarized in Table 1.

TABLE 1. CHARACTERISTICS OF BAEC TRIGA REACTOR

Characteristics	BAEC TRIGA Mk-II Reactor
General:	
Reactor Type	Pool Type
Thermal Power	3000 kW
First Criticality	14 September, 1986
Total Operating Hours	5624 Hours (upto July 2006)
Total Fuel Burnup	445 MWd (upto July 2006)
Neutron Flux	9.12×10^{13} $n.cm^{-2}.s^{-1}$ (Max.)
Fuel	
Contents of Uranium	20 %
Enrichment	19.7 %
Cladding	Al SS 304
Chemical Composition	$Er\text{-}U\text{-}ZrH_{1.6}$
Moderator	$ZrH_{1.6}$ and Demineralized Water
Coolant	Demineralized Water
Reflector	Graphite
Control Rod	B_4C

3. Description of fuel element

The fuel element of the BAEC TRIGA reactor is a homogeneous mixture of $Er\text{-}U\text{-}ZrH_{1.6}$, containing about 20% by weight of uranium enriched to about 19.7% U-235 and about 0.47% by weight of Erbium (burnable poison). The hydrogen-to-zirconium atom ratio of the fuel-moderator material is about 1.6 to 1. The active section of the fuel-moderator element is 38.1 cm (15 in.) long and 3.63 cm (1.43 in.) in diameter. The active fuel section together with the top and the bottom graphite reflector pieces are contained in a 0.5 mm (0.02 in.) thick stainless steel cladding. The cladding is welded to the top and bottom end fittings. The top end fitting is grooved and specially shaped to fit and lock into the fuel-handling tool. The overall weight of the fuel element is about 3.64 kg (8 lbs) [2]. The U-235 content is about 100 gm (0.05 lb). Details of the TRIGA fuel are shown in Fig. 1.

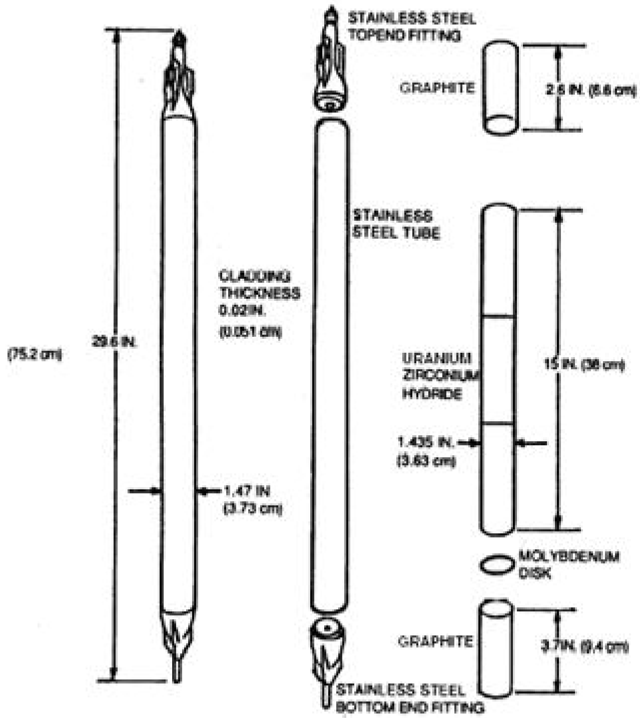

FIG. 1. BAEC TRIGA Fuel Element.

4. Present status of the fuel burnup

To meet the national requirements for medical radioisotope (I-131), reactor operation has been increased significantly in the recent years. The annual burnup for the BAEC research reactor fuel is shown in Fig. 2. The reactor has so far been operated for 5 624 hours with a total cumulative burnup (BU) of 10 690 MWh (445 MWd). The present fuel loading of the BAEC research reactor is good for an accumulated burnup of about 1 200 MWDs. The Reactor Physics and Engineering Division (RPED) of the Institute of Nuclear Science and Technology (INST) is performing the burnup calculations. They are using TRIGAP, MVP-BURN and MCNP-ORIGEN for reactor burnup calculations. From these calculations it has been seen that from September 1986 to December 2005, the burnup of reactor fuel is about 4.5%. So, at present no spent fuel is generated in the BAEC reactor facility. However, with the RI production enhancement program taken recently, it is expected that the reactor will start to generate spent fuels from the year 2012.

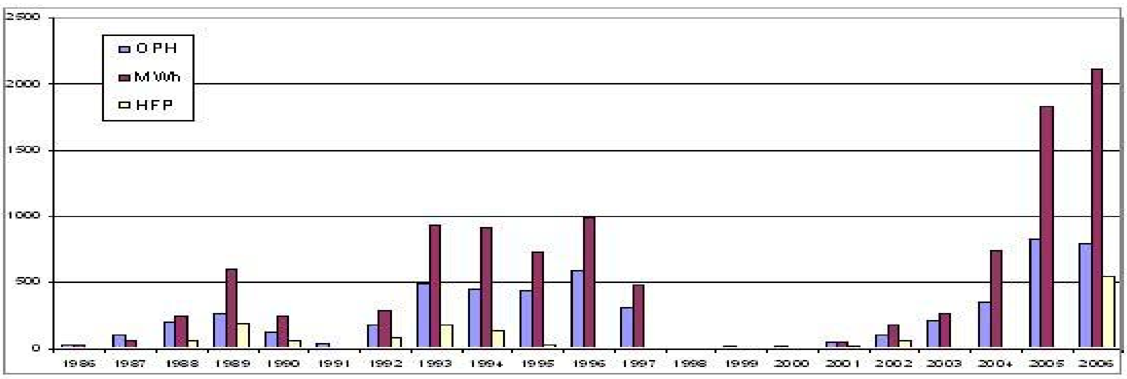

FIG. 2. . Reactor operation data (Sept. 1986 – July 2006).

5. Description of storage facility

5.1. Spent fuel storage facility

The reactor facility is equipped with three spent fuel storage pits located at the ground floor of the reactor hall. The pits are made of stainless steel pipes of diameter 25.4 cm (10 inches) and of depth 457.2 cm (15 feet). Each one of the pits is provided with a lock on its stainless steel cover plate to limit access to the pit and also an M.S. cover plate (with lifting hook) that fits flush with the floor. Each storage pit is capable of storing 19 spent fuel elements (total capacity: 57). Figure 3 shows the detailed drawing of the fuel storage pit. For storing the fuel elements into these pits, suitable storage racks are needed. At present the BAEC reactor facility does not have any rack of such kind. However, efforts have been undertaken to design and develop the storage racks compatible with the storage pits and also with the handling and lifting facilities available in the reactor hall. Besides these pits, there are three submerged fuel storage racks located along the inner wall of the reactor tank at a depth of about 610 cm (20 feet). The purpose of these racks is to provide temporary storage for the spent fuel and for the graphite dummy elements.

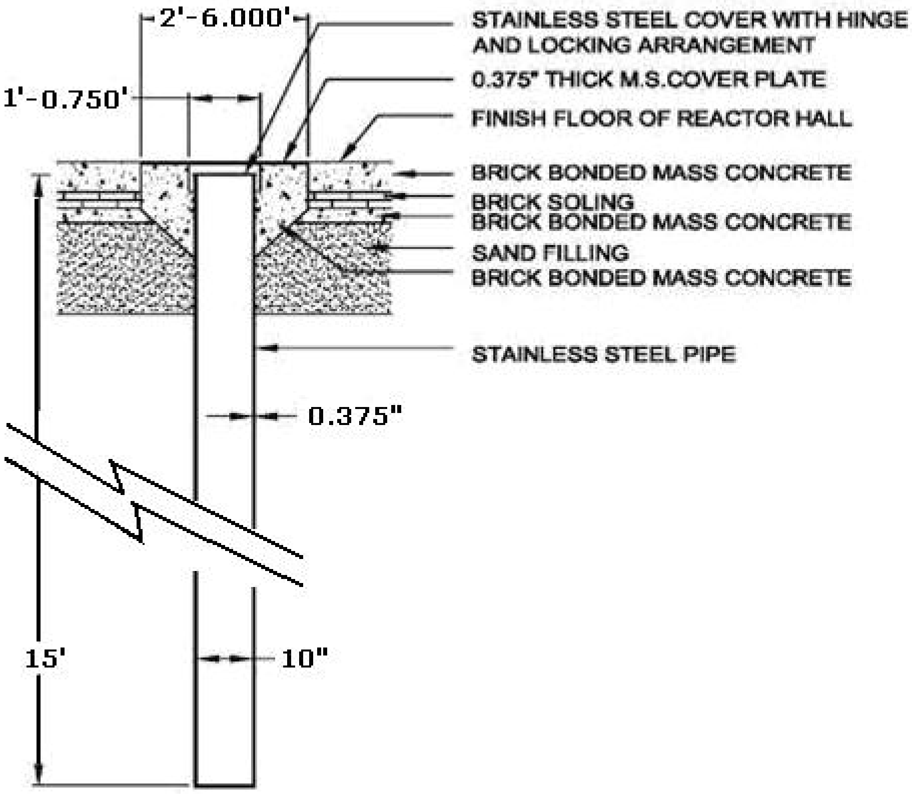

FIG. 3. Typical spent fuel storage pit of BAEC TRIGA LEU fuel.

Each one of the racks is capable of holding 10 fuel elements (total capacity 3 × 10 = 30). Figure 4 represents the photographic view of the storage rack.

5.2. Central waste processing and storage facility (CWPSF)

Recently a Central Radioactive Waste Processing and Storage Facility (CWPSE) has been constructed in AERE campus located near the research reactor facility. The activities of this facility include: collection, handling, segregation, characterization, classification, treatment, conditioning, storage and disposal of all kinds of radioactive wastes generated in the country from nuclear installations, and also from application of radioactive materials in medicine, industry, research, agriculture, education, etc.

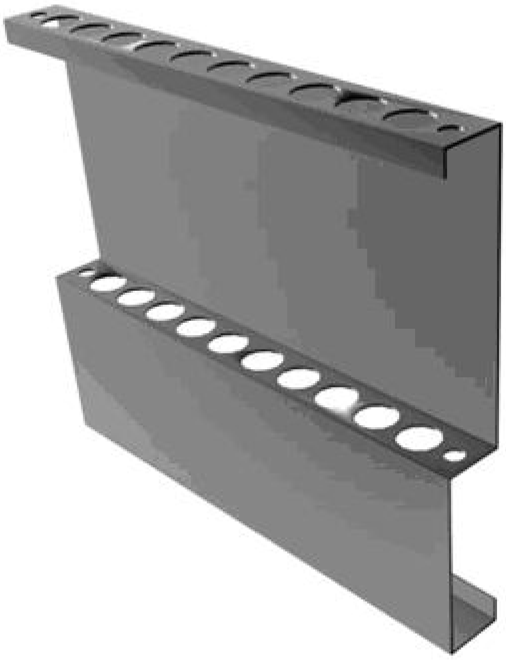

FIG. 4. Fuel storage rack.

The reactor facility is not susceptible to produce liquid waste in bulk quantity. However, any liquid waste produced in the facility will be processed and, if required, stored in the CWPSF. The layout plan of the CWPSF, which has a floor area of about 1 160 Sq. m (12 480 Sq. ft), is shown in Fig. 5. The CWPSF has a liquid waste treatment facility (Aqua-Express) designed for treatment and purification of Low and Intermediate Level Waste (LILW) at a rate of about 300 liters (79.4 gallon) of liquid radioactive wastes per hour using ion-exchange-cum-ultra-filtration technique.

As there does not exist any comprehensive national program for radioactive waste management, the CWPSF will be used as the storage for all sorts of radioactive wastes produced in the reactor and other facilities. CWPSF is equipped with facilities such that solid wastes could be categorized, compressed for volume reduction, and immobilized in the steel drums each having a capacity of 200 liter (52.9 gallon) [IAEA Std. 200 liter steel drum having diameter and height of ~58 cm (22.8 in.) and ~88 cm (34.6 in.) respectively]. The filled up radioactive waste drums will be stored in appropriate storage room of the CWPSF for further decision to be taken in future (disposal). It is to be mentioned that the CWPSF has a provision for storing 112 IAEA standard 200-liter capacity radioactive waste storage steel drums. For conditioning and handling of the radioactive wastes, the facility (CWPSF) is equipped with the followings:

− 40 tonnes (40 000 kg) capacity low force compactor	1 no.
− In-drum cement mixture	1 no.
− Commercial cement mixture	1 no.
− 3 tonnes (3 000 kg) capacity forklift truck	1 no.
− 3 000-liter (794 gallon) capacity LAD (low activity drainage) tank	1 no.
− Sorting machine/box	1 no.
− Decontamination machine/box	1 no.

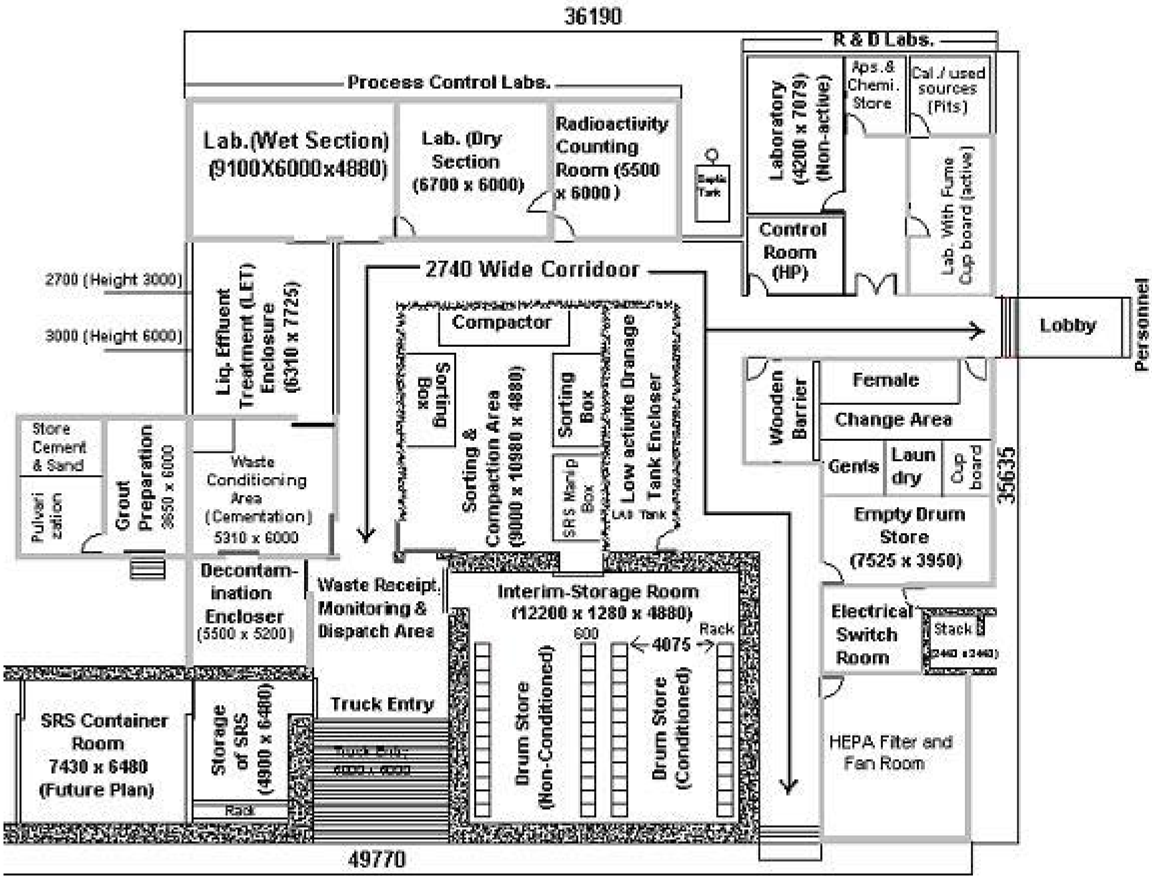

FIG. 5. Central radioactive waste processing and storage facility (CWPSF).

Other equipment/machine/apparatus available in the CWPSF include gamma, beta, alpha spectrometers, UV-visible spectrophotometers, compressive strength testing machine, pH and conductivity meters, weighing scales/balance, water bath, centrifuge, drying oven, sieve shaker, miscellaneous glass and plastic apparatus, etc.

It is to be mentioned that all spent Radium-226 sources (35.6 GBq) collected from different hospitals and industries of the country have been conditioned and safely stored at the CWPSF using 200 liter (52.9 gallon) capacity mild steel drums. The drums are designed and developed in Sibersdorf, Austria and supplied by the IAEA [3][4]. Used Co-60 sources of the gamma irradiator of the Institute of Food and Radiation Biology (IFRB) have also been stored in the CWPSF.

6. Fuel handling infrastructures

6.1. Fuel handling tools

Fuel handling tool is used to handle the fuel elements, graphite dummy elements and the fuel inspection tool calibration elements. Two types of fuel handling tools are available in the reactor facility. One is flexible type and the other is rigid type. Both of the fuel handling tools were supplied by General Atomics of USA, the reactor supplier.

6.2. Fuel cask

At present the BAEC reactor facility has no spent fuel transfer cask for transferring irradiated fuel elements from the reactor pool to the spent fuel storage pits. BAEC has taken measures to design and develop such cask for the reactor facility. Figure 6 shows the proposed fuel transfer cask.

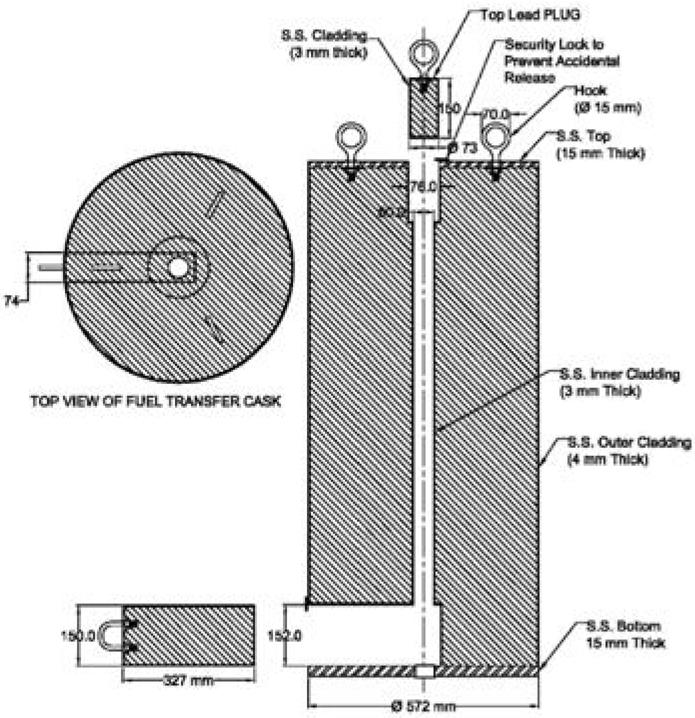

FIG. 6. Cross-sectional view of the proposed fuel transfer cask.

6.3. Crane

An overhead crane of 5 tonnes (5 000 kg) capacity is used for transferring irradiated samples (TeO2 targets for I-131 production). An IAEA supplied lead transfer cask is used to transfer irradiated samples from the reactor top to the U-cask of the RI production laboratory. The weight of the lead transfer cask is about 1.5 (1 500 kg) tonnes. The crane can also be used for handling the proposed spent fuel transfer cask during transfer of spent fuel from reactor pool to the spent fuel storage pits.

7. Spent fuel management policy

Bangladesh has strong commitment towards nuclear nonproliferation and as such, it has signed almost all multilateral and bilateral agreements, protocols, treaties, etc. prevailing in the International Nuclear Non-proliferation regime. Bangladesh signed a Nuclear Cooperation Agreement with the USA on 17 September 1981 [5],[6]. This facilitated export of nuclear technology from the USA to Bangladesh. In recent years, the tenure of the Agreement has been extended up to 2012. It is to be mentioned that Bangladesh is aware of the US DOE's 'Take Back Program' in connection with the research reactor spent fuel of US origin. It would be highly appreciated if all the spent fuels generated in the research reactor of Bangladesh are taken back to the USA under this US DOE's programme.

8. National regulation

The authority for the control of all radiological and nuclear practices in Bangladesh is vested on the BAEC as the competent authority. The legal basis for this control are the 'Nuclear Safety and Radiation Control (NSRC) act 1993 [7] and the Nuclear Safety and the Radiation Control Rules, 1997 [8] which essentially incorporate the requirements of the international Basic Safety Standards [9]. For addressing safety of radiation sources, protection of man and the environment and in compliance with the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management [10], the national NSRC Act (1993) and the NSRC Regulation (1997) are in operation; while the Radioactive Waste and Spent Fuel Management Rule-2005 [11] is awaiting for the approval of the Ministry. BAEC is speeding-up the process of becoming a party to the Joint Convention.

9. Conclusion

Because of various reasons, operation and utilization levels of the BAEC TRIGA reactor were rather low (a cumulative BU of 445 MWDs in about 20 years). However, with the on set of RI production enhancement program, which was undertaken in 2003, the operation of the reactor has increased significantly (Fig. 2). In 2005, total operating hours of the reactor were 830 hours and the corresponding BU was 1 830 MWh (76.3 MWDs). It is expected that in 2006, these figures would increase by a factor of 2. It is also expected that in a couple of years, aspects related to the management and handling of spent fuel will come out to be one of the most important issues to be addressed in the reactor facility. Keeping this in mind, BAEC is doing the needful such that the spent fuel that would be generated from the operation of the 3MW TRIGA research reactor could be managed in a safe and competent way. In order to meet this objective, efforts have been undertaken to design and develop spent fuel storage racks compatible with the existing 3 spent fuel storage pits located at the ground floor of the reactor hall. BAEC has also taken up measures to design and develop spent fuel transfer cask. BAEC expects that the US DOE's 'Take Back Program' would of great use in determining the ultimate fate of the spent fuels generated from the operation of the only nuclear reactor of Bangladesh.

REFERENCES

[1] Safety Analysis Report of the BAEC 3 MW TRIGA Mark-II Research Reactor, BAEC, Dhaka (2004).

[2] GA TECHNOLOGIES, TRIGA Mark II Reactor Mechanical Operation and Maintenance Manual, GA-A17054, GA Project 2299, USA (1984).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Conditioning and Interim Storage of Spent Radium Sources, Technical Report Series No. 886, IAEA, Vienna (1996).

[4] RABBANI, G, ALAM, K., KUDDUS, A., A Proposal for the Management of Spent Radium Sources in Bangladesh, Technical Report, BAEC, Dhaka (1999-2001) page 166-167)

[5] ZULQUARNAIN, M.A., et al., "Management of Spent Fuel from Reactor of Bangladesh", (Proc. International Conference on Management of Spent Fuel from Nuclear Power Reactors, Vienna Austria 2006) IAEA-CN-144/9P, IAEA, Vienna

[6] JALIL, A., MONZURUL HAQUE, M., ZULQUARNAIN, M.A., RABBANI, G., "The Status of Storage of Spent Fuel from Reactors; Bangladesh Perspective", IAEA-CN-102/3P (2003) p.453-459. available at: http://www-pub.iaea.org/MTCD/publications/PDF/csp_020c/PDF/CSP-20_Part_3.pdf

[7] Nuclear Safety and Radiation Control Act, Bangladesh Gazette, Act No. 21 of 1993.

[8] Nuclear Safety and Radiation Control Rules, Bangladesh Gazette, Registered No DA-I, Deputy Controller, Bangladesh Forms and Publication office, Dhaka, Bangladesh (1997).

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Basic Safety Standard, Safety Series No. 115, IAEA, Vienna (1996).

[10] Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, GOV/INF/821-GC (41)/INF/12, IAEA, Vienna.

[11] Radioactive Waste and Spent Fuel Management Rules- 2005 (under approval).

National experience on return of research reactor spent fuel to the country of origin
Democratic Republic of Congo: some useful information for shipment operation

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Abstract. For more than four decades, the Democratic Republic of the Congo (DRC) operated two research reactors, a Triga-I (50 kW) and a Triga-II (1 MW), both located at the "Centre Régional d'Etudes Nucléaires de Kinshasa" (CREN-K) situated in the campus of the University of Kinshasa. The Triga-I reactor was definitely shutdown and partially decommissioned in 1970, and its premises converted to a spent fuel storage facility with 56 low enriched uranium (LEU) spent fuel elements stored in the reactor pool. On 24 March 1972, the Triga-II reactor reached its first criticality with a core loading capacity of 70 LEU fuel elements. The reactor was used for the purpose of training, research and isotope production until November 2003. Since then, it's in a stage of extended shutdown due, among others, to technical safety related problems. This notwithstanding, the fuel elements currently loaded in the reactor core added to the 9 spare fresh fuel elements available in the facility, all of LEU type, are available to operate the reactor for another 10 to 15 years.

This paper describes the overall situation of the CREN-K Triga I & II fuel. Also are presented information on available fresh and spent fuel, on in-site and off-site infrastructure and on national regulations for transport of radioactive materials, information considered essential for the future shipment of the 138 fuel elements available at CREN-K facilities.

1. Introduction

Just one year after the creation of the International Atomic Energy Agency in Vienna, in 1958, was held in Geneva, Switzerland, the first of a conference series on the peaceful utilization of atomic energy. In the course of this conference, the Congo bought the first Triga Mark I reactor exposed by the USA General Atomic, becoming therefore the first African country launching in the peaceful utilization of nuclear energy [1].

Later, in 1970, it was decided to provide the Congo with a Triga II, more powerful reactor.

This paper describes the overall situation of the CREN-K Triga I & II fuel as far as its repatriation to the country of origin is concerned. Information on available fresh and spent fuel, in site and off site infrastructure and on national regulations for transport of radioactive materials, considered essential for any shipment operation, are presented.

2. Present status of the CREN-K research reactor facilities

2.1. Triga Mark I

Triga Mark I was commissioned in 1959 with an initial power of 10 kW which was subsequently upgraded to 50 kW. The reactor was definitely shutdown 11 years later and partially decommissioned. Most of the supporting systems were removed and parts of the reactor, such as the reflector, the rotary specimen rack, graphite elements, were used in the second research reactor, while the 56 type

102 Triga standard fuel elements (Triga catalog) composing the core loading were stored in the Triga-I pool as spent fuel since 1970.

2.2. Triga Mark II

Triga Mark II became first critical in 1972 with an initial power of 50 kW which was upgraded in 1974 to 1 MWth in a steady state mode and 1 600 MW in pulsed mode. The reactor operated with these characteristics almost for three decades. Since November 2003, it was shutdown to allow the local team address some technical and managerial safety related problems which prevent a safe operation of the facility in conformance with international safety standards. The country is presently implementing the recommendations made by an INSARR mission conducted in the country in October 2004, with the main objective of improving the regulatory supervision and the operational safety of the CREN-K research center.

3. Operation statistics of CREN-K Triga reactors

Since the start up of the first Triga Mark I in 1959, the staff has accumulated 47 years of experience in the maintenance and operation of this type of reactor. The accumulated energy generation up to November 2003 in the two reactors is shown in Table 1.

TABLE 1. ENERGY GENERATION IN THE TWO TRIGA REACTORS AT CRENK FACILITY

Triga Reactor	Year critical	Steady sate/ Pulsing limit		Energy generation to 11/2003 (MWD)
		Steady state (MW)	Pulse (MW)	
Mark I	1959	0.05	-	3.86 decommissioned in 1970
Mark II	1972	1	1 600	38

4. Reactor dismantling and re-assembling experiences

In 1987, following an optical underwater inspection performed by four IAEA experts on the Triga-II reactor, it was decided to repair several corrosion spots observed on the bottom of the reactor vessel, their depth ranging from 1.5 to 2.5 mm. The repair operation consisting essentially in sealing these spots with the “RTV silicone rubber adhesive sealant, maker General Electric” and covering them with small aluminium pastilles was entirely conducted by local staff. The reactor was totally dismantled to permit a free access to the corroded areas, re-assembled after the repair was completed and checked for safe operation. All the process was conducted in co-operation with experts from CEN-SCK/Mol, Belgium and General Atomics, USA.

During all process, tasks preparatory for the shipment of fuel to the country of origin have been performed, such as:

- fabrication of auxiliary tank for the storage of the control rods and some highly irradiated parts of the reactor internals;
- fabrication of the transfer cask;
- fabrication of aluminium racks for the storage of irradiated fuel in the storage wells;
- acquisition of emptying pumps;
- transfer of irradiated fuel from the reactor core to the storage wells;
- dismantling of reactor fixed parts;…

5. Nuclear material description

The fuel of the two CREN-K Triga reactors is a mixture of an alloy fissionable material (uranium) with a moderator that is a combination of hydrogen and zirconium (ZrH), leading to “fuel-zirconium-hydrogen” fuel-moderator elements.

5.1. Fresh fuel

5.1.1. Description, inventory and storage facilities

Fresh fuel elements available at CREN-K facilities are of LEU type, 19.9% ^{235}U enrichment. Referring to Triga catalog, 8 are of type 104 and one type 304 Fuelled Follower Control Rod (FFCR). Full description of the fuel is given in Table 2.

All the nine fresh fuel elements are currently stored in the fresh fuel storage area, consisting of a room adjacent to the Triga-I spent fuel pool. Elements are laid down in vertical metallic racks as can be seen in Fig. 1. Storage is dry.

TABLE 2. CHARACTERISTICS OF TRIGA I & II FUEL ELEMENTS

Characteristics	Element type			
	102	104	204	304
Physical form	Metal rod	Metal rod	Metal rod	Metal rod
Chemical form (composition of fuel)	U – Zr H Metal alloy	U – Zr H Metal alloy	U – Zr H Metal alloy	U – Zr H Metal alloy
Nuclear material	$^{235}U + ^{238}U$	$^{235}U + ^{238}U$	$^{235}U + ^{238}U$	$^{235}U + ^{238}U$
Quantity (g)		≈ 193	≈ 192	≈ 160
Fissionable material	^{235}U	^{235}U	^{235}U	^{235}U
Content U, wt %	8	8.5	8.5	8.5
Enrichment (%)	20	19.9	19.9	19.9
Quantity (g)	37	≈ 38	≈ 32	≈ 32
Geometric form	Cylindrical rod	Cylindrical rod	Cylindrical rod	Cylindrical rod
Dimension				
Total length (cm)	72.5	72.08	114.93	114.30
Total external ϕ (cm)	3.56	3.76	3.76	3.43
External ϕ of U (cm)		3.63	3.63	3.33
Length of U (cm)	38	38.1	38.1	38.1
Ratio H / Zr	1.1	1.6	1.6	1.6
Composition of alloy	8 wt % U, 90.9wt% Zr, 1.1 wt % H, 0 % Erbium	8.5 wt % U, 90 wt % Zr, 1.5 wt % H, 0 % Erbium		
Cladding material				
Thickness (cm)	0.07	0.05	0.05	0.05
Composition	Al	SS – 304 type GGA	SS – 304 type GGA	SS – 304 type GGA

5.1.2. Fresh fuel handling capability

Fresh fuel is manually handled and no special equipment is required to unload it from its storage area. A fuel handling tool is used to lower the fresh fuel into the reactor core.

5.2. Spent fuel

5.2.1. Description, inventory and storage facilities

As for the fresh fuel, spent fuel elements are also described in Table 1. All are of LEU type, 19.9% enrichment, and types 102, 104 and 304.

Spent fuel elements are stored in three different locations.

FIG. 1. Fresh fuel storage area.

As mentioned previously, 56 type 102 elements from the first Triga I reactor are stored in this reactor's pool, in vertical racks where they are positioned in vertical rows.

Two damaged type 104 elements from Triga-II are stored in one of the 4 spent fuel storage wells located at the Triga II hall. Each of these wells may contain one cylindrical rack with 19 vertical fuel storage positions arranged in 3 concentric rows.

One type 204 Triga fuel temperature thermocouple instrumented spent fuel is in a rack in the Triga II reactor vessel since 1985.

Apart from the above, there are 70 other LEU irradiated fuel elements composing the core loading of Triga-II. They are of types 104, 204 and 304. Irradiated fuel elements are temporarily stored in racks in the Triga-II pool and usually returned to the core. In principle, in the event an element is completely burned or an irradiated element is damaged (case of the two elements mentioned above) the element is placed temporarily in one of the two spent fuel storage locations inside of the Triga-II building before its return to the vendor. So far, neither irradiated fuel nor spent fuel have been returned to the vendor.

Table 3 contains a summary of the inventory of nuclear material at CREN-K facilities.

TABLE 3. SUMMARY OF NUCLEAR MATERIAL INVENTORY

Area	Item description	Number of items
1. Fresh fuel storage	Standard fuel elements	8
	Control rod (FFCR)	1
2. Triga MK II reactor core	Standard fuel elements (type 104 GGA)	66
	Control rods (FFCR, type 204 GGA)	3
	Instrumented element (with thermocouple, type 304 GGA)	1
3. Spent fuel storage		
a. Wells in Triga II building	Irradiated fuel element	2
b. Triga I pool	Irradiated fuel element	56
c. Triga II pool	Instrumented element	1
4. Other items		
a. Triga building	Fission chamber	1
b. Triga II pool	Fission chamber	2

5.2.2. Spent fuel handling

For the transfer of irradiated or spent fuel inside the Triga-II vessel (core to rack or vice versa), a special fuel handling tool is used. In case of transfer out of the Triga-II pool or out of the spent fuel storage wells but inside the Triga-II building, a locally made transfer cask weighting 1.5 tonnes is available. The cask can host only one fuel element at a time. A 5-tonne bridge crane is used to move the transfer cask within the reactor hall.

In Triga-I hall, the same fuel handling tool is used to move fuel from one position to another in the spent fuel storage pool. A 2-tonne bridge crane is also available in the Triga-I hall.

There is no vehicle transport capacity of spent fuel in the all facility. As for now, it's not possible to move spent fuel from Triga II building to Triga I pool and vice-versa due to lack of appropriate equipment. More, there is neither irradiated fuel transport equipment nor shipping cask available in the site.

5.2.3. Cooling time for the spent fuel

The cooling time for CREN-K spent fuel elements is over 21 years as it can be seen from Table 4 below.

TABLE 4. COOLING TIME FOR CREN-K SPENT FUEL

Spent fuel	Status	Storage location	Period	Cooling time
6896, 6899	Damaged	Rack pool Triga-II Well Triga-II hall	March 76-May 89 May 89 – August 2006	> 30 years
6822 TC	Defected thermocouples	Rack pool Triga-II	Dec. 85 – August 2006	> 21 years
Fuel from dismantled Triga-I.	-	Rack, pool Triga-I	Dec.70 – August 2006	> 36 years

6. Preparing the future for spent fuel repatriation

With the promulgation of the national nuclear law no 017/2002 [2] in 2002, and the next future implementation of the national regulatory body, the reactor dismantling as well as the transport of spent fuel to the country of origin, and other related operations, will be submitted to more stringent constraints. To afford them, quality assurance programmes (QA) should be developed since now to plan and implement corresponding operations so as to fulfil related international national regulatory body [3,4].

QA will assist the managers to have a better understanding and control of operations, and thus more easily prevent degraded decommissioning, storage, packing and transport safety and non-compliance. This assistance will consist in:

- Early detection of the defect
- Review of the appropriate aspects of existing practices and of various stages of operations from project design to the effective return operations, to see what change, if any, are necessary to identify and prevent similar occurrences; and
- Implement any changes necessary in a controlled and recorded manner.

The stages of such a QA will include:

- IAEA and national statutory requirements related to the reactor decommissioning and spent fuel transport operations.
- Institutional SWOT (Strengths, Weaknesses, Opportunities and Threats-) analysis referring to the recovery of existing written procedures related to the reactor dismantling and waste management, and relating expert mission reports.
- Modification / updating of these procedures.
- Completion with additional reactor dismantling procedures.
- Generate appropriate transport and transport-related procedures.

7. Conclusion

So far, neither irradiated nor spent fuel have been returned to the vendor. Nevertheless, some important experiences in prerequisite operations related to transport of spent fuel have been accumulated by the CRENK staff, in particular the experience of totally dismantling and re-assembling Triga Mark II internals. To fulfil with international and national requirements in the matter, QA programmes related to the return of Democratic Republic of Congo spent fuel to the country of origin should be considered since now in the course of the development of the CGEA quality management programme.

REFERENCES

[1] MALU WA KALENGA, Historique et acquis de la recherche au Zaïre, Revue Zaïroise des Sciences Nucléaires, VI(1985)1.

[2] JOURNAL OFFICIEL, Loi n° 017/2002 du 16 octobre 2002 portant dispositions relatives à la protection contre les dangers des rayonnements ionisants et à la protection physique des matières et des installations nucléaires.

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Quality assurance in decommissioning, Safety Series No 50-C/SG-Q, Q.14, IAEA, Vienna (1996).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Quality assurance for safe transport of radioactive material, Safety Series No 113, IAEA, Vienna (1994).

The Jamaican Slowpoke HEU-LEU core conversion

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Abstract. The HEU core of the Jamaican SLOWPOKE research reactor is scheduled for conversion to LEU. The actual conversion process will most likely be contracted to Atomic Energy of Canada Limited (AECL). Preliminary calculations have indicated that the total activity of used HEU core in Jamaica (~8 TBq) should be about half that of the Montreal used HEU core. There is sufficient infrastructure both onsite and offsite to maneuver the loaded transportation flask to the shipping vessel. Appropriate licenses for the importation of the new fuel and exportation of the used fuel will be applied for once a provisional timetable has been established.

1. Introduction

The process to convert the core of the SLOWPOKE reactor in Jamaica is in agrrement with the spirit of the the Global Threat Reduction Initiative (GTRI) and Reduced Enrichment for Research and Test Reactors (RERTR) programs. Conversion to LEU for SLOWPOKE actually provides a superior reactor, with both operating time and core life-time significantly increased.

The actual core conversion for the SLOWPOKE reactor in Jamaica must be contracted to AECL who have proprietary rights and experience in the LEU fuel development and core conversion process for SLOWPOKE research reactors.

2. HEU reactor core

The detailed specifications of the reactor are available [1]. The reactor core is illustrated in Fig. 1. It consists of an assembly of 296 fuel pins containing a total of 817 g of 93% enriched ^{235}U as co-extruded alloy containing 28% by weight of U in Al. A 100 mm thick pure beryllium annulus encases the fuel cage, which is a cylinder of size 22.8 cm by 22 cm. The annulus acts as a side reflector for neutrons and a 50 mm thick beryllium disc forms the bottom reflector. The top reflectors, known as shims, consist of semi circular plates of beryllium each only a few millimeters thick. Since no adjustments to the core are allowed, burnup is corrected for by the increased neutron reflection provided by adding shims as required. The core assembly is immersed in an aluminum tank containing very pure water (deionized weekly to a resistivity of 4 x 10^7 ohm cm) which is both moderator and heat transfer medium. The tank is suspended in a pool 6.4 m deep containing water that is continuously deionized to a resistivity of 10^6 ohm cm. This provides for both heat transfer from the core water and for biological shielding.

There are five small inner irradiation sites within the beryllium annulus and four large sites outside of the annulus. Additional irradiation sites are provided by the in-pool irradiation carousel, which is position on the surface of the reactor vessel adjacent to the core, Fig. 1.

The design and operating conditions of SLOWPOKE eliminate the need for the conventional complex instrumentation and electromechanical emergency shutdown systems. This high degree of intrinsic safety is achieved by a large negative temperature coefficient and by severe limitations on both the excess reactivity (maximum 0.40%) and the operating conditions. The power level is controlled by a single cadmium control rod via a feed back to a neutron detector located within the beryllium annulus.

The neutron flux is measured by a Reuter-Stokes self-powered flux detector with a nominal sensitivity of 1 x10^{-20} amps per unit flux.

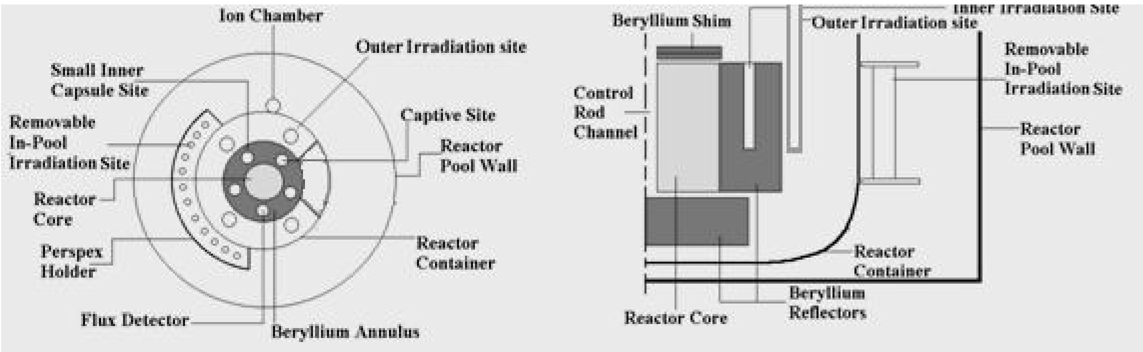

FIG. 1. Schematics of the Slowpoke 2 reactor core showing in-core and in-pool irradiation sites.

3. LEU fuel composition

The previously developed LEU fuel was fabricated from zircaloy-4 clad uranium oxide pellets and contained 1100 g of ^{235}U (total mass of U ~5 600g) at an enrichment of 19.9%. The core itself was 22 cm in diameter and 22.7 cm in height. At criticality there were a total of 198 fuel pins in the fuel cage, each pin was 5.26 mm in diameter and 234 mm in length. At present, AECL are in the process of defining the requirements to re-qualify the fuel production process.

A side by side comparison shows that the fuel pins are physically very similar, as shown on Table 1.

TABLE 1. COMPARISON OF THE HEU-FUELLED AND THE LEU-FUELLED REACTOR CORES

	HEU-fuelled	LEU-fuelled
core diameter	220 mm	220 mm
core height	228 mm	234 mm
number of fuel pins	296	198
Fuel pin diameter, with cladding	5.23 mm	5.26 mm
Fuel length	225 mm	234 mm
cladding	Aluminum	Zircaloy-4
Fuel	U-Al 28% alloy	UO_2
total mass of uranium	0.9 kg	5.6 kg
enrichment U-235	93%	19.89%
total mass of U-235	0.82 kg	1.12 kg
volume of water in core	7.8 L	8.1 L

This similarity simplifies the core conversion as the beryllium annulus and other auxiliary systems can be reused.

The large negative temperature coefficients of both the HEU and LEU cores ensure that power excursions are self-limiting[1], however the characteristics of the temperature coefficients differ greatly, as shown in Fig. 2.

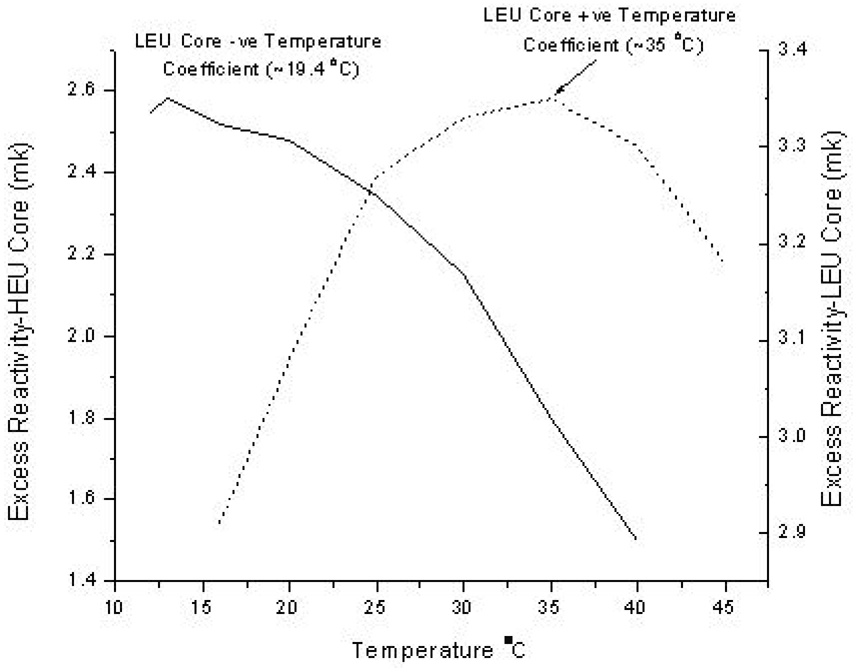

FIG. 2. Temperature coefficients for HEU & LEU cores.

The difference in the temperature coefficient has resulted in improved performance of the LEU SLOWPOKE core, notably in the reactor runtime; this is particularly true for the Jamaican situation where higher ambient temperatures further reduced our operating time of the HEU core, as demonstrated on Table 2.

TABLE 2. PERFORMANCE COMPARISON OF HEU & LEU CORES

Parameters	HEU	LEU
Maximum operating power	20 kW	20 kW
Maximum operating time at 3 mk	6 hrs	12 hrs
Maximum operating time at 4 mk	16 hrs (13 hrs*)	24 hrs
Operating* range between shim additions	2.5 – 4.0 mk	1.5 – 4.0 mk
Core Life-Time	20 Years	40 Years

* Operating time in Jamaica

4. Fuel burnup

As SLOWPOKE reactors are built with a lifetime core, there is no need for on-site spent nuclcear fuel storage. As shown on Fig. 3, the addition of the next berillyum shim is predicted for February 2008. At current rate of usage the current core configuration will last another 17 years, at which time an additional beryllium annulus can be added giving a further 15 years. Figure 4 shows the power already generated by the reactor.

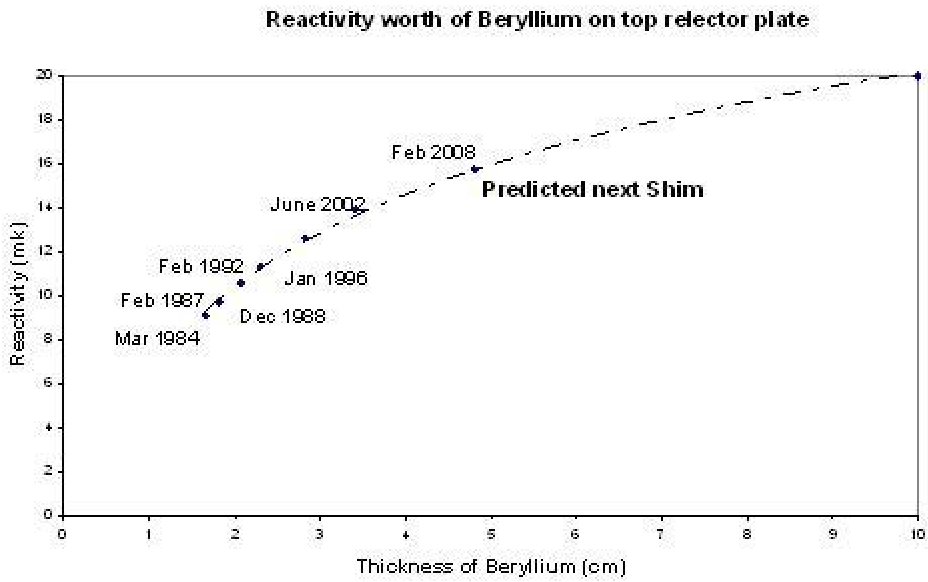

FIG. 3. History of Shim adjustments.

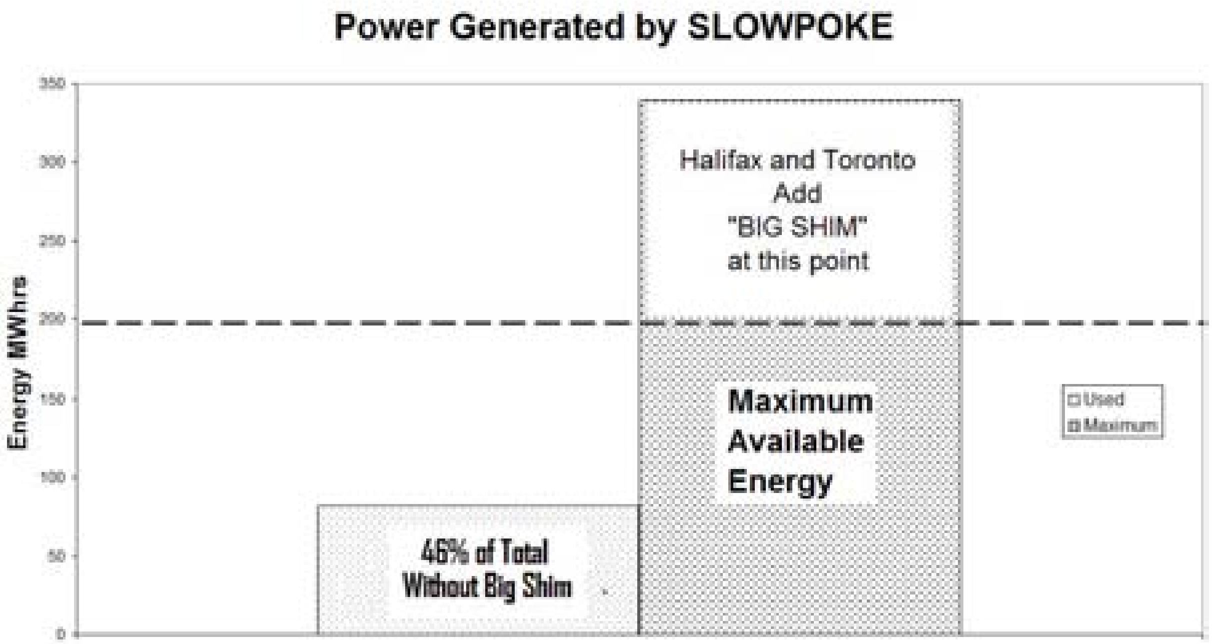

FIG. 4. Power generated by SLOWPOKE.

5. Fission product activity

The fission product activity of the used fuel, which is a function of the reactor flux hours, will be estimated based on an AECL calculation. The calculation is based on a SLOWPOKE reactor which was operated for 5 years at a neutron flux of 1×10^{11} n.cm-^{2}s^{-1} (2kW) and then 10 hours at a neutron Flux of 1×10^{12} n.cm-^{2}s^{-1} (20 kW) [2]. The calculated activity, 30 days after shutdown, was 23 TBq. The average flux over the last 5 years (8766 hours per year) for the Jamaican SLOWPOKE is approximately 0.37×10^{11} n.cm-^{2}s^{-1} (0.69 kW). Based on our average flux over the last 5 years the expected activity of core, 30 days after shutdown, will be approximately 8 TBq. Previous experience (Montreal) has shown this calculation to be reasonably accurate (~18 TBq) [3] and that a one month cooling period is sufficient before the conversion process takes place. It is therefore our intention to shutdown the reactor 6 weeks before conversion. The reactor water and auxiliary systems will continue to be maintained as per standard operating procedures outlined in AECL document CPR-26.

6. Radiation protection

The International Centre for Environmental and Nuclear Sciences provides radiation monitoring services and is responsible for monitoring all radiation workers in Jamaica, Barbados and the Turks and Caicos Islands, as such; we are well equipped to provide all radiation monitoring services during the conversion process. Additional consultation will be provided from the Government Health Physics Department.

7. Core replacement

In all likelihood the core will be removed in accordance with procedures developed for the Montreal research reactor. The moving of the F 257 transportation flask will be contracted to a local haulage company with experience in moving heavy equipment. A block and tackle attached to the I-beam (load capacity 5 500 lbs) above the reactor pool will be used to maneuver the ~ 4 000 lb F257 transport flask in and out of the pool. Shown below in Fig. 5. are the forces as calculated for the Montreal core conversion [3], as the cores are identical and the same transportation flask is to be used the values remain valid.

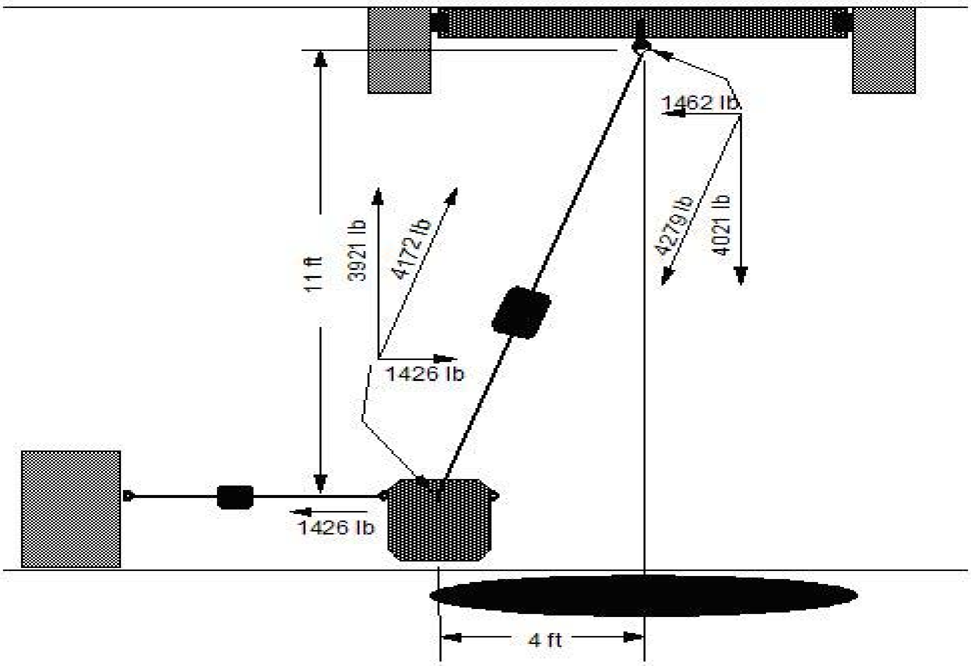

FIG. 5. Block and tackle assemblies for moving the F257 transportation flask.

There is sufficient space and floor load capacity, 4 500 lbs per square inch, to move the transport flask through the building using a six wheel hydraulic cart (load capacity ~8 000 lb), each of the 4 doorways to be passed through are 1.6 m by 2.1 m and are large enough to allow the loaded hydraulic cart to pass through.

The Proposed route through building is shown below in Fig. 6.

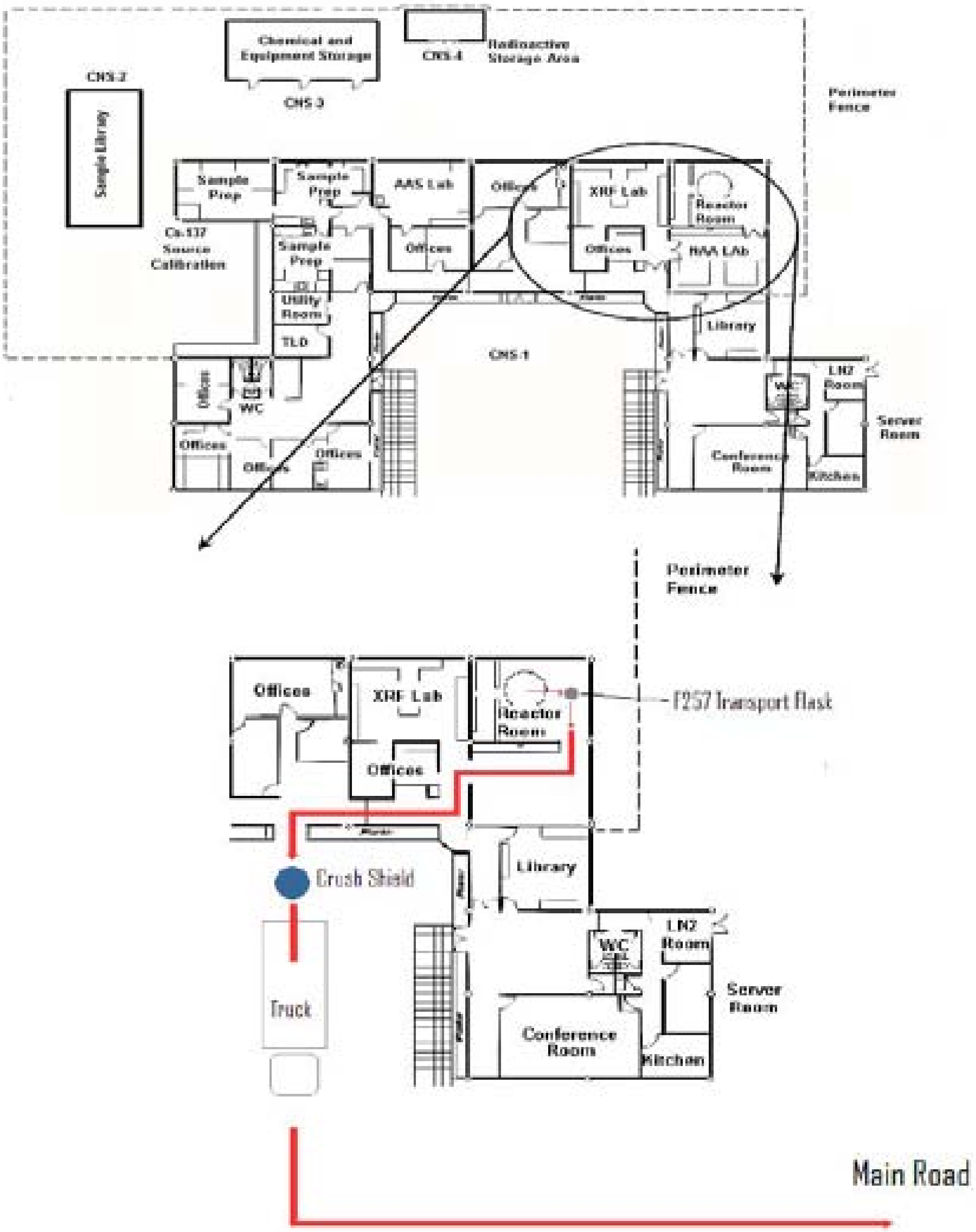

FIG. 6. Proposed route for the movement of the F257 transportation flask.

There are no foreseen obstacles to maneuver the load F257 flask to the entrance of the building. The fire/crush shield will be installed at the entrance of the building and a fork lift used to place the flask and shield on the truck for transportation to the dock for shipment. The truck containing the core will be given armed escort to the wharf; armed guards will remain with container until its loading onto shipping vessel. In the event that temporary storage of the loaded transportation flask is required, a secured radiation storage room (within ICENS perimeter fence) is available, Fig. 7.

FIG. 7. ICENS radiation storage room.

Due to the small size of the core there are no foreseen obstacles en route to the wharf and there will be no need for specialized lifting equipment once there.

8. Regulatory approval

At present licenses for the use of, importation and exportation of radioactive materials are granted through Ministry of Health Pharmaceutical Division. Once a provisional timetable for the conversion has been established appropriate licenses will be applied for from the Ministry of Health.

9. Conclusion

The core conversion, with funding provided by the DOE, in all likelihood will be contracted to AECL due to contractual obligations and their previous experience in the fabrication of the fuel and conversion process for SLOWPOKE research reactors. It is envisaged that the process from shutdown to commissioning can be completed in a six week window. Presently we see no major legal or physical obstacles that could hinder the conversion process. A firm timetable for the conversion cannot be established until the LEU fuel fabrication process has been re-qualified.

REFERENCES

[1] KAY, R.E., STEVENS-GUILLE, P.D., HILBOURNE, J.W., JERVIS, R.E., SLOWPOKE: A new low cost laboratory reactor, Inter. J. Appl. Radiation and Isotopes **24** (1973) 509.

[2] KENNEDY, G., Activity of the used Core and Beryllium Reflectors after Shutdown of a SLOWPOKE Reactor, Ecole Polytechnique de Montreal, Private Communication (2003).

[3] KENNEDY, G., Removal of Irradiated Fuel from Ecole Polytechnique SLOWPOKE Reactor, Private Communication (2006).

Present situation of unused uranium fuel in Tokyo Institute of Technology

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Abstract. Present situation of unused enriched uranium fuel in Tokyo Institute of Technology is described. The fuels were for sub-critical experiments. There is no special facility for transportation in the site. But there is no technical problem for it. One of the important issues to be done is a duty by national regulation against nuclear disaster.

1. Introduction

The predecessor of Research Laboratory for Nuclear Reactors (RLNR) in Tokyo Institute of Technology was founded in 1956, only 18 years after the discovery of nuclear fission phenomena. In 1957, Department of Nuclear Engineering in Graduate School of Science and Engineering was established. It was the time when the research of nuclear energy was resumed in Japan after the war and great efforts were made to catch up the cutting-edge of the nuclear energy engineering in the world. At that time RLNR aimed to build a small nuclear reactor. However this plan was replaced with construction of a subcriticality experiment facility because RLNR was located in a residential area of Tokyo. In 1961, Fission Experiment Facility was constructed. A subcritical experiment facility was installed using 19.86% enriched uranium in light water system, which was named TITAN. The construction started in 1961 and completed in 1962. Various important experiments were performed using the low enriched uranium in sub-critical condition by the TITAN. Most of the experiments have finished by 1970s and the TITAN was officially closed in 2000.

2. TITAN subcritical experiment facility[1]

The room installing TITAN has an area of 240 m^2 (16 m x 15 m) and height of 14m. With regard to the radiation shielding, special considerations were not taken for wall structure of the room, because TITAN itself had a shielding thick enough around it, about 150 cm of ordinary concrete. Figure 1 shows the arrangement of the room. At the central part of it, TITAN was constructed. A large dump tank, ion beds for purifying feed water, a heat exchanger, a heater, pumps, valves and all additonal items were installed in a corner of the room. Two other pits for waste water (12 m^3 capacity, each) were constructed at the corner [1].

Figures 2 and 3 show the horizontal and vertical cross sections of the TITAN, respectively. This assembly has three components: (1) a sub-critical 19.86% enriched uranium-light water system, (2) a graphite assembly, and (3) a large water tank for the shield experiments.

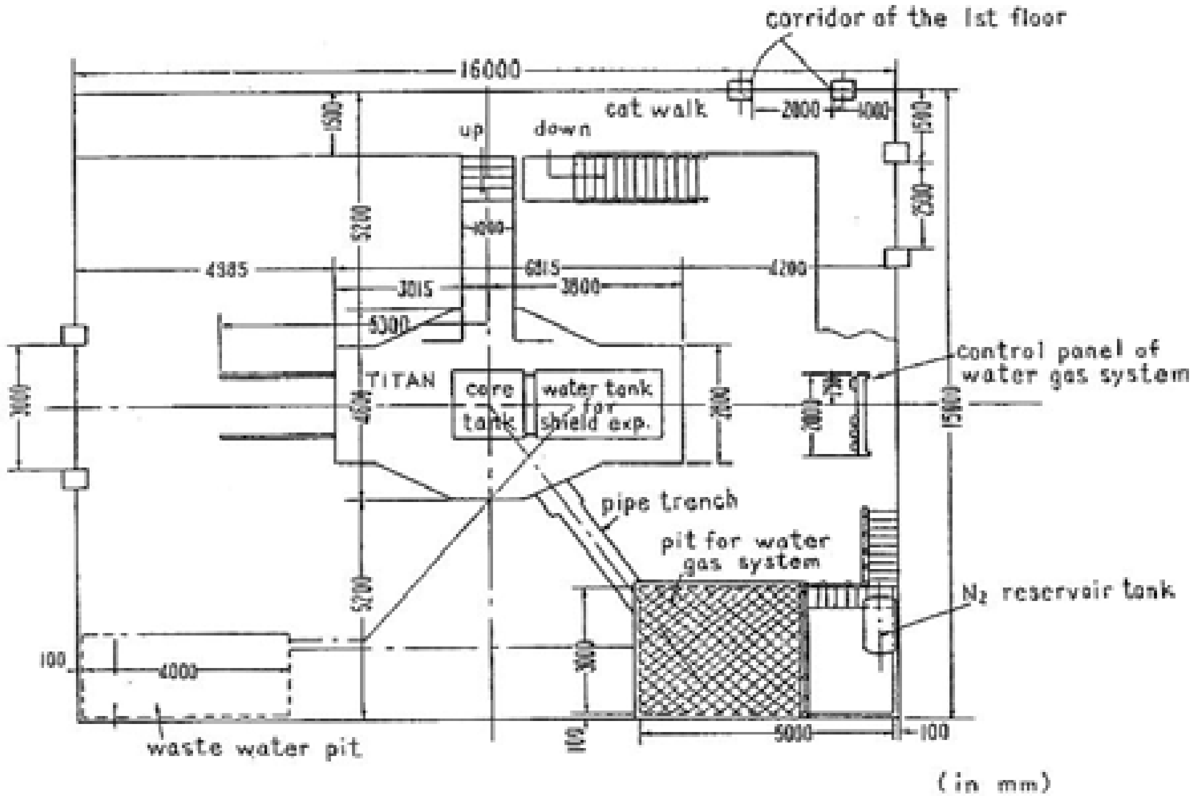
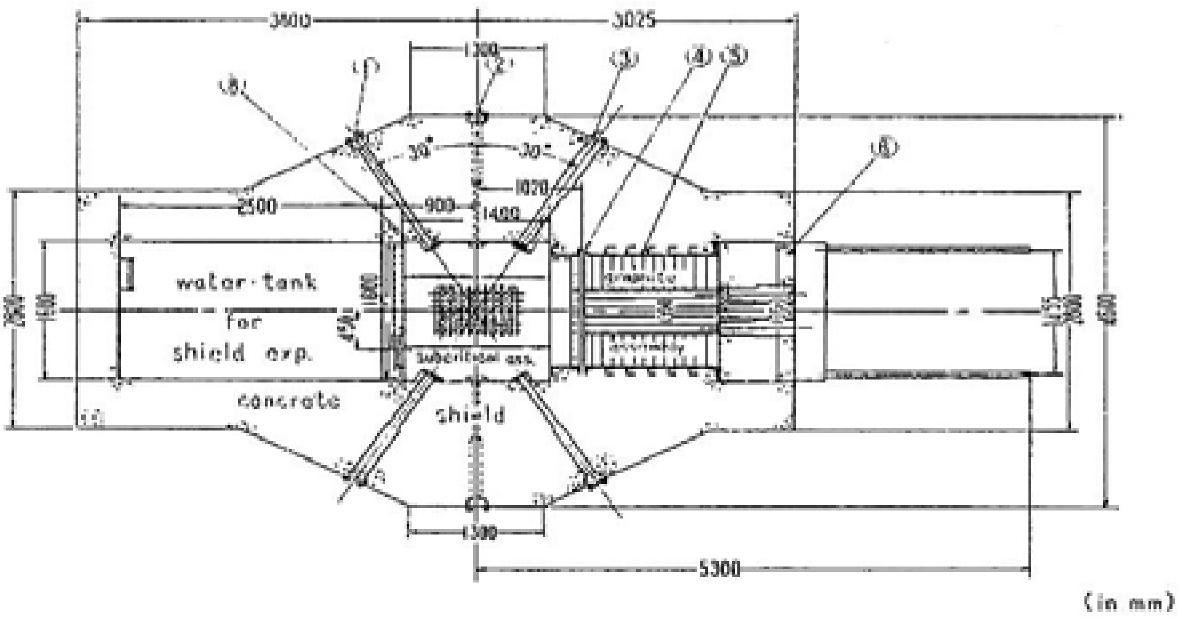


FIG. 1. Arrangement of TITAN facility [1].



① peeping hole ② peeping hole
③ 〃 ④ Cadmium shutter
⑤ Graphite assemble ⑥ heavy concrete shield
⑦ neutron source ⑧ main tank

FIG. 2. Horizontal cross section of TITAN sub-critical assembly [1].

The sub-critical system consisted of MTR-type fuel plates and light water. The fuel plates were 19.86% enriched uranium-aluminium alloy meat covered by 2S-aluminium, and pure water was used as moderator as well as reflector. This sub-critical system was used for the following experiments:

1. Critical mass estimation on the various core configurations.

2. Measurements of various reactivity coefficients: temperature effect, void effect, and absorber effect.

3. Measurement of the neutron spectra at the various core positions.

Even though this system is subcritical, it has shim-safety rods system and moderator dump system as safety devices. Table 1 shows the summary of the main items of TITAN subcritical assembly.

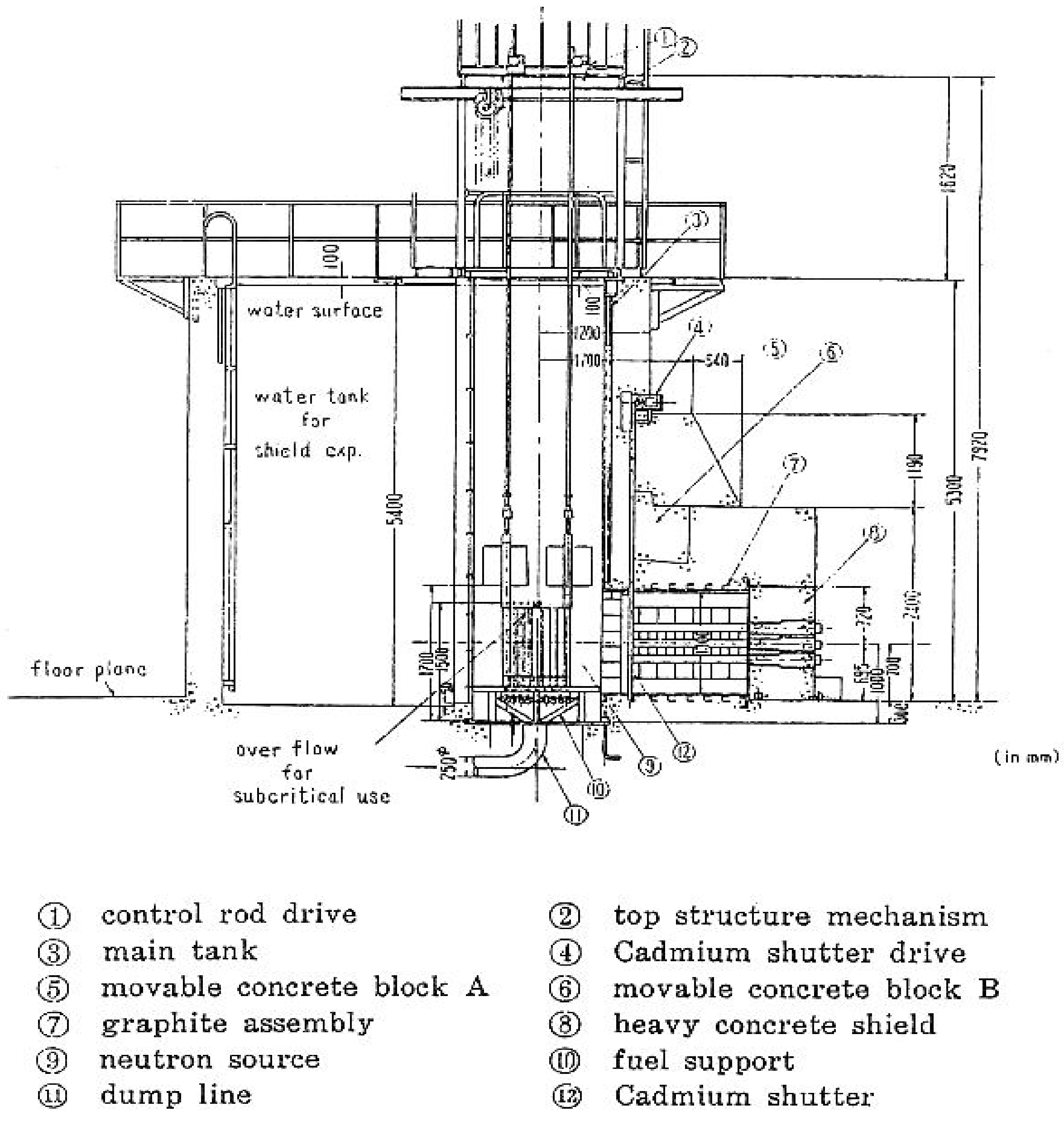


FIG. 3. Vertical cross section of TITAN sub-critical assembly [1].

TABLE 1. MAIN ITEMS OF SUB-CRITICAL ASSEMBLY OF TITAN EXPERIMENT FACILITY

Item	Specification
Fuel	Meat: 19.86% Enriched U-Al allory Cladding: 2S Al Size of fuel plate: 800mm in length, 69.0mm in width, 1.5mm in thickness Size of fuel meat: 610mm in length, 58mm in width, 0.70mm in thickness U-235 content in a fuel plate: 8.3g
Moderator and Reflector	Purified Light water Temperature: Room temperature (Can be raised up to 80º C)
Absorber Rods	2 Cd rods and 2 Boron-steel rods
Dump System	Diameter of dump valve and line: 250mm I.D.
Sparge System	Sparge of N_2 gas Reduction of moderator density: 15%
Operation Condition	Sub-critical

3. Fuels and core of the TITAN subcritical assembly

The core tank of the TITAN sub-critical assembly was a rectangular, open top type as shown in Fig. 4. The fuel element support plate and grid plate fixed on the base plate of the core tank. The fuel elements are inserted on the grid plate, as shown in Fig. 5. The detail of the fuel element is shown in Fig. 6. The fuel elements were made of 52S aluminum, and had the size of 76.2 mm x 76.2 mm and 1 060 mm in length. Desired number of the fuel plates up to 19 could be inserted in them through small ditches engraved vertically in the side plate of the fuel box with pitch of 1.8mm. The fuel plates are flat and made by covering U-Al alloy meat with 2S-aluminum, and have the dimensions of 1.5mm thick, 69.0 mm wide and 800 mm long. Each plate contains about 8.3 g of U-235 in 19.86% enriched form. Total number of fuel plates is 300.

4. In site infra-structure

The fuels are stored within a dry cabinet in a storage. The fuels can be transported using two casks. The space for loading is rather small and there is no crane. But the fuel assembly with the fuel is not heavy and the radiation dose is quite low, so it is possible to do the loading of fuel assemblies to a cask by hands. The cask can be transported using a special cart. The floor has no problem for the transportation of the casks. There is no interference for the transportation and there is no need for cropping. The fuels were used only for sub-critical experiments, so there is little radiation from the fuels, which means that there is no need of special shielding for radiation from the fuel.

Two casks have been kept for transportation of the fuel elements. For the transportation of the fuels abroad, the licenses both domestic and from the foreign country are needed. The domestic license has been issued already.

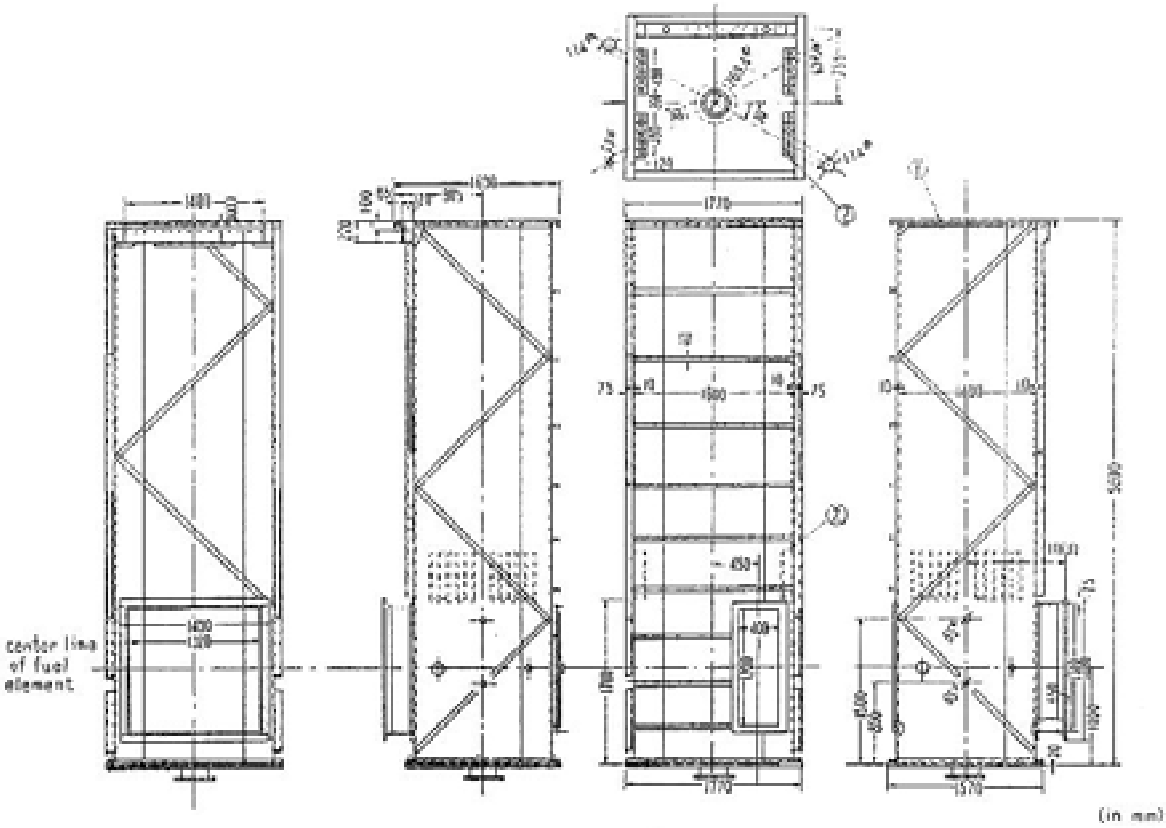


FIG. 4. Core tank of TITAN sub-critical assembly [1].

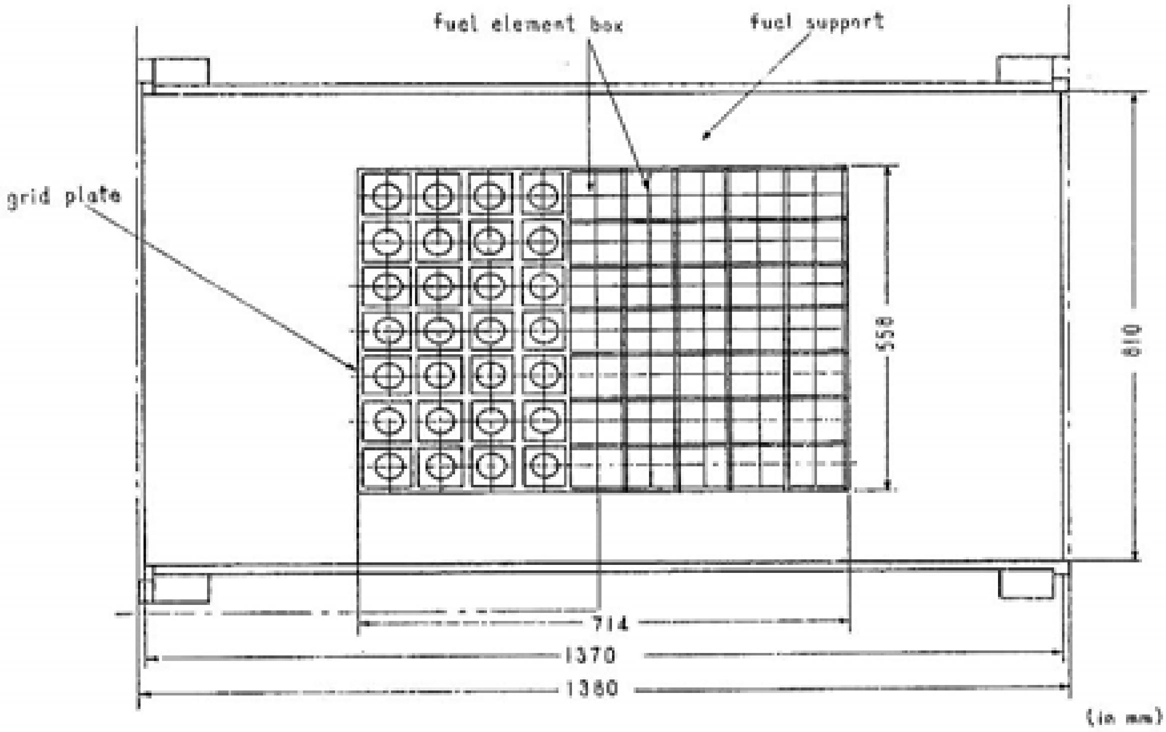


FIG. 5. Arrangement of Fuel Elements in Core Tank [1].

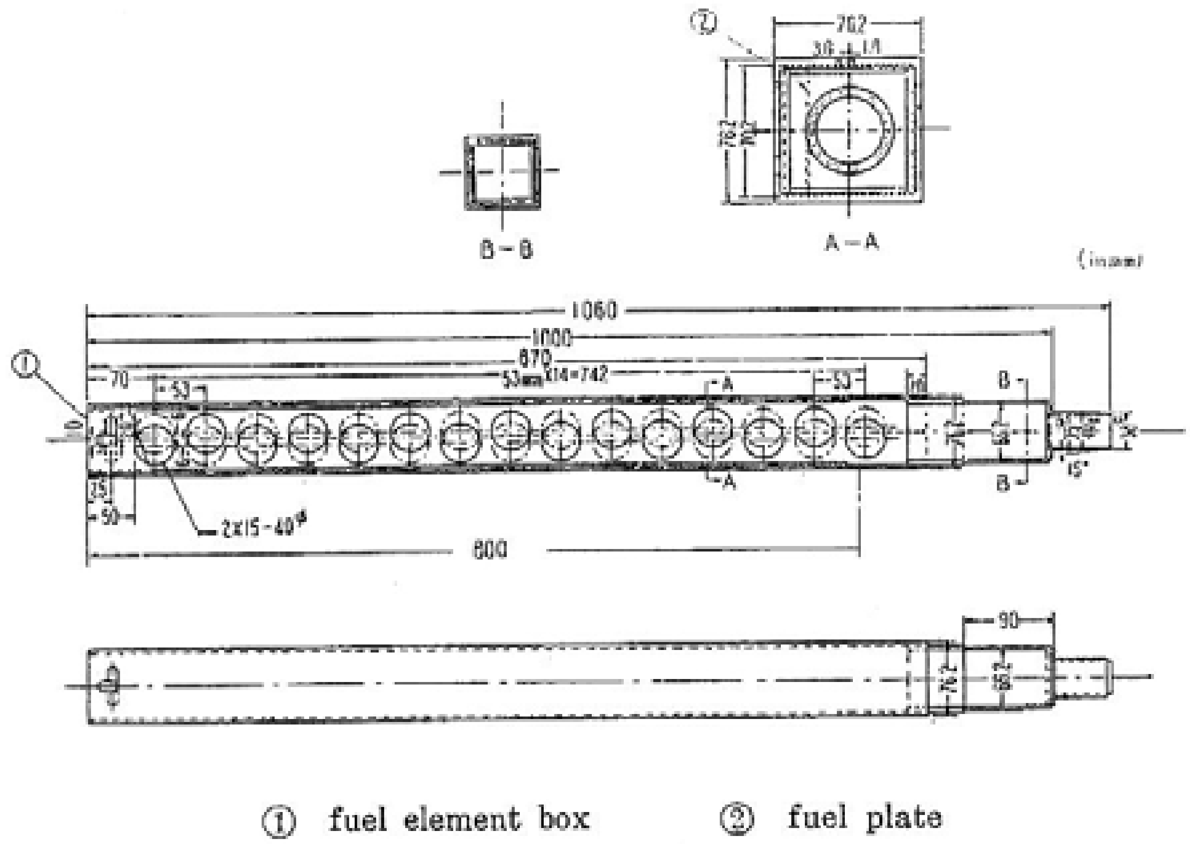


FIG. 6. Fuel element for TITAN sub-critical assembly [1].

5. National regulation

After the criticality accident occurred in the JCO Co nuclear fuel plant, in 1999, Japanese regulation became quite strict for nuclear facilities. The fuels are not exception. It has been required to be in very strict management. Because of the low activity, the duty of nuclear facility is not only related to the safety from radiation, criticality safety, and physical protection, but it also involves the system and equipments against nuclear disaster. In this case, while in the facility, the duty related to the nuclear disaster is exempted from the government, because the nuclear fuels are not used and are kept in storage in a stable condition. Essentially there is no possibility that a nuclear disaster occurs. For example the possibility to have a criticality accident, involving the nuclear fuels is not feasible, because of the low enrichment and because of the little amount of fuel plates. In case of transportation of the fuel plates, there is no technical problem essentially, because they have never been irradiated in a nuclear reactor and the amount of Uranium is very small. However for transportation the duty related to the nuclear disaster becomes one of the most difficult issues. In the transport of the fuels, the exemption of the duty related to nuclear disaster is canceled, even if there is no possibility that a nuclear disaster happens by the carrying out procedure of the nuclear fuels during the transportation. So it is necessary to establish the system and equipments for the duty before the transportation. The duty also requests detail discussions with local government about a plan against nuclear disaster.

6. Summary

About 300 low enriched uranium fuel plates are kept in good condition in the storage of the Tokyo Institute of Technology. They were for TITAN sub-critical assembly and have never been loaded in a nuclear reactor. There is no special in site infrastructure for the transportation. But basically there is no major technical problem for their transport because of little burn up and radiation from the fuels. Exemption of the duty related to nuclear disaster is canceled in transportation of the fuels. As a consequence it is needed to postulate a nuclear disaster involving the fuels, and to establish a system against it, to have equipments prepared, and to have discussions with local government about a plan to mitigate it.

REFERENCE

[1] OYAMA, Y., et al., "TITAN- A Nuclear Assembly of Tokyo Institute of Technology", Bulletin of the Tokyo Institute of Technology, No.49 (1962) 1-18.

Consideration factors on the spent fuel shipment for PUSPATI TRIGA Reactor (RTP)

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Abstract. Malaysian Institute for Nuclear Technology Research (MINT) operates a 1MW TRIGA MARK II type research reactor since 1982. The PUSPATI TRIGA Reactor (RTP) reached its first criticality on 23 June 1982 and since then, it has been used for beam experiments, neutron activation analysis, radioisotopes production, education and training. RTP uses three types of fuel elements, namely, 8.5 wt%, 12wt% and 20 wt%. For all the three type the enrichment level of U-235 is 20%. Until the end of 2005, RTP has accumulated 21 906 hrs of operation time, and 13 166 MWhrs of burnup. Based on the neutronics calculation, all the fuel elements are expected to be fully utilized by the year 2015. At present, there is no decision for the government to take part in return of the spent nuclear fuel back to the country of origin, where it was enriched. This paper describes the current status of the fuel elements and the availability of local infrastructure, considering the eventual agreement of the government to join the US Foreing Research Reactor Spent Nuclear Fuel Acceptance Programme for the shipment of the spent nuclear fuels. The involvement of national regulatory body is also briefly described.

1. Introduction

The PUSPATI TRIGA research reactor achieved its first criticality on 28 June 1982. It is a light water moderated and pool type research reactor with a maximum steady state power of 1 MW and a pulsing capability of 1 300 MW. Until the end of 2005, RTP has accumulated a burn-up of 13 166 MWhrs, and 21 906 hrs of operation time. The reactor has been used mainly for neutron activation analysis, isotope production, beam experiments, education,and training of human resources. Presently, the reactor operation is generally geared toward the neutron activation analysis activity.

2. Fuel description

The reactor uses standard TRIGA-type fuel elements with uranium content of 8.5, 12 and 20 weight percent, all they enriched to 20% in uranium-235. The general dimensions of the fuel are shown in Table 1.

TABLE 1. FUEL ELEMENT DIMENSIONS

	Value
Overall element length	752 mm
Fuel length	381 mm
Diameter	38 mm
Cladding material	304 SS
Cladding Material	.5mm

The core is composed of 7 rings, named A, B, C, D, E, F and G. Ring A is central and has only one position. Rings B through G have 6,12,18,24,30 and 36 positions, respectively. In total there are 127 locations in the core which can be filled either with fuel elements or other components like control rods, irradiation channels, etc. At present there are 114 fuel elements placed in the core. During the past 24 years only six fuel elements had been removed from the core. They are stored under water on the storage racks available in the reactor pool. Figure 1 shows the current core configuration.

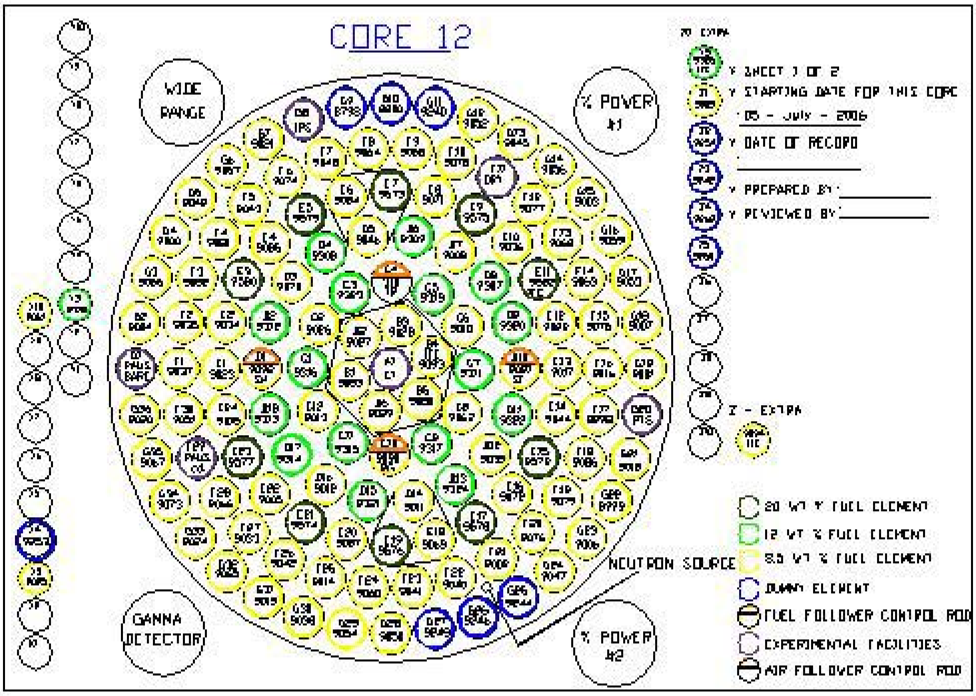

FIG. 1. Core configuration.

Since the commissioning of the reactor, the core of RTP has undergone 12 different configurations. Only the first core configuration consisted of all standard 8.5 wt% fuel supplied by General Atomic. Subsequently, the next seven cores consisted of mixed cores of 8.5 wt% and 12 wt%. Today the reactor core has 8.5 wt%, 12.0 wt% and 20 wt% fuel types. Because of the low reactor power level, the burn-up of the fuel is very small and all the fuels loaded into the core in 1982 are still there.

A properly administered reactor program was established, in order to provide core management function for RTP, and to maximize the lifespan of the fuel. At presently only nine fresh fuels are available in the inventory. Referring to the current operational trend, all of the fuel elements in the inventory are projected to be fully loaded into the core, and utilized, by the year of 2015. Table 2 lists the number of different types of fuel elements used in each core cycle.

3. Infrastructure

Six underground storage pits were constructed in the reactor building for the interim storage of the spent nuclear fuel elements. Each pit is capable of storing up to 24 spent nuclear fuel elements. Currently, a fuel transfer cask, for transferring the spent nuclar fuel element within the reactor hall, is not available at the facility.

The service vehicle access to the reactor hall is through a 12 ft wide double access sliding door located in the northen part of the reactor building. The reactor floor slab is designed to take the maximum load of 3 tonnes/m^2. A 10-tonne (safe working load) crane is available to serve the whole area of the reactor hall. Enough space is available for eventual loading of a transport cask inside of the reactor building.

A large parking area is available behind the reactor building to accommodate a 20-foot container. There is no interferences of fences, light poles, etc

The reactor site is easilly accessed via a secondary road which leads to the highway. The suitability of the roads and bridges for heavy cargos poses no technical problem for the transportation. Port Klang, a major port is about 80 km away.

TABLE 2. CORE CONFIGURATION AND CHANGES

Core configuration	**1**	**2**	**3**	**4**	**5**	**6**	**7**	**8**	**9**	**10**	**11**	**12**
Ring B	6	6	2	2	2	2	1	0	6	6	6	6
	0	0	4	4	4	4	5	6	0	0	0	0
	0	0	0	0	0	0	0	0	0	0	0	0
C	11	6	6	2	2	2	2	2	5	5	5	5
	0	5	5	9	9	9	9	9	6	6	6	6
	0	0	0	0	0	0	0	0	0	0	0	0
D	18	18	18	18	18	18	17	17	12	12	9	8
	0	0	0	0	0	0	1	1	6	6	9	10
	0	0	0	0	0	0	0	0	0	0	0	0
E	24	24	24	24	24	24	24	24	14	14	14	14
	0	0	0	0	0	0	0	0	0	0	0	0
	0	0	0	0	0	0	0	0	10	10	10	10
F	27	29	29	29	29	29	28	28	28	28	28	28
	0	0	0	0	0	0	0	0	0	0	0	0
	0	0	0	0	0	0	0	0	0	0	0	0
G	0	0	0	0	10	10	13	13	19	19	24	27
	0	0	0	0	0	0	0	0	3	3	0	0
	0	0	0	0	0	0	0	0	0	0	0	0
Sub-Total	86	83	79	75	85	85	85	84	84	84	86	88
	0	5	9	13	13	13	15	16	15	15	15	16
	0	0	0	0	0	0	0	0	10	10	10	10
Total in Core	**86**	**88**	**88**	**88**	**98**	**98**	**100**	**100**	**109**	**109**	**111**	**114**

Note: The first row lists the number of fuel elements with 8.5 weight percent of U-235
The second row lists the number of fuel elements with 12.0 weight percent of U-235
The third row lists the number of fuel elements with 20.0 weight percent of U-235

4. National regulation

Following the enactment of the Atomic Energy Licensing Act of 1984, the Atomic Energy Licensing Board (AELB) is responsible for the regulation and control of all activities dealing with atomic energy throughout the country. The major task of the AELB in pursuant of the objectives stipulated in the act and related to the activity of shipment, is to authorize activities “dealing with radioactive material and radiation producing device” (defined in the act as an activity involving the manufacturing, trading, producing, processing, purchasing, owning, using, transporting, transferring, handling, selling, storing, importing or exporting of radioactive material, nuclear materials, specifics substances prescribed in the act, or irradiation apparatus). For this purpose, an important partnership with AELB needs to be established to effectively meet the regulation and requirements regarding any spent nuclear fuel shipment. AELB is also responsible for the country Emergency Preparedness Plan.

5. Conclusion

Since RTP is the only research reactor in the country, the current discussion is to operate the reactor as long as possible. The fresh fuel stock enables the reactor operation for another decade. In view of this situation, the government decided not to participate in the initial phase of the ‘Foreing Research Reactor Spent Nuclear Fuel Acceptance Programme, in order to return the nuclear fuel to its country of origin. The extension of the US programme opens the possibility that our government participate of the program in the near future. Furthermore, the reactor will be 34 years old in 2016.

The experience of shipping spent nuclear fuel from Uzbekistan to the Russian Federation

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Abstract. In April 2006 the last of four shipments of spent nuclear fuel left the Institute of Nuclear Physics outside of Tashkent, Uzbekistan and traveled to the Mayak site in the Russian Federation. The shipment marked the completion of the first campaign under the National Nuclear Security Administration's Russian Research Reactor Fuel Return (RRRFR) Program to return highly enriched spent nuclear fuel to its country of origin. In total, 252 spent fuel assemblies containing over 63 kg of highly enriched uranium were returned. The project proved to be an excellent example of cooperation as four countries, Uzbekistan, Russia, Kazakhstan and the United States, were involved in its planning and implementation. This paper describes the shipment process from planning to completion with emphasis placed on the critical activities. Specifically the paper will discuss: the activities performed to prepare for the shipments; the roles and responsibilities of each country; the shipment details; the lessons learned; and the future plans of the Institute and the RRRFR program.

1. Introduction

In January 2006, Uzbekistan became the first country in fifteen years to return spent nuclear fuel (SNF) to the Russian Federation and the first under the Russian Research Reactor Fuel Return (RRRFR) Program. The RRRFR Program was created in 1999 from a tri-partite initiative between the Russian Federation, United States, and the International Atomic Energy Agency (IAEA) to return Russian-origin research reactor fuel containing high enriched uranium (HEU) from countries of the former Soviet Union. After the signing of the Implementing Agreement between the Government of Uzbekistan and the United States Department of Energy (DOE) in March 2002, the Institute of Nuclear Physics (INP) began the initial planning for the return of SNF from its WWR-SM research reactor[1]. Progress of the project was slow at first until the signing of the, '*Agreement Between the Government of the United States of America and the Government of the Russian Federation Concerning Cooperation for the Transfer of Russian-Produced Research Reactor Nuclear Fuel to the Russian Federation'* in May 2004, which gave the project the legal basis to proceed. The project experienced frequent challenges due to the fact that many of the laws, regulations, and procedures had

[1] Initial criticality was reached in 1959 using EK-10 fuel assemblies. The reactor operated at 10MW with 90% enriched IRT-3M fuels from 1971 to 1997. Since conversion in 1997, the reactor has been using 36% IRT-3M fuel assemblies.

changed dramatically since the last shipment of spent fuel in 1991. Through persistence and commitment to support the goals of the Global Threat Reduction Initiative (GTRI) and the RRRFR Program, the Russian Federation, Uzbekistan, Kazakhstan and the United States successfully coordinated the completion of the required preparation activities and shipped 252 SNF assemblies containing 63 kg of HEU. The timeline shown in Fig. 1 highlights the major milestones of the project.

This paper does not attempt to discuss all of the activities completed over the past two years but it focus on providing the details of the critical preparation activities and the actual shipment. The critical preparation activities were: Government-to-Government Agreements; Unified Project; TUK-19 cask licensing; Kazakhstan transit requirements; and facility preparations. The organizations involved are also identified along with a brief description of their responsibilities. The details of the shipments (i.e. cask loading, logistics) are described, and followed by the lessons learned and future plans of the reactor.

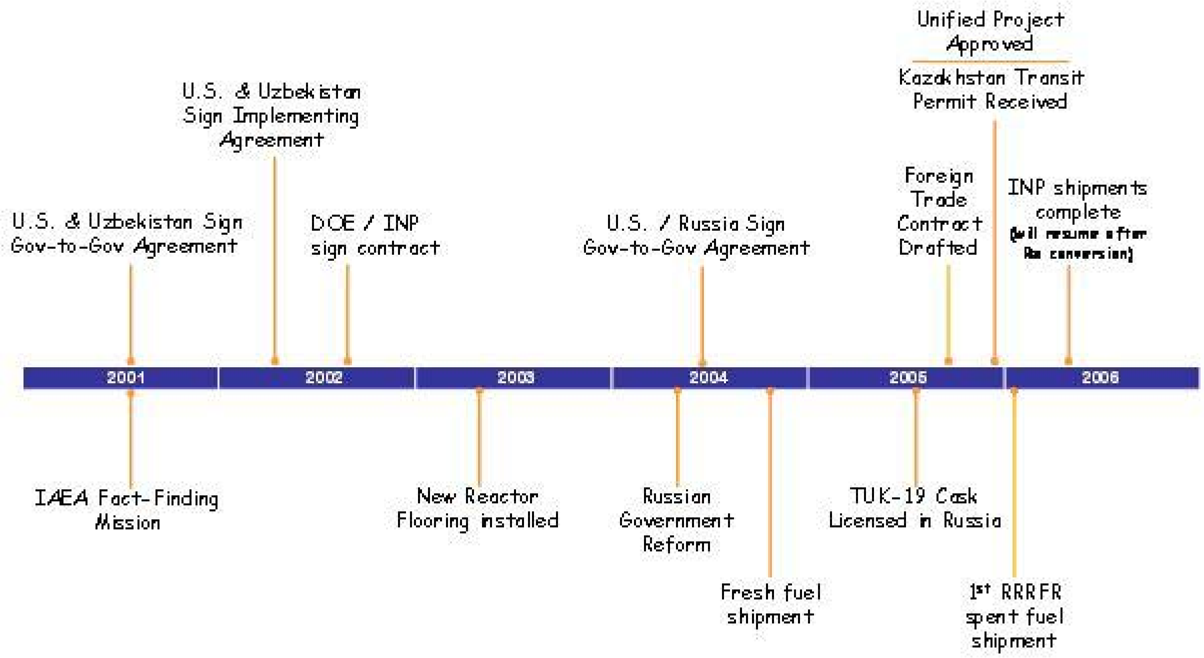


FIG. 1. Timeline of Shipment of research reactor fuel from INP to Mayak (Russia).

2. Critical preparation activities

Before the critical preparation activities are highlighted and discussed, it is important to identify the major organizations who were involved with the shipments and their roles and responsibilities. Table 1 includes the information on these organizations.

2.1. Government-to-government agreements (GTGA)

The first critical activity and major prerequisite for the preparation activities was the establishment of the government-to-government agreements between Uzbekistan and the United States and Uzbekistan and the Russian Federation. Two GTGAs provided the legal framework between the U.S. and Uzbekistan. They were: the "US/Uzbekistan Non-Proliferation Agreement" signed in June 2001; and the "DOE/Ministry of Foreign Affairs (MFA) Non-Proliferation Agreement" signed in March 2002.

TABLE 1. MAJOR ORGANIZATIONS INVOLVED IN THE PROJECT

Organization	Country	Description and Responsibilities
NNSA	United States	National Nuclear Security Administration – division of the DOE that manages and funds the RRRFR Program.
INP	Uzbekistan	Primary contractor with NNSA and primary contractor with Mayak. INP provided project management and was responsible for all of the activities within Uzbekistan.
Rosatom	Russian Federation	Federal Atomic Energy Agency – responsible for regulating the import of research reactor fuel.
Mayak	Russian Federation	Prime contractor with INP. Mayak was the shipper of record, provided the shipping containers and rail cars, unloaded the fuel, and is responsible for reprocessing and interim storage.
Techsnabexport (TENEX)	Russian Federation	TENEX – one of the two companies in the Russian Federation authorized by the Russian Government to import spent nuclear fuel. TENEX was subcontracted by Mayak to complete the Unified Project and authorize the import of the SNF.
VNIPIET and VNIIEF	Russian Federation	Subcontracted by TENEX to perform the safety analyses, prepare the required documentation, and obtain the licenses for the TUK-19.
KATEP	Kazakhstan	Company authorized to manage spent fuel shipments in Kazakhstan. KATEP coordinated all activities for the transit of the spent fuel.
KAEC	Kazakhstan	Kazakhstan Atomic Energy Committee – nuclear regulator for Kazakhstan. Approved the transit permits and cask license.

The first agreement provided liability protection and tax exemption for non-proliferation activities and the second delegated the DOE and the MFA as agents with permission to formulate the contract for the spent fuel return. The Governments of Uzbekistan and the Russian Federation agreed upon and signed an agreement (1997) on the peaceful use of atomic energy in which the focus was the dedication of the management of spent fuel. This GTGA was important because it served as the legal basis for the importation of spent nuclear fuel and the basis for the development of the Unified Project, as discussed in section 2.2, which was another critical path activity. With the government-to-government agreement in place, INP was permitted to formalize the contract with Mayak to return the spent fuel. This contract was called "Foreign Trade Contract" and addressed the following specific issues:

(1) The scope of services to be provided by the Russian Federation, including temporary storage of SNF, SNF processing, interim storage, and radioactive waste return.
(2) A clear definition of the owner of the SNF after its importation, and the owner of the reprocessing products after SNF reprocessing.
(3) Confirmation from Uzbekistan regarding the acceptance of the radioactive waste after a period of twenty years and assurances that all requirements are and will be met for the safe transportation of the SNF. The issue of the return of the radioactive waste after reprocessing is important because if the country decides to have the waste remain in the Russian Federation, additional costs (sometimes substantial) would result.

For future shipments of spent fuel, Rosatom has stated that the legal issues above should be included as part of the government-to-government agreement with the Russian Federation. The negotiation and approval of the government-to-government agreements are a lengthy evolution. Experience dictates that at least one year should be allotted for planning purposes.

2.2. *Unified project*

A Unified Project was required to allow the importation of spent nuclear fuel into the Russian Federation per Russian Law [1]. It was basically an overall assessment of the radiological, economical, social, and environmental impacts to the Russian Federation, particularly the areas surrounding the Mayak Plant (Chelyabinsk Region) [2]. The various elements that are included in the Unified Project are briefly discussed in the following paragraphs.

The first part of the Unified Project included the documents that make up the Special Ecological Programs (SEPs). The SEPs are used to rehabilitate the radioactive contaminated areas of the territory surrounding Mayak and are financed by the receipt of the SNF from foreign customers. In this case, the programs provide support to activities associated with the V-9 Industrial Water Basin and the development of systems for dosimetry, radiometry, and spectrometry monitoring. The SEPs went through a vigorous review process, with reviews by Rosatom, the Ministry of Economic Development and Commerce, and Medbiokestrem, culminating in a State Environmental Expert Review (SEER) by Rostechnadzor. A positive outcome from Rostechnadzor meant that the SEPs could be included in the final Unified Project package.

The second part of the Unified Project was the draft Foreign Trade Contract for the processing and storage of the SNF. The draft Foreign Trade Contract contained:

(1) The number of SNF assemblies to be shipped.
(2) The scope and cost of the services provided.
(3) Confirmation of the decision by the originating country to accept the return of the high level waste.
(4) Total project cost.
(5) Durations of temporary and interim storage.

The third part comprised of a set of documents that substantiate an overall radiation risk reduction and environmental safety increase as a result of the Unified Project implementation. These documents also address the storage durations and hazards associated with the products of the reprocessing activities. An additional document entitled ‘Assessment of Environmental Impact (AEI)’, not required at the time of the Uzbekistan shipment Unified Project, has been recommended by the SEER to be included in future Unified Projects.

The fourth and final part of the Unified Project was the set of materials used to discuss the SNF importation project with the community members and public organizations in the areas affected by the shipments. In this case, the records of discussions included people of the Chelyabinsk Region, city of Ozersk, and Mayak employees.

Once all of the required documents were collected into the final Unified Project package, it was submitted to Rostechnadzor for the State Ecological Expert Review. Positive results were transmitted to Rosatom, the Foreign Trade Contract was signed, and the Russian Government issued the declaration authorizing the importation of the SNF from Uzbekistan. Based on this experience, an interval of 15 months is recommended to develop and obtain the SEER approval for the Unified Project.

2.3. *TUK-19 cask licensing*

In the Russian Federation, casks transporting radiological materials must be licensed for both design and transportation [3]. The design license for the TUK-19 cask, shown in Fig. 2 had expired in 2000 due to its inactivity. Therefore, the transportation license, which is shipment specific and issued for each shipment campaign, required a new development. The transportation license included information such as: duration of the shipment; actual radioactive content; mode of transport; emergency card information; and proposed shipment category to name a few. Both the design and transportation licenses were analyzed and prepared by VNIPIET in less than five months with approval by Rosatom following shortly thereafter.

FIG. 2. The TUK-19 Transport Cask.

The Kazakhstan license for the utilization of the TUK-19 cask was issued by the Competent Authority, Kazakhstan Atomic Energy Committee (KAEC) of the Ministry of Energy and Mineral Resources (MEMR). The approval process involved reviews from both independent and state experts. The TUK-19 license in the Russian Federation was issued according to the regulatory guidelines developed in Russia, not the IAEA TS-R-1 guidelines adopted by the KAEC. Therefore, KAEC requested VNIPIET to prepare a comparative analysis that confirmed the compliance of the TUK-19 safety analysis to the IAEA TS-R-1 guidelines. KATEP was chosen to coordinate all necessary activities related for the licensing ot the TUK-19 cask. The license was issued by the KAEC in less than four months.

The TUK-19 license validation in Uzbekistan was issued by the State Inspectorate on Safety in Industry Mining using the Russian license. This activity was completed within two months.

2.4. Kazakhstan transit requirements

As with the TUK-19 cask license, KATEP coordinated all the necessary activities for the thansit of the spent fuel within Kazakhstan. This included the following main activities:

- Development of an ‘Assessment of Radiation Impact of SNF Transit to Environment and Population (EIA)’ and receiving the State ecological conclusion.
- Purchase of the required insurance (obligatory and voluntary) policies for the SNF transit.
- Development and approval of the SNF Transit Program.
- Obtaining the permission to transit through Kazakhstan
- Signature of all contracts for: rail transportation; physical protection; emergency preparedness; and customs.

The transit program was quite extensive and included provisions for liability, route selection, security, physical protection, and emergency preparedness. All competent authority approvals were coordinated by KATEP and received in less than three months.

2.5. Facility preparations

A number of facility and equipment enhancements were completed to support the loading and shipping of the TUK-19 casks. The major activities completed were:

- A new reactor hall flooring was installed to increase safety and help prevent the spread of contamination.
- New reactor hall lighting and remote operated cameras were installed to improve the conditions for fuel and cask handling. Previously, the crane operator used mirrors and visual cues to assist with the alignment of the basket and cask. The utilizaton of the new remote operated cameras improved the loading operations, making loading quicker and safer, as shown in Fig. 3.
- A backup generator was installed to provide emergency power if electrical power was lost during the loading operations.
- New transport racks were fabricated to secure the TUK-19 casks to the trucks during transport from the reactor to the rail yard.
- New trucks were procured to ensure the safe transport of the SNF and to reduce the number of road transports.
- Additional radiological monitoring and communications equipment was purchased.
- A self-releasing grapple was designed, fabricated and used to load the basket containing fuel assemblies into the cask.

FIG. 3. Utilization of a remote operated camera during loading operation.

The reactor staff and support organizations received extensive training on the operations and procedures of every aspect of the fuel shipment. Many practice exercises were performed on: fuel loading; cask handling and loading; cask preparation; criticality and personnel safety; radiological safety; and security.

3. Shipment details

The shipment consisted of the transport of 252 IRT-3M spent fuel assemblies enriched to 36% and 90% 235U. The IRT-3M assembly is shown in Fig 4.. Several months prior to the shipment, the fuel assemblies were inspected by Mayak experts. All assemblies met the acceptance criteria [4] for shipment and receipt with none requiring encapsulation. The TUK-19 cask was chosen because it was designed for Russian research reactor fuel and for use in the Russian designed reactors. The TUK-19 has the capacity to hold 4 IRT-3M assemblies and a total of 16 casks were available for each shipment. The casks were transported to Mayak by rail in 2 TK-5 railcars. Each TK-5 railcar holds 8 TUK-19 casks and has a roof that can be opened for loading and unloading operations (below, right). With a maximum of 64 IRT-3M fuel assemblies transported in each shipment, 4 shipments were needed to return the 252 spent fuel assemblies to Mayak.

FIG. 4. The IRT-3M fuel assembly.

FIG. 5. Cask loading into transport truck.

The shipment process was identical for each one of the 4 operations. The TK-5 railcars transported the empty TUK-19 casks from Mayak, through Kazakhstan to a rail yard near INP. The casks were off-loaded and transported to the reactor hall and staged for loading. The casks were allowed to acclimate for 24 hours before opening. Detailed cask loading plans were prepared in advance to ensure that none of the cask license contents limits (i.e. decay heat, activity, cooling time) were exceeded. IAEA inspectors were present and verified the presence of ^{137}Cs in 100% of the spent fuel assemblies. Each of the measurements was taken during the basket loading process and did not significantly affect the loading process. Once the four spent fuel assemblies were placed in the basket, the basket was remotely raised out of the spent fuel pool by the overhead crane and allowed to drip dry (removal of most of the water) for 15 minutes. After the drying period, the basket was placed into the cask. The remotely operated cameras at this point proved to be a tremendous improvement to historical loading operations. Due to the in-air loading the grapple was designed to self-release once the basket was fully lowered in the cask. This grapple worked flawlessly during all 64 cask loading operations. One reactor operator along with two radiological protection operators entered the reactor hall carefully monitoring the radiation levels. The operators were able to approach the cask and connect the crane hook to the cask lid. The cask lid was then installed on the cask and secured with two bolts prior to movement to its assigned storage spot. The remaining bolts were installed and torqued and the cask prepared for hermetic seal testing. A helium detector was used to confirm a proper seal per the TUK-19 handling instructions. The time spent to load a TUK-19 cask averaged less than one hour per cask.

On the day of the shipment, the TUK-19 casks were transported to the rail yard and loaded into the TK-5 railcars under constant security surveillance. Final surveys were conducted and the shipment left by a dedicated train at the predetermined time specified by the authorities. The transit time from Tashkent to Mayak was less than four days and the total turnaround time to return the empty casks was approximately three weeks. All four shipments were completed in less than four months, four months ahead of the baseline schedule. There were no incidences reported during the loading of the casks at INP and unloading of the fuel at Mayak.

4. Future plans

As reported earlier, the reactor currently operates with 36% enriched IRT-3M fuel assemblies. The Government of Uzbekistan and INP have decided to convert the reactor to use low enriched fuel, specifically, 19.7% IRT-4M fuel assemblies. Reactor conversion analyses with assistance from Argonne National Laboratory under the Reduced Enrichment for Research and Test Reactors (RERTR) program have been performed and final reactor parameters are being reviewed. There is a possibility that the reactor's power level may be increased to approximately 11 or 12 MW in order to preserve previous flux values. Other projects such as the refurbishment of the secondary cooling loop piping and the reactor control system are planned with their completion contingent upon funding support.

5. Lessons learned

Because this was the first shipment of research reactor spent fuel to the Russian Federation, a significant amount of information was learned that would apply to future shipments from other RRRFR Program countries. At the time this paper was written, the Uzbekistan shipment project team and the IAEA were finalizing the plans for a lessons learned workshop to be held in the near future. The focus of this workshop is to transfer knowledge and information regarding the necessary technical and administrative preparations to institutions planning future shipments. Ultimately, this information will be collected and published in a document intended for use as a reference tool. Sample passport and receipt documents, agreements, and contracts will be shared at the workshop. The experience demonstrated the need to identify all legal and technical requirements as soon as possible. It is also recommended that a dedicated project manager or team be appointed due to the significant work load. Lastly, it is important to have Mayak experts characterize and inspect the fuel to be shipped well in advance of the shipment to help reduce delays associated with suspect or deformed fuel assemblies.

6. Conclusion

The completion of the four shipments of spent HEU fuel from the Institute of Nuclear Physics of the Uzbekistan Academy of Sciences to the Russian Federation was a tremendous accomplishment for the Russian Research Reactor Fuel Return Program and the Global Threat Reduction Initiative. It marked the first return of spent research reactor fuel to the Russian Federation in over fifteen years and the first under the RRRFR program. Much was learned during the preparations phase, with many of the challenges requiring the development of new procedures to meet the updated regulations. In the end, the project proved to be an excellent example of international cooperation between the Russian Federation, Uzbekistan, Kazakhstan, and the United States in the area of nonproliferation.

The authors would like to extend their appreciation to Rosatom, the Governments of Uzbekistan and Kazakhstan, and the U.S. Department of Energy for their support of these spent fuel shipments and the RRRFR Program. Special thanks are also expressed to the team of experts from INP, Mayak, KATEP, TENEX, INL, SRS, the IAEA and others for their professionalism and excellent work.

REFERENCES

[1] Decree #418, ‘Government of the Russian Federation on the Procedure for Importation of Irradiated Nuclear Reactor Fuel Assemblies Into the Russian Federation’, July 11, 2003.

[2] LEBEDEV, A.E., ‘Unified Project, Importation of Irradiated Fuel Assemblies (IFA) from Foreign Research Reactors to Russian Federation’, presented at the RRRFR Transport Options Workshop, Varna, Bulgaria, September 21, 2005.

[3] PVSR-92, ‘On the Procedure for Issuing Certificates/Permits for Special Type Radioactive Substances, and For the Design and Transportation of Packing Sets with Radioactive Substances’ (modified based on Amendment 2 and Amendment 3 established by the Orders No. 448 of 07/17/98 and No. 663 of 10/25/99).

[4] OST 95 10297-95, ‘Spent Nuclear Fuel Assemblies of Nuclear Research Reactors, General Requirements for Delivery’.

Pictorical retrospective on return of foreign research reactor spent fuel to the United States of America:
The First Forty-three years

Jack Edlow

Edlow International Company, Washington, D.C., United States of America

On May 13, 1996, the DOE announced the Record of Decision for the Final Environmental Impact Statement on a Proposed Nuclear Weapons Non-proliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel [1]. Based on this Policy and subject to certain conditions, the US accepted and managed, in the US, foreign research reactor MTR fuel, TRIGA fuel and target material that was originally enriched in the US.

Under the terms of the Policy, aluminium-clad MTR spent fuel is accepted at DOE's Savannah River Site (SRS) and TRIGA fuel is accepted at the Idaho National Engineering and Environmental Laboratory (INEEL).

The preparation of spent fuel for shipment off-site is, many cases, a new activity involving higher levels of radioactivity than ever previously handled at the facility.

Nowadays, the shipment of spent fuel is a highly regulated activity requiring extensive co-ordination between the reactor facility operating organization and the cask owner/operator; and the transportation company, international, national and local governmental agencies, port authorities and the receiving facility (SRS or INEEL). While the operating organization may use consultants and contractors, it generally has the responsibility for the entire process, including co-ordination. Making arrangements for the first time for the shipment of spent fuel, especially internationally, requires a long lead-time and ample time must be allowed to perform the many procedures and tasks necessary.

This paper is a short historical review of pictures showing how were things prior to 1966, because we should not forget that the story start with Eisenhower's "Atoms for Peace" Address in December 8, 1953,

> *"The United States knows that peaceful power from atomic energy is no dream of the future… That capability is here today. … I would be prepared to submit to the Congress…any such plan that would: …encourage world-wide investigation into the most effective peace time uses of fissionable material, and with the certainty that they had all the material needed for the conduct of all experiments that were appropriate…"*


Savannah Morning News

Atomic Shipment Heads West

FIG. 1. First shipment of FRR returns to the U.S. in 1963 and leaves Savannah.

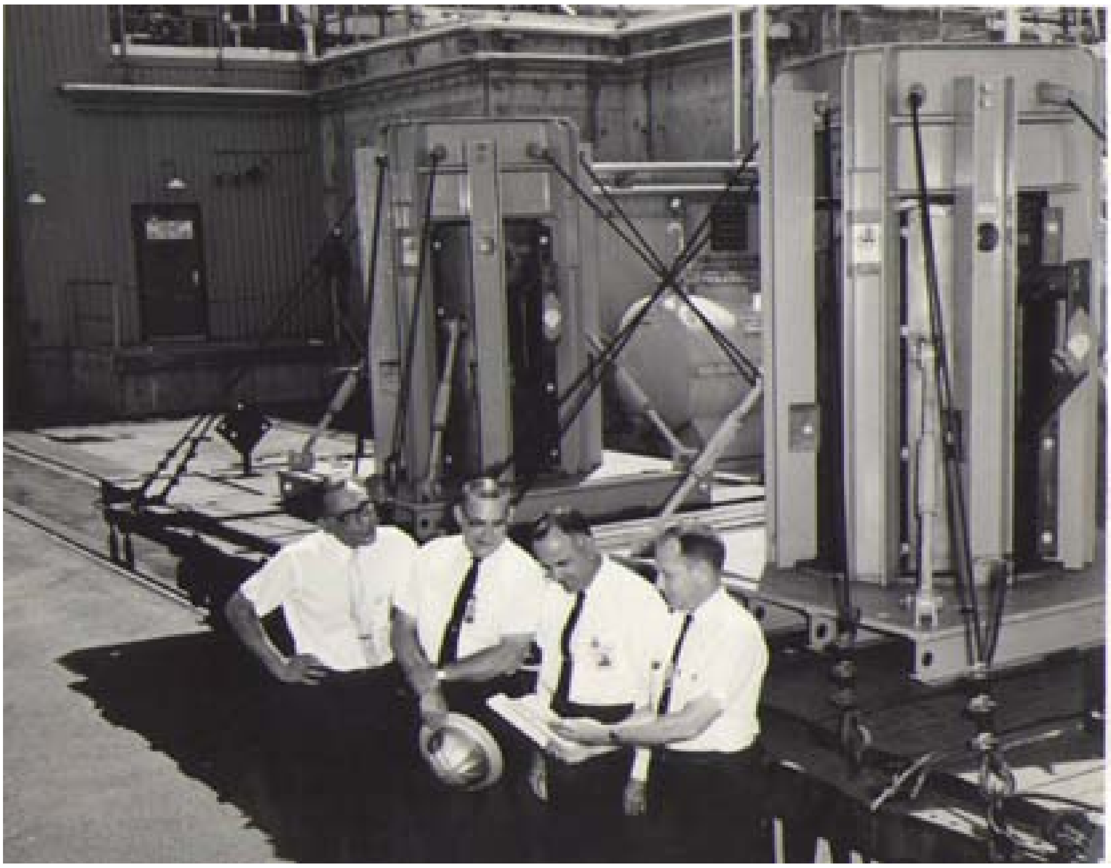
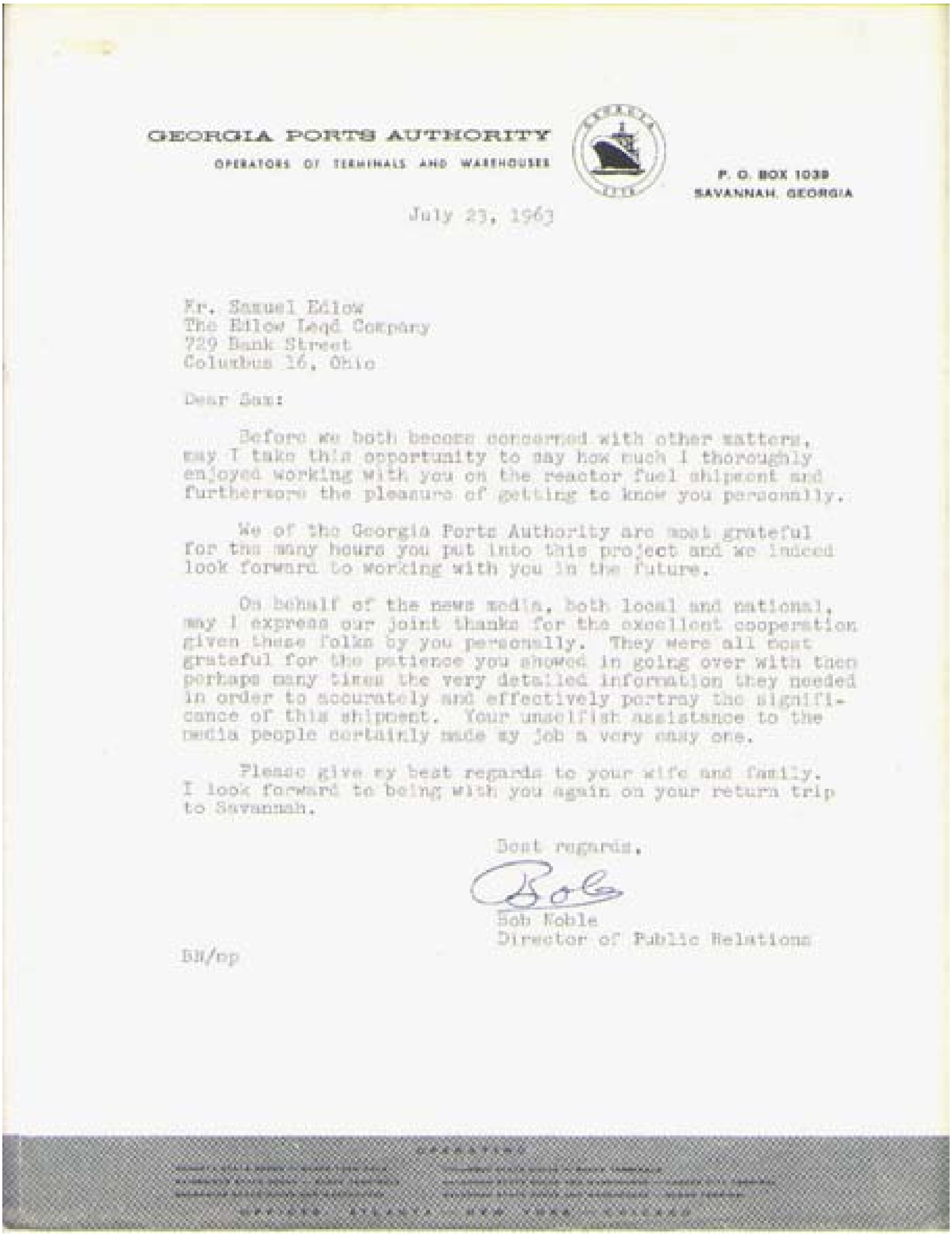


FIG. 2. Sam Edlow and USAEC officials receive a shipment.



GEORGIA PORTS AUTHORITY
OPERATORS OF TERMINALS AND WAREHOUSES

P. O. BOX 1039
SAVANNAH, GEORGIA

July 23, 1963

Mr. Samuel Edlow
The Edlow Lead Company
729 Bank Street
Columbus 16, Ohio

Dear Sam:

Before we both become concerned with other matters, may I take this opportunity to say how much I thoroughly enjoyed working with you on the reactor fuel shipment and furthermore the pleasure of getting to know you personally.

We of the Georgia Ports Authority are most grateful for the many hours you put into this project and we indeed look forward to working with you in the future.

On behalf of the news media, both local and national, may I express our joint thanks for the excellent cooperation given these folks by you personally. They were all most grateful for the patience you showed in going over with them perhaps many times the very detailed information they needed in order to accurately and effectively portray the significance of this shipment. Your unselfish assistance to the media people certainly made my job a very easy one.

Please give my best regards to your wife and family. I look forward to being with you again on your return trip to Savannah.

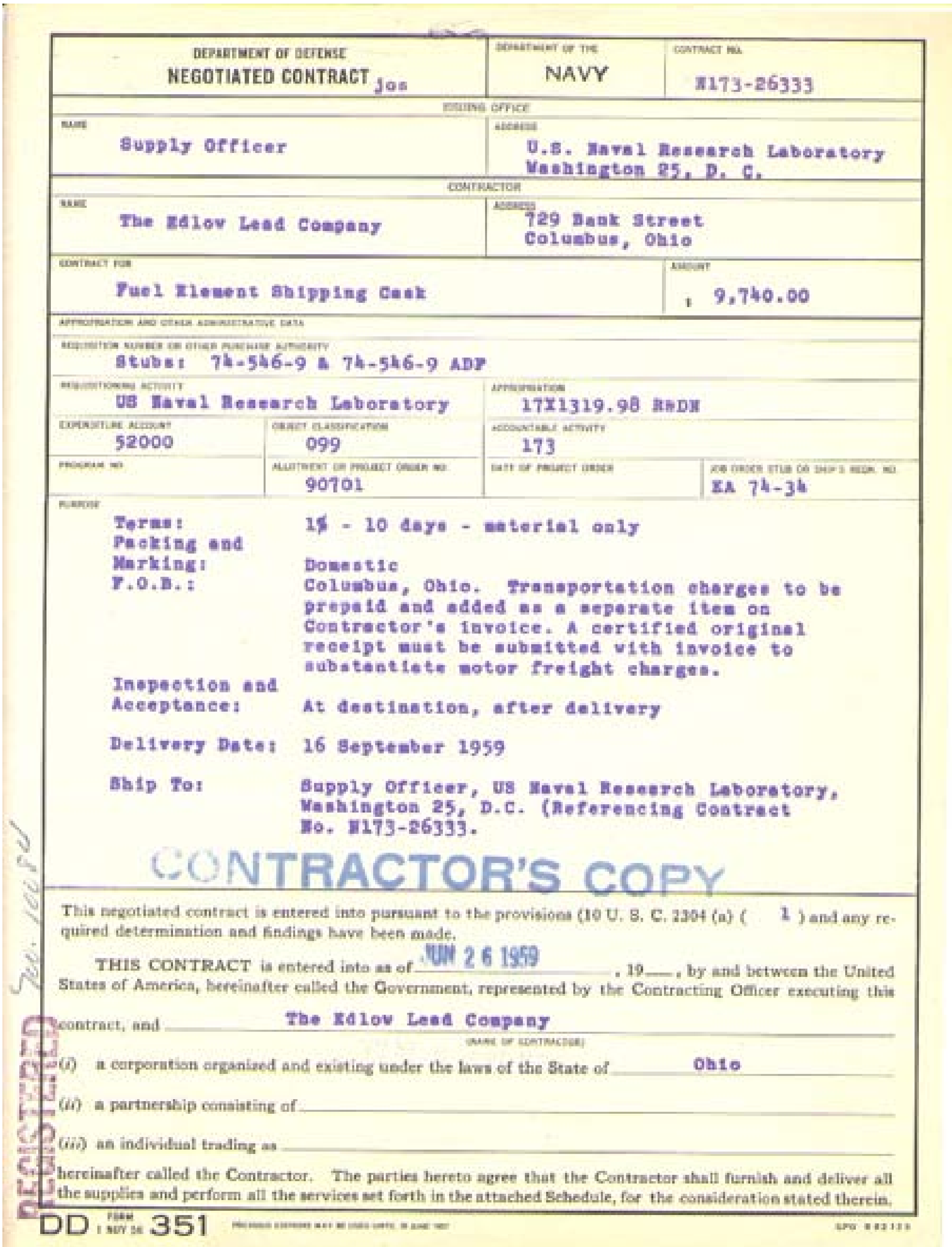
Best regards,

Bob

Bob Noble
Director of Public Relations

BN/np

FIG. 3. Ports were happy receiving the fuel.


DEPARTMENT OF DEFENSE
NEGOTIATED CONTRACT jos

DEPARTMENT OF THE: NAVY

CONTRACT NO.: N173-26333

ISSUING OFFICE

NAME: Supply Officer

ADDRESS: U.S. Naval Research Laboratory, Washington 25, D. C.

CONTRACTOR

NAME: The Edlow Lead Company

ADDRESS: 729 Bank Street, Columbus, Ohio

CONTRACT FOR: Fuel Element Shipping Cask

AMOUNT: $ 9,740.00

APPROPRIATION AND OTHER ADMINISTRATIVE DATA

REQUISITION NUMBER OR OTHER PURCHASE AUTHORITY: Stubs: 74-546-9 & 74-546-9 ADP

REQUISITIONING ACTIVITY: US Naval Research Laboratory

APPROPRIATION: 17X1319.98 R&DN

EXPENDITURE ACCOUNT: 52000

OBJECT CLASSIFICATION: 099

ACCOUNTABLE ACTIVITY: 173

PROGRAM NO.:

ALLOTMENT OR PROJECT ORDER NO.: 90701

DATE OF PROJECT ORDER:

JOB ORDER STUB OR SHIP'S REQN. NO.: EA 74-34

PURPOSE

Terms: 1% - 10 days - material only

Packing and Marking: Domestic

F.O.B.: Columbus, Ohio. Transportation charges to be prepaid and added as a seperate item on Contractor's invoice. A certified original receipt must be submitted with invoice to substantiate motor freight charges.

Inspection and Acceptance: At destination, after delivery

Delivery Date: 16 September 1959

Ship To: Supply Officer, US Naval Research Laboratory, Washington 25, D.C. (Referencing Contract No. N173-26333.

CONTRACTOR'S COPY

This negotiated contract is entered into pursuant to the provisions (10 U. S. C. 2304 (a) (1) and any required determination and findings have been made.

THIS CONTRACT is entered into as of JUN 26 1959, 19__, by and between the United States of America, hereinafter called the Government, represented by the Contracting Officer executing this contract, and The Edlow Lead Company (NAME OF CONTRACTOR)

(i) a corporation organized and existing under the laws of the State of Ohio

(ii) a partnership consisting of

(iii) an individual trading as

hereinafter called the Contractor. The parties hereto agree that the Contractor shall furnish and deliver all the supplies and perform all the services set forth in the attached Schedule, for the consideration stated therein.

REGISTERED

DD FORM 1 NOV 56 351

FIG. 4. Casks were less expensive (1959).

FIG. 5. Casks were transported by air without significant difficulties.

FIG. 6. Casks were transported by train without significant difficulties.

FIG. 7. More recently, one of many shipments to return HEU fuel to the U.S.

TABLE 1. FOREIGN RESEARCH REACTOR FUEL SHIPMENTS TO SRS 1978-1989

Year	Country	# of Casks	# of Assemblies
1978	Sweden, Netherlands, Canada, Germany, Denmark, France	64	1715
1979	Belgium, France, Germany, Sweden, Netherlands	49	989
1980	Germany, France, Sweden, Belgium, Austria, Canada, Italy, Denmark, Japan	46	952
1981	France, Sweden, Netherlands, Japan, Denmark, Germany	45	1107
1982	South Africa, France, Germany, Japan, Italy, Sweden, Belgium	36	939
1983	France	6	92
1984	France, Canada	5	80
1985	France, Canada	10	313
1986	France, Spain, Netherlands, Japan, Germany	17	433
1987	France, Germany, Netherlands, Sweden, Denmark, Taiwan	33	728
1988	France, Germany, Netherlands, Japan, Canada, Taiwan	41	920
1989	France, Netherlands, Japan, Denmark, Taiwan	41	695
	SUBTOTAL – 1978-1989	**393**	**8963**

TABLE 2. FOREIGN RESEARCH REACTOR FUEL SHIPMENTS TO SRS 1978-1989

Year	Country	# of Casks	# of Assemblies
1978	Sweden, Netherlands, Canada, Germany, Denmark, France	64	1715
1979	Belgium, France, Germany, Sweden, Netherlands	49	989
1980	Germany, France, Sweden, Belgium, Austria, Canada, Italy, Denmark, Japan	46	952
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1987	France, Germany, Netherlands, Sweden, Denmark, Taiwan	33	728
1988	France, Germany, Netherlands, Japan, Canada, Taiwan	41	920
1989	France, Netherlands, Japan, Denmark, Taiwan	41	695
	SUBTOTAL – 1978-1989	**393**	**8963**

TABLE 3. FOREIGN RESEARCH REACTOR FUEL SHIPMENTS TO SRS 1964-1977

Year	Country	# of Casks	# of Assemblies
1964	Sweden	3	90
1965	Sweden, France	7	234
1966	Sweden, Germany, France, Canada	23	523
1967	Sweden, Canada	34	684
1968	Canada	33	352
1969	Canada	11	243
1970	Canada	11	233
1971	Canada	5	107
1974	Canada	3	62
1975	Sweden, Canada, Puerto Rico	6	72
1976	Sweden, Canada	22	357
1977	Sweden, South Africa, France	19	240
	SUBTOTAL	**177**	**3197**

REFERENCE

[1] "Record of Decision on a Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel (ROD)", United States Federal Register, 61 FR 25092, May 17, 1996, as amended July 25, 1996..

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