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European Nuclear Society
Rue de la Loi 57
1040 Brussels, Belgium
Phone + 32 2 505 30 54
Fax +32 2 502 39 02
E-mail ens@euronuclear.org
Internet www.euronuclear.org

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Session VII

Fuel Back-end Management

SESSION VII - FUEL BACK-END MANAGEMENT

THE JASON REACTOR: FROM CORE REMOVAL TO FUEL REPROCESSING

P. BEELEY, A. WILLIAMS, R. LOCKWOOD
Nuclear Department, Defence College of Electromechanical Engineering
HMS SULTAN, Gosport PO12 3BY, UK

B. RAYMOND, N. SPYROU
Department of Physical and Electronic Sciences, University of Surrey
Guildford, Surrey, UK

P. AUZIERE
AREVA NC, Treatment Business Unit
Velizy, Cedex, France

ABSTRACT

The 10 kW JASON Argonaut reactor was operated at the Royal Naval College, Greenwich, London, between 1962 and 1996. After initial cooling in the core, the MTR type fuel (80% enriched ^{235}U) was dry stored on site before transport in 1998 to BNFL, Sellafield for interim wet storage. Arrangements for reprocessing of the fuel at AREVA NC, La Hague are now in progress and this paper will describe various aspects of the storage, transfer and monitoring, including criticality calculations using MCNP, prior to its possible final transport to France for reprocessing.

1. Introduction

The JASON 10 kW Argonaut reactor was used to educate and train military and civilian personnel involved in the naval nuclear submarine propulsion programme. It was situated in a reactor hall within King William Building, a Grade 1 listed building within the Old Royal Naval College at Greenwich, which itself is a Scheduled Ancient Monument having World Heritage Site status. The reactor was first taken critical at the College in November 1962, having been previously operated by the Hawker Siddley Nuclear Power Corporation at Langley from February 1959. The decision to decommission Jason was taken in 1996, following the ministerial decision that the Royal Naval College would pass to non-defence use by the millennium.
Following the final reactor shut-down, the decommissioning programme began with the setting up of the management, safety and project teams, obtaining the various heritage and planning approvals for the works and gaining regulatory approval of the nuclear safety cases. The first physical stage in the programme, called Post-Operational Clear Out (POCO), involved disabling the reactor and removing particular operational equipment and was completed by June 1998. The rest of the programme consisted of fuel removal, reactor dismantling, waste removal, site survey and clean up and the final radiological clearance of the site by the Environment Agency, which was achieved on 4 November 1999. A full description of the JASON decommissioning has been given previously [1,2].

Criticality calculations have been performed for the JASON fuel in the reactor hall fuel store and in the UNIFETCH flask using the general purpose Monte Carlo code MCNP4c2. Each model included as much geometrical detail as was available. The most recent ENDF/B-VI cross-section data was used. In addition, MCNP was used for dose rate calculations. The MCNP source term was created from the fission product inventory calculated by AEA Technology in 1998 from the JASON power history modified to account for radioactive decay. It should be noted that due to the very low power operation (25 MWh over 36 years) and nine years of storage, the activity in the fuel is very low. The dose rate at one metre from a single fuel box in air is approximately15 μSv/h now.

2. Storage on Site and Fuel Transfer to Sellafield

Fuel removal from the core was completed under the initial POCO phase and the fuel was stored in dry fuel pits within the main reactor hall to allow for radioactive decay of the short lived fission products. In total eight fuel pits were used to store a maximum of 16 fuel boxes and MCNP calculations indicated k_{eff} of 0.23456 and 0.38909 respectively for normal dry storage and worst case flooding of the pits. With respect to fuel burnup the following data was generated at the time of decommissioning [3]:

- The standard core used in JASON used 10 fuel boxes of 16 plates and 1 box of 12 plates. Calculations performed at RNC give the "start of life" fuel inventory in the core as 1980.98 g of U-235, 0.0274 g of U-234 and 495.245 g of U-238.

- Total mass of Uranium in the core at start of life is therefore 2476.2524 g.

- The JASON reactor power history gives the total integrated power between 1961 and shutdown in 1997 as 25.13 MWh. (The reactor had been operated by Hawker Siddley Nuclear Power Company Ltd. for a total of 1.4 MWh before 1961).

- Fuel inventory and activation level calculations for the standard core and power history were performed by AEA Technology using the codes WIMS, TRAIL and FISPIN in 1997 as part of the decommissioning.

- At shutdown in 1997, these calculations give the total mass of Uranium in the core as approximately 2480 g (calculation rounding), and the total mass of Plutonium in the core as 0.0142 g.

The overall fuel removal phase consisted of several interrelated off-site and on-site preparatory activities, including the procurement, preparation and modification of a UKAEA UNIFETCH fuel transfer flask. Modifications were made off-site to interface the flask to the existing fuel removal transfer equipment held at the College. Extensive trials were also carried out using a dummy fuel module to test the modified fuel handling equipment.
The other major off-site activities included the design, manufacture, test and installation of a range of custom made and proprietary equipment needed to install the UNIFETCH flask in the reactor hall, transfer the Jason reactor fuel modules, remove the flask and subsequently transport it by road to British Nuclear Fuels (BNFL), Sellafield (now BNGSL). After this equipment had been installed and tested, non-active UNIFETCH flask commissioning trials were carried out. The internal fuel transfer route was then proven using a dummy fuel element, witnessed as required by the regulator and the Jason Reactor Safety Committee. Final approvals were obtained

by 11 September 1998. The fuel was then transferred into the UNIFETCH over a two-day period and it was subsequently transported by road to BNFL, Sellafield, on 16 September 1998. The open UNIFETCH flask showing the fuel baskets is shown in Fig. 1 and the UNIFETCH during transport is shown in Fig 2.

Fig 1. Open UNIFETCH Flask showing the baskets containing JASON Fuel

Fig 2. UNIFETCH Flask during transportation

Criticality calculations using MCNP indicated k_{eff} of 0.03046 and 0.67800 respectively for normal dry storage and worst case flooding of the flask.

3. Interim Storage at Sellafield

After transport to BNFL Sellafield the JASON fuel remained in dry storage in the UNIFETCH flask until the end of November 1998 after which each fuel box was transferred into individual storage locations within a designated storage rack in a suitable pond. Each rack was fitted with a

steel sleeve to accommodate the fuel within the rack. All fuel modules were inspected for corrosion and damage to provide benchmarks for future inspections. Criticality calculations were carried out by BNFL using the MONK code and the JASON fuel store fell within the safety criterion for the pond.

The pond chemistry in the JASON store at Sellafield must continue to be carefully controlled to allow for interim storage of JASON fuel. The pH levels of the JASON pond at various intervals between 10/08/1998 and 21/12/2000 are shown in Fig 3. During this time the pH varied but still remained managed within the acceptable band of 5.3 to 9.0. Peak conductivity levels were measured at 8.1 µS/cm. While this is above the level of 1 µS/cm that suppresses corrosion, it is much lower than the levels that result in rapid corrosion (100 – 600 µS/cm). It was therefore concluded that while corrosion may not be eliminated, the rate of corrosion will be acceptably low.

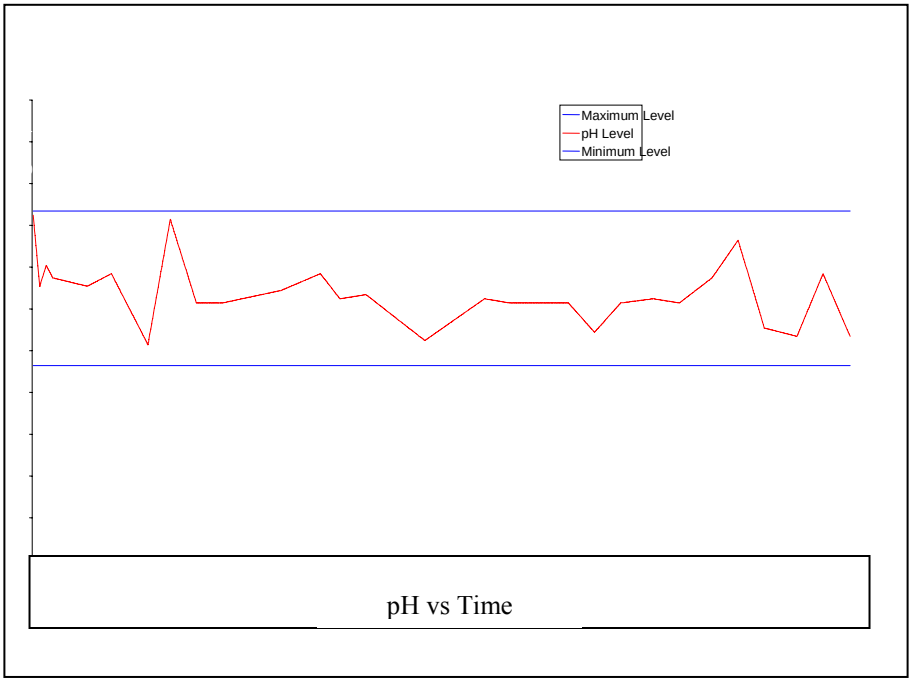

Fig 3. pH in the JASON Fuel Pond –August 1998 to December 2000 [6]

In addition to routine sampling, the JASON fuel has been visually examined using CCTV to monitor any potential corrosion. Fig. 4 shows screen captures from a video inspection of the JASON fuel made in 2001. Fig. 4-C shows a healthy fuel box with no visual corrosion. Fig. 4-B shows a structurally intact fuel box but with pit and crevice nodules on the surface of the lower aluminium location cone. Since the fuel boxes showed no signs of corrosion prior to storage, the difference in corrosion rates could be the result of localised poor chemistry or galvanic corrosion where some of the aluminium cladding has been in contact with the steel sleeves. Additionally, some fuel boxes had been in longer contact with water through more frequent use in the reactor at Greenwich. Consequently some minor corrosion/damage may have been sustained which could have made them more susceptible to corrosion later.

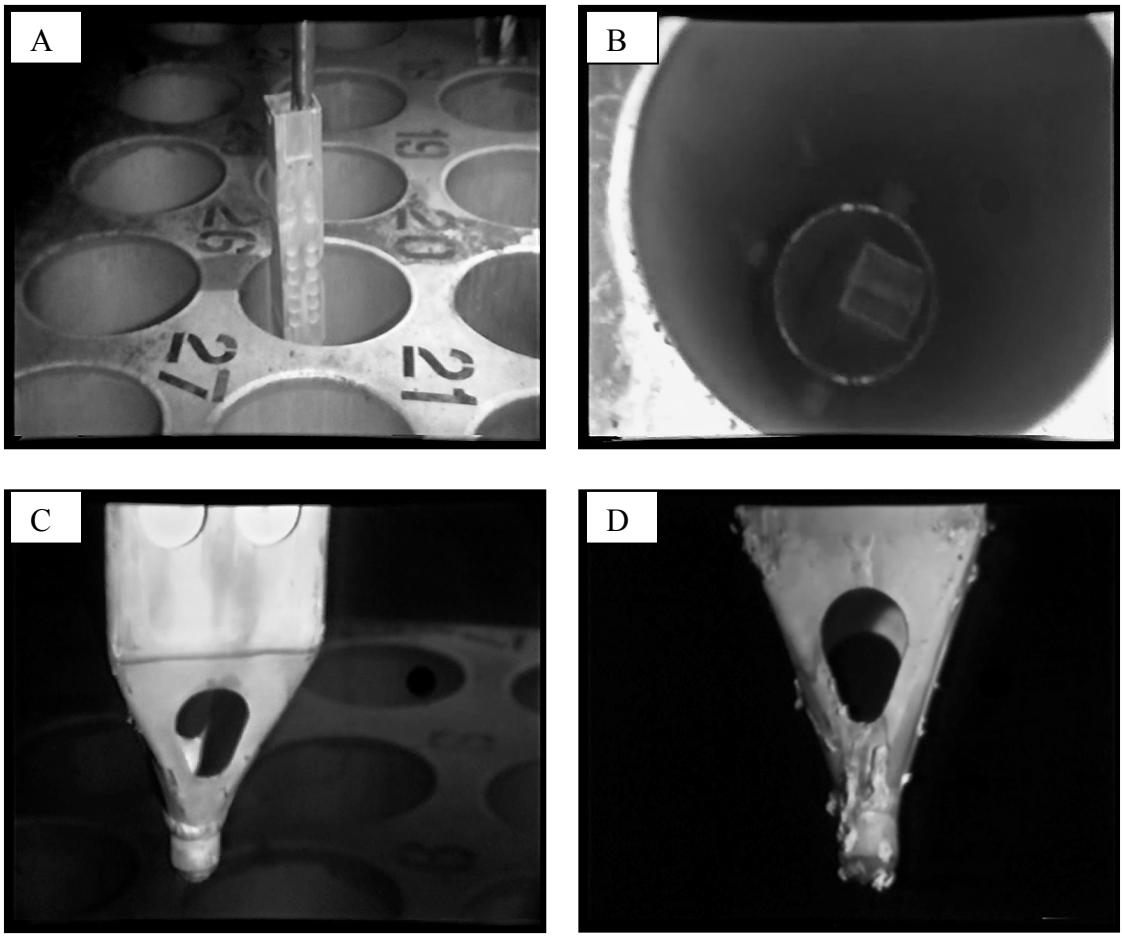

Fig 4. JASON Fuel in the BNGSL Fuel Pond [6]

4. Planning for Transportation to AREVA NC La Hague for reprocessing

The JASON fuel is now being considered for reprocessing at AREVA NC La Hague site, France. It should be noted that the fuel was originally intended to be reprocessed at UKAEA Dounreay before closure of the MTR reprocessing facility.

The various industrial operations that may be carried out during reprocessing are shown in Fig. 5. At present, it is assumed that the JASON fuels will be transported, possibly with another consignment from the UK, in a TN MTR 52 cask designed and operated by TN International (a subsidiary of AREVA NC). The basket inside the cask can hold up to a maximum of 52 MTR used fuel elements, which will be sufficient for this operation [4,5]. TN International will be responsible for the transport except for the nuclear responsibility which will be supported by UK authorities and by AREVA NC in France. TN International will witness and provide guidance to loading operations at selected UK sites in order to guarantee the right application of the TN MTR 52 operating instruction manual (already transmitted to UK's MTR project team) especially for the dryness operations, as well as supplying the necessary equipment and tools to the UK sites.

Once the cask reaches La Hague site, it is planned that the used fuel elements and the AREVA NC Boxes will be unloaded into a single temporary storage basket which capacity is 16 x 4 used fuels. AREVA NC La Hague plant has a long feedback experience in the reception and

unloading of the TN MTR transport cask at the La Hague site, and the AREVA NC Box is fully compatible with the unloading equipment.
The interim storage basket is then placed in one of the interim wet storage pool of La Hague plant. AREVA NC will be responsible for maintaining the integrity of the fuel during this interim storage phase.

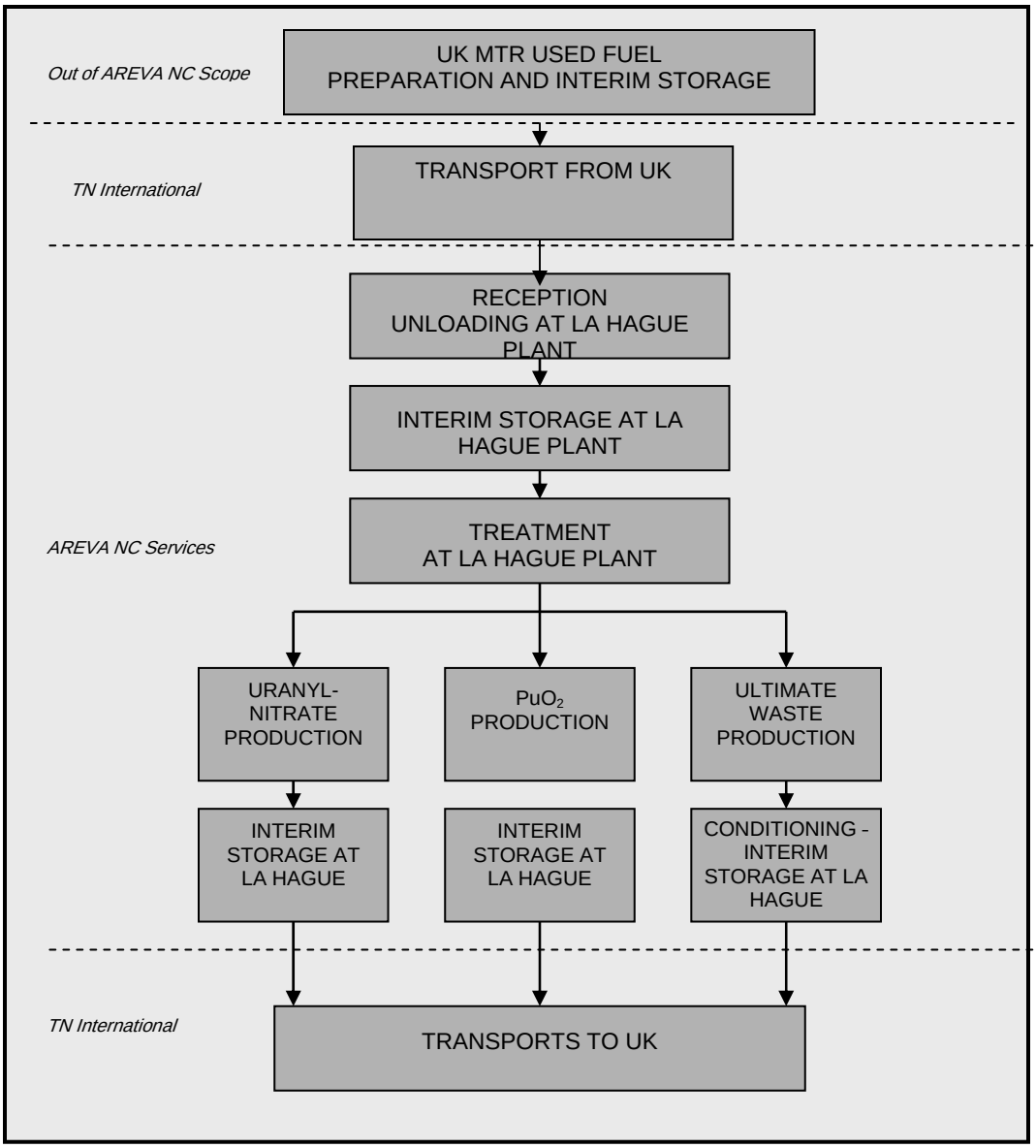

Fig. 5 Scope of Work for Reprocessing of JASON Fuel

At a suitable time, the JASON fuels will be removed from interim storage for treatment. The treatment consists doing nitric acid dissolution batches according to appropriated operating conditions.
The reprocessing operations will comply with EURATOM accountancies rules from fissile material management point of view, therefore different consignments (JASON and other UK

MTR fuel) will be dealt with separately. Other rules state that for dissolution the number of batches required depends on the amount of aluminium present (limited to 140 kg of Al per batch). After dissolution, the liquid solution will be sent to the Accountability Unit for nuclear material balance.
Next, the solution will be mixed with UOx solution for chemical treatment, including; fissile material partition and purification, fission product concentration and vitrification. At the outlet of the reprocessing line the ^{235}U concentration will be less than 1%.
The radioactive waste resulting from the processing of these used fuels will be conditioned into a suitable package for return to UK.

5. Summary

Following decommissioning of the JASON reactor the MTR type fuel has been in interim storage at BNG Sellafield. Review is now in progress for the possible reprocessing of the fuel at the AREVA NC La Hague site. The earliest provisional time scales for this back end of the JASON fuel cycle are shown in Fig. 6.

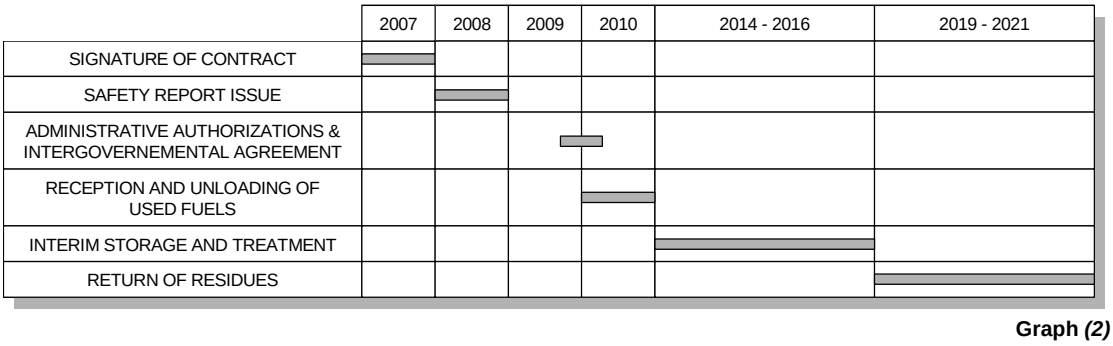

Fig. 6. Time Scales for the Back End of the JASON Fuel Cycle

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Research and Test Reactor Fuel Treatment at AREVA NC la Hague

Estelle Hélaine, Philippe Bernard[1], Jean-Luc Emin[2]

Dominique Lepoittevin, Frédéric Gouyaud[3]

[1]SGN, AREVA Group 25 avenue Tourville, 50120 Equeurdreville , France
[2]AREVA NC, AREVA Group, 4 rue Paul Dautier, 78143 Vélizy Cedex, France
[3]AREVA NC, AREVA Group, 50444 Beaumont Hague Cedex, France
estelle.helaine@areva.com

ABSTRACT

The La Hague plant has a long successful history of treatment power reactor spent fuel. To accommodate a new contract, the process needed to be adapted to fuel from Research Test Reactors (RTR) taking into account the dimensional characteristics, the chemical composition and the high uranium enrichment of RTR fuel. Characteristics that are very different from those of power plant spent fuel.

Several treatment options resulted from the R&D performed by CEA and conceptual studies and preliminary design proposed by AREVA/SGN. In addition, SGN benefited from previous experience processing RTR fuel near Marcoule. The chosen approach for the detailed design consists of dissolving the RTR fuel in a dissolution pit in one line of dissolver facility (T1B). The process was designed for small RTR fuel elements composed of aluminium and uranium alloy.

"Taxi" baskets containing the RTR fuel elements are transferred from the fuel storage pools, to the feed cell in T1B. The fuel elements are loaded into canisters and transferred to a rack in the general maintenance cell. The RTR fuel canisters are picked up one by one and the fuel element is dropped into a transfer tube leading to the dissolution pit.

The RTR dissolution process is fundamentally different from the continuous power reactor fuel dissolution process and is done in batches. At the end of a batch, the solution is drained from the dissolver and diluted with UOX solutions from line T1A.

The resulting UOX/RTR mixture complies with the specifications for downstream processing operations, including high level waste vitrification, where aluminium is incorporated into glass in accordance with specified limits.

The innovative nature of the process, demanded by the special characteristics of the RTR fuel, required a major qualification program. The main objectives of the qualification program were to validate the dissolver pit concept, to verify basic RTR process data for the T1B production environment and to acquire data showing control of the process.

The first active batch dissolution was successfully performed from June 9th to June 25th in 2005. About 2 tons of Research and Test Reactor fuel coming from Australian and Belgium have been reprocessed by the end of 2006.

Feed-back has been collected to improve the process and get in-depth knowledge about RTR fuel treatment. The feed back is also useful to enlarge the scope of RTR treatment to uranium/silicon or uranium/molybdenum based fuels.

The paper will describe the process implemented in the La Hague plant and the qualification program, as well as the main results from the first year RTR treatment.

1 INTRODUCTION

Research and testing reactors (RTR) are used for nuclear applications in many fields, including medicine, with boron neutron capture therapy, industry, with gauges, detectors, and other devices, research and education with irradiators and calibration sources. All these applications generate radioactive waste. Used Nuclear Fuel (UNF) back-end management has experienced many stops and starts in the past ten years.

Until the end of 1988, US obligated materials were subject to the "Off Site Fuels Policy" which required spent fuel to be returned to the United States and to be reprocessed there. Since this policy terminated on the 31st of December 1988, research reactors operators were then required to implement other management solutions.

At the same time, the Reduced Enrichment for Research of Test Reactors (RERTR) Program was leading to a new Low Enrichment Uranium (LEU) fuel to replace High Enrichment Uranium (HEU) fuel.

Since new LEU fuel was not as easy to reprocess as HEU fuel (the LEU fuel is made of silicide, whereas the HEU fuel is made of aluminide), a new US spent fuel return policy was introduced in early 1996 for all research reactors converted (or that had agreed to be converted) to LEU, and for reactors operating with HEU for which no suitable LEU was available. This policy covers all the spent fuels discharged.

At the end of this return program, each operator will again be fully responsible for its spent fuel. For the ultimate back-end management, there are three options [1] :

① Interim storage
② Direct disposal
③ Treatment-conditioning by treatment

The interim storage option does not constitute a reliable solution, while some research reactor operators have been confronted with corrosion and material degradation problems in existing facilities. Extended storage of RTR fuel would obviously require extensive R&D programs, as well as new facilities designed for long term storage. Most significantly, this option does not provide a definitive solution.

The direct disposal option entails several unsolved difficulties. First, one has to ensure that the enriched uranium content will not lead to criticality hazards through long term processes as selective leaching. Moreover, RTR spent fuel is generally considered as unstable (high corrosion rate, hydrogen build-up) under the conditions of a geological repository. It requires watertight and durable conditioning on a geological time-scale, for which no satisfactory solution has yet been found. Finally, direct disposal remains a "virtual" solution that has never been implemented.

The treatment option avoids the above difficulties because it produces residues which are suitable for direct disposal. The 40 years of experience gained at AREVA NC's treatment site in La Hague demonstrates the industrial expertise achieved in commercial treatment.

2 PRINCIPLES OF TREATMENT : THE EXAMPLE OF LA HAGUE

Treatment has two main objectives:

① Recover the recyclable materials (mainly uranium and small quantities of plutonium),
② Generate final waste according to their potential hazards, in order to dispose them safely for the environment.

Treatment at the La Hague complex, in the UP2-800 and UP3 plants, uses the PUREX process, including the following steps (see flow sheet in figure 1) :

- *Transport* of fuel to the plant and *cooling* in storage ponds. This cooling, or "deactivation", decreases the radioactivity of the fission products substantially.

- *Shearing* and *dissolution* of the fuel, followed by *clarification* of the liquor generated:

The first treatment operation consists of stripping the fuel rod to prepare it for chemical attack.

The process employed for zircalloy cladding rods is cutting them into pieces with a shearing machine. At La Hague, the shearing machine is placed above a continuous dissolver, and the rod pieces fall into a perforated basket, which, placed in the dissolver, allows the selective dissolution of the oxide in nitric acid without attacking the hulls. At the end of the operation, the hulls are removed and sent in a workshop for compacting and conditioning as a solid Medium Active Level Waste (MALW).

Depending on the fuel type, it is possible that some insoluble products may remain after dissolution. This is the case of oxide fuels for which the insoluble particles are made of cladding residues and metallic inclusions. These solids, which would hamper further purification steps, are removed from the solution by centrifugation. A centrifugal clarifier has been selected for La Hague because it provides good efficiency with high throughput.

- *Uranium and plutonium splitting and purification* by a liquid-liquid extraction process :

Basically, extraction consists in transferring a solute from one liquid phase to another one that is not miscible with the first. This operation enables separation of salts whose suitability for extraction by a given solvent is different.

For the extraction of uranium and plutonium, tri-butyl phosphate diluted in hydrocarbons has been universally adopted. In the extraction operation, most of the fission products and actinides, except U and Pu, remain in the aqueous phase. Scrubbing by nitric acid improves the separation by stripping most of the fission products entrained by the solvent.

Several extraction cycles of the clarified liquor, in pulsed columns, mixer-settler banks, or centrifugal extractors are necessary to meet the end-product specifications. At the end of these cycles, different kinds of solutions are generated:

- a solution containing specifically the uranium
- a solution containing specifically the plutonium
- raffinates containing the fission products and the minor actinides
- the solvent, which is regenerated by a treatment with sodium carbonate followed by caustic soda, and then recycled.

- *Final conversion of uranium and plutonium* to end-products :

The uranium solution is concentrated by evaporation, stored, and eventually converted to UF6 for a new isotopic enrichment.

In the same way, the plutonium is precipitated as an oxalate salt by the addition of oxalic acid. This salt is then filtered, dried and calcinated to form the PuO_2 oxide that is used to make the MOX fuel. The mother liquor is concentrated and recycled.

- *Management and treatment of process waste :*

The process waste comprises :

- The hulls, produced during shearing and dissolution operations, which are compacted in a canister, and intended for final disposal.
- The High Activity (HA) liquid waste made up of solutions containing :
  - the insoluble particles from the clarification (fines),
  - the fission products and minor actinides separated during the extraction process,
  - the concentrates generated by evaporation of the aqueous acidic process sewage in an acid recovery unit. Acid generated in this unit is recycled in the process, and distillates, with very low activity, are discharged into the sea.
- The various streams, except the fines, are concentrated and generate the HALW concentrates which are stored in large vessels fitted with cooling and pulsation devices. The concentrates are then mixed with the fines and treated in a vitrification facility to form

a glass matrix with high resistance to leaching. Today, this matrix appears to be the most suitable and safer packaging for long term storage.

- The gases, which are collected according to type and level of activity, washed and treated on specific traps to recover elements such as iodine, and then filtered through high efficiency filters before discharge through a stack.

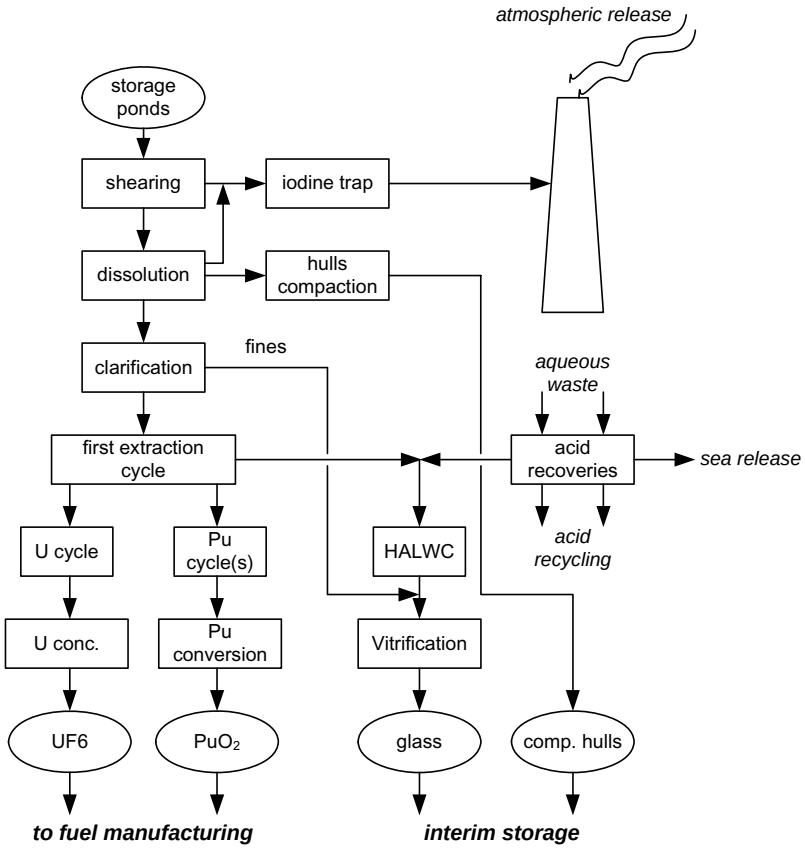

Figure 1 : LWR process scheme at La Hague

Treatment plants such as La Hague are designed to be operated for very long periods. During their lifetime, they will have to reprocess fuel with changing characteristics, although they were primarily designed for fuel from light water reactors. That can include LWR with an increased burn-up, but also different fuels such as RTR which are highly diverse in terms of weight, shape, and composition, and therefore require high flexibility of back-end services. Fortunately, in La Hague, design adaptations can be realised thanks to evolutionary plant design and the significant experience in performing modifications even in the active part of the plant.

3 COMPARISON OF POWER PLANT FUEL AND RTR FUEL

The characteristics of RTR and power plant spent fuel elements are very different as shown in the following table:

Characteristics	Power plant UOX fuel Typical value	RTR fuel Typical value
Length	4 m	1 m
Weight	500 kg	5 kg
Cladding	Stainless steel + zirconium	Al alloy
Enrichment 235U	4%	95%
Burn-up	< 45 MWd/tHm	14 to 680 MWd/tHM

Table 1 : Comparison of power plant and RTR fuel elements characteristics.

The different characteristics of those fuel elements demanded adaptations of the process :

- handling operations to fit with the small dimensions of RTR fuel elements,
- dissolution chemistry to deal with the cladding that is soluble in nitric acid,
- criticality to adapt to the high enrichment of RTR fuel elements.

4 RTR PROCESS DESCRIPTION

4.1 General description

The chosen approach consists of dissolving the RTR fuel element in a dissolution pit in the existing dissolver. The dissolution pit specially design for RTR fuel dissolution is placed in the dissolver instead of an air-lift not needed for RTR process configuration (it aims at recycling hulls in UOX process in the perforated baskets).

The fuel elements are transferred in “taxi” baskets from the pool storage till the maintenance cell of shearing – dissolution facility, which was the most suitable place for loading of fuel elements. The baskets are the one-used to store BWR spent fuel. They have been adapted to the shape and the high enrichment of RTR spent fuel. The fuel elements are stored in a rack in the cell maintenance until they are loaded one by one in the dissolution pit.

Unlike power plant fuel elements, the RTR fuel elements are completely dissolved in nitric acid. The solution of dissolution is circulating between three existing tanks to optimize the volume of each batch of dissolution. The required number of fuel elements in a batch is assessed to reach a suitable final concentration of aluminium to manage the risk of precipitation of aluminium nitrate. During the dissolution, the couple acidity / aluminium concentration is controlled to limit risks of corrosion and precipitation of aluminium nitrate. At the end of a batch the solution is drained from the dissolver and diluted with UOX solution from another dissolution unit of the plant.

The resulting UOX/RTR mixture complies with the qualification for back-end operations at UP3: particularly the uranium enrichment and the plutonium spectrum.

For the chemical elements that were not present (or in traces in) UOX dissolution solution (such as Al), it was checked that they were not changing the efficiency of the process and that their presence was compatible with the products specifications.

4.2 Qualification program

4.2.1 Necessity of qualification program

The RTR treatment affects every process in use at UP3, from the storage fuel to vitrification, including the analytic laboratories and waste processing.

The special characteristics of RTR demanded particular demonstration in each field of studies as shown in the following examples:

Mechanical field: Gravity feeding of the dissolution pit from the maintenance cell had to be demonstrated for a long term use,

Lay-out : The implementation of the pit in the dissolver was a challenge. The space available was just a little bigger than the dimension of the pit.

Chemical field : The risk of cristallisation of aluminium had to be managed properly

Safety, criticality: the management of the risk of criticality in the dissolution pit had to be demonstrated, taking into account the high uranium enrichment.

4.2.2 Steps of the qualification program

The main objectives of the qualification program were to validate dissolver pit concepts, to verify the basic RTR process data for the T1 production environment, and to acquire data demonstrating control of the process.

A major test program was conducted in two phases at the SGN development and testing laboratory (HRB):

- the development testing phase involved dissolving simulated fuel elements in full-scale mock-up of a fully-instrumented pit made out of glass,
- the qualification testing phase involved dissolving simulated fuel in a pilot unit consisting of a full-scale pit and dissolver in an environment as representative as possible of T1 environment.

Those testing phases allows :

- to improve the dissolution pit concept for instance by developing a device allowing the follow-up of fuel loading and dissolution,
- to ensure the efficiency of the process for instance to validate the chemical kinetics of dissolution,
- to manage risks such as the risk of dispersion of fuel material out of the dissolution pit.

5 INDUSTRIAL FEED-BACK OF RTR TREATMENT

Thanks to a good qualification of the process and a training program, the first active dissolution of RTR fuel elements unfolds successfully in June 2005. About 2 tons of RTR UAl fuel have been processed by the end of the year 2006.

In addition of the usual control of the process, special analysis and observations were made during the first batches to ensure the efficiency of the process and gain knowledge on RTR processing.

5.1. Insoluble material

Special analysis have been made to ensure the total dissolution of RTR fuel elements. The results are satisfying as:

- The amounts of residual suspension material in RTR solution are low and are comparable to those in clarified UOX solution.

- At that time no significant amounts of deposit have been observed in the dissolving vessels.

5.2. Corrosion

As expected because of higher nitric acid and temperature, the corrosion rate of stainless steel equipments placed in the dissolving unit is higher than the corrosion during UOX dissolution. Therefore, a special program of measurement is implemented to check the acceptability of corrosion rate. The first measurement of dissolution pit confirms a significant life time for this equipment which has been designed and implanted to be easily replaced.

5.3. Dissolution rate

It appeared that the dissolution rate is higher than expected allowing more flexibility in operation. Thanks to the gain in operating skills for both mechanical and analytic operations, the global rate of dissolution operations allows important margins towards the back-end of the process.

Improvements have been made to gain in operating efficiency and reduce the amount of effluents generated. The dissolution process has been successfully performed on two types of RTR UAl fuels (ANSTO, BR2). AREVA NC will keep on gaining knowledge on RTR UAl fuels treatment in the coming years.

5.4. Back-end operation

The global rate of treatment is limited by the amount of aluminium compatible with high active glass wastes. Research and developments are going on to increase the amount of aluminium coming from RTR dissolution in the glass by decreasing the amount of aluminium oxide coming from non active glass used in vitrification process. The industrialisation of the R&D program is expected in 2008.

6 PERSPECTIVES

The RTR treatment is limited by now to RTR fuel composed of uranium aluminium alloy and small enough to fit with the dissolution pit size. Studies are going on to adapt the process to :

bigger UAl fuel,

- Studies are consisting in finding a way to cut the fuel elements taking into account the criticality risks

USi fuel (fuel made of aluminium alloy mixed with U3Si2 particles)

- the process is adaptable to USi fuel but some issues are still under progress to manage properly the siliceous component in the process and to validate a sufficient treatment rate,

UMo fuel (fuel made of aluminium alloy mixed with UMo particles)

- UMo Fuel development is still under progress at the same time promising research and development about its treatment is made.

5 CONCLUSION

Treatment RTR fuel is an important issue for Research ant Testing Reactors Operators. AREVA NC la Hague plant is now equipped to reprocess some RTR UAl fuel and has begun actual operation since June 2005 with success. This new functionality required studies and qualification to accommodate the different characteristics of the RTR fuel elements in comparison with power plant reactor fuel elements and is now entering in an industrial maturity phase.
Studies are going on to extend the type of RTR fuel that could be reprocessed and to increase the capacity of treatment.

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D.Kolupaev

Reprocessing of Research Reactor Spent Nuclear Fuel at the PA "Mayak"

Paper

The production association "Mayak" was founded in 1948 within the framework of Soviet Union defense program. It's situated in the Ural, in Chelyabinsk region.

Nowadays PA "Mayak" is a big complex of industrial plants and production departments. The enterprise management is consolidated.

PA "Mayak" has several kinds of activity. The main of them are reactors, radiochemical, chemical-metallurgical and radioisotope productions.

The first Russian reprocessing facility, known as RT-1, was started on the radiochemical plant base in 1977. Nowadays RT-1 remains the single reprocessing plant in Russia.

The fact is that at present there is a full-scale reprocessing in France, Grate Britain and Russia. All these reprocessing facilities use similar technological processes, such as: water-pool storage of spent nuclear fuel (SNF), shearing, nitric acid dissolution, extraction by means of Purex-process, vitrification of High Level Wastes (HLW) and others. However, each enterprise has its own technological features, sometimes very different from others.

The main characteristic property of RT-1 is a broad spectrum of reprocessing spent nuclear fuel.

The following spent fuel types are reprocessed here:

- SNF of PWR reactors (WWER-440) and FB reactor (BN-600);
- SNF of transport ship reactors;
- Production reactors SNF;
- Research reactor spent nuclear fuel.

As it was mentioned above, the world-known technological processes are used at RT-1. But there are some distinctive features.

There are the following distinctive features of RT-1 technology:

1 Universality of the three technological lines which allows not only reprocessing of various SNF kinds, but also to implement the combined reprocessing of some types of them.

2 Extraction of neptunium during SNF reprocessing which is used to implement its separate storage and for radioisotope production.

3 Target enrichment of recycled uranium is achieved by mixture of uranium from reprocessing SNF of various kinds.

4 Extraction of various elements (such as cesium, strontium, promethium and etc.) which are used for radioisotope production.

Basic technological processes for research reactor SNF are:

Shipment of SNF. Shipment of research SNF is accomplished by rail in transport packages such as TUK-19, TUK-32, TUK-128.

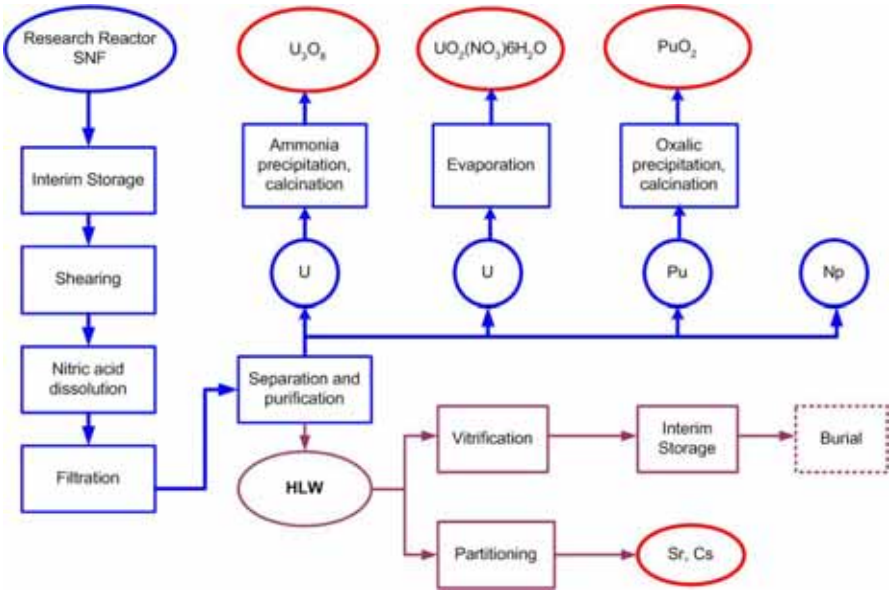
TUK-19

TUK-128

"Skoda" VPVR/M transport packages will be used in future for shipment of spent fuel from the Czech Republic. All above mentioned transport packages have Russian certificates for package design and may be validated in other countries in accordance with safety rules of TS-R-1 (IAEA).

Interim storage of SNF. Transport packages delivered to the plant are unloaded in a hot cell by "dry" way. SNF is placed into the water pool. The layer of water of more than 3m high guarantees reliable biological safety. As a rule, the duration of interim water-pool storage of research reactor SNF is up to 2 years.

Reprocessing. The first stage of reprocessing is shearing of spent nuclear assemblies into pieces of less than 60 mm. After that pieces are dissolved by nitric acid in a cycling dissolver. After filtration the solution of spent fuel is separated by Purex-process into uranium, plutonium, neptunium solutions, and liquid radwastes. Multistage mix-settle extractors with air mixing are used for Purex-process.



Technological scheme

Target products of reprocessing are:

- hexahydrate of uranylnitrate obtained by evaporation of nitric acid uranium solution;

- triuranium octaoxide obtained by means of ammonia precipitation and calcination;

- dioxide of plutonium by oxalic precipitation and calcination.

All recycled uranium is supplied for nuclear fuel production.

Extracted plutonium is placed into a special storage.

Besides above-mentioned products the technological process provides extraction of neptunium and iodine for their isolation.

Krypton (Kr-85), strontium (Sr-90), cesium (Cs-137), americium (Am-241), promethium (Pr-147) and other radionuclides are extracted for radioisotope production.

Safety of radioactive wastes management is one of the most important tasks of RT-1 activity.

Since 1987 vitrification facility has been used in RT-1. The main task is immobilization of high- and middle-level liquid wastes into phosphate glass.

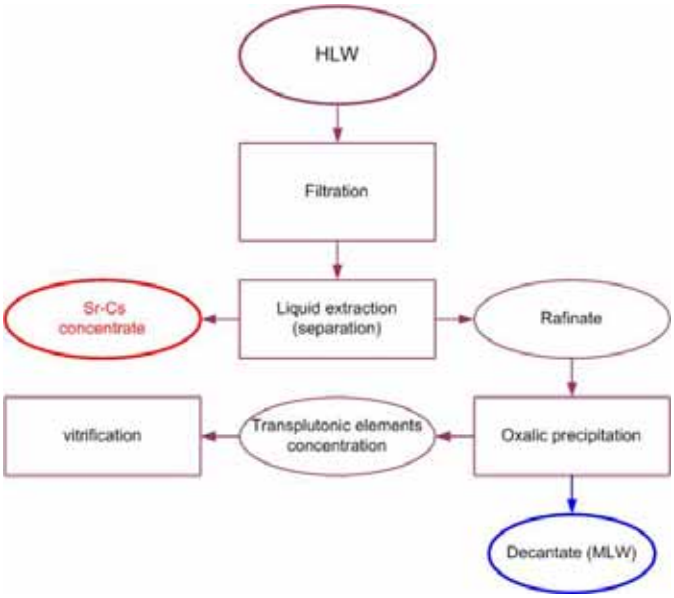
The main technological unit of this facility is a furnace with direct electric heat. Its production capacity is 500 liters per hour of reprocessed solution. This is the continuous action furnace with a working volume of 5,5m^3. The results of vitrification facility operation are shown in the next picture.

№ Furnace	**Period**	**Volume of immobilized HLW, m^3**	**Glass quantity, t**	**Total activity, x 10^6 Ku**
1	1988-1989	1 215	161,2	3,864
2	1991-1997	11 474	2 190,8	282,728
3	2001-2006	7 974	1 818,7	171,957
4	2007-	–	–	–
Total	–	**20 663**	**4 170,7**	**458,549**

The total quantity of immobilized liquid wastes is 20663 cubic meters. At present the 4th furnace is being put into operation.

The first world full-scale partitioning facility was founded at RT-1 in 1996. The main reason of it was that a lot of high level wastes with complicated composition were accumulated. Direct effective vitrification of such wastes was impossible.

Partitioning technology uses processes of strontium and cesium extraction by cobalt dicarbollide as extractant and oxalic precipitation for extraction of transplutonic elements and rare earths. Nominal capacity is 180 liters of liquid wastes per hour. Nowadays partitioning is used for strontium and cesium extraction from high level wastes for the radioisotope production.



Partitioning scheme

Besides the main production activity PA "Mayak" solves problems related to past military activity. Such as:

- treatment of accumulated liquid high-level wastes;
- closing of the technological reservoir named "Karachay";
- decommissioning of old facilities and constructions.

Research reactor SNF reprocessing directly. More than 20 years the RT-1 has been receiving nuclear fuel for processing from research reactors located on the territory of the former USSR, nowadays - Russia. At present on a regular basis SNF is received only from Russian research reactors.

Before disintegration of the USSR SNF of research reactors was taken from Latvia, Uzbekistan, Ukraine, Kazakhstan.

The range of SNF to be processed includes mainly fuel compositions on the basis of aluminium and magnezium. But it can be extended if necessary. For example, the technological process can be adapted to the reprocessing fuel on the basis of metal uranium from "Vinca" institute (Serbia).

It's necessary to say about the problem of shipment and reprocessing of damaged or leaking spent nuclear fuel. Some institutes have a big quantity of leaking fuel because of its long water storage. There are two tasks of safety in this case. The first: safety ensuring during shipment, and the second: safety during interim storage at the reprocessing plant.

The most reliable way to ensure safety at all stages is hermetical canning of damaged (leaking) spent fuel before shipment. It is the most reliable way but rather difficult technically.

Shipment and reprocessing of leaking spent nuclear fuel without hermetical canning will be possible if two safety reports are prepared: for the shipment stage and for the

stage of management at the reprocessing plant. The first is needed for certification of the shipment. The main problem at the reprocessing plant is interim water storage of damaged (leaking) fuel. Based on the safety report, time of the interim storage will be limited if it is necessary.

Shipment and reprocessing of physically damaged fuel may be done only in cans. For example, canning must be provided for SNF from Vinca institute.

Under the auspices of the "Agreement between the Government of USA and the Government of the Russian Federation concerning cooperation for the transfer of Russian-produced research reactor nuclear fuel to the Russian Federation" in the first half of 2006 transportation of research reactor SNF from Uzbekistan was made. At the end of 2006 this fuel was reprocessed. As a result more than 60kg of high-enriched uranium was transformed into the low-enriched category and was prepared for use at nuclear power plants.

The Czech Republic and Latvia are the next countries for shipment of SNF in the framework of the Agreement.

The U. S. Department of Energy / Idaho National Laboratory's Research Reactor Spent Nuclear Fuel Acceptance Program

Jim Wade
Department of Energy – Idaho Operations Office
1955 Fremont Avenue
Idaho Falls, Idaho 83401
wadejr@id.doe.gov

I. PROGRAM 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. 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. 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. 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.

Table 1 provides a listing of all of the facilities included within these programs, while Figure 1 provides a map identifying the countries with TRIGA foreign research reactor facilities that are eligible to participate.

II. 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 1500 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.

I.A. 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. 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 Figure 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.

I.A.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 forms are known as the Fuel and Packaging - Required Shippers Data (F & P RSD). The fuel data is to be included in Section III of the F&P RSD form.

- Second, a team of DOE/INL personnel 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 packaging information and data is also required within Appendix A of the contract, and is included as Section IV of the F&P RSD form.

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 possibly affect receipt, handling and storage at the INL dry storage facility. (The INL has remote video inspection and recording capabilities that have been used in the past for specific fuel inspections, which could be used, as needed, for future inspections as well). 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.

I.A.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.

I.B. 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.

III. 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.

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, Virginia
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 economy countries -	Bangladesh	Indonesia*	Mexico	South Korea*	
	Brazil	Malaysia	Philippines	Thailand	
	Democratic Rep. of Congo	Romania*	Turkey		

*Identifies facilities that have made shipments to the INL

Countries with Spent Nuclear Fuel Eligible for Shipment to the INL

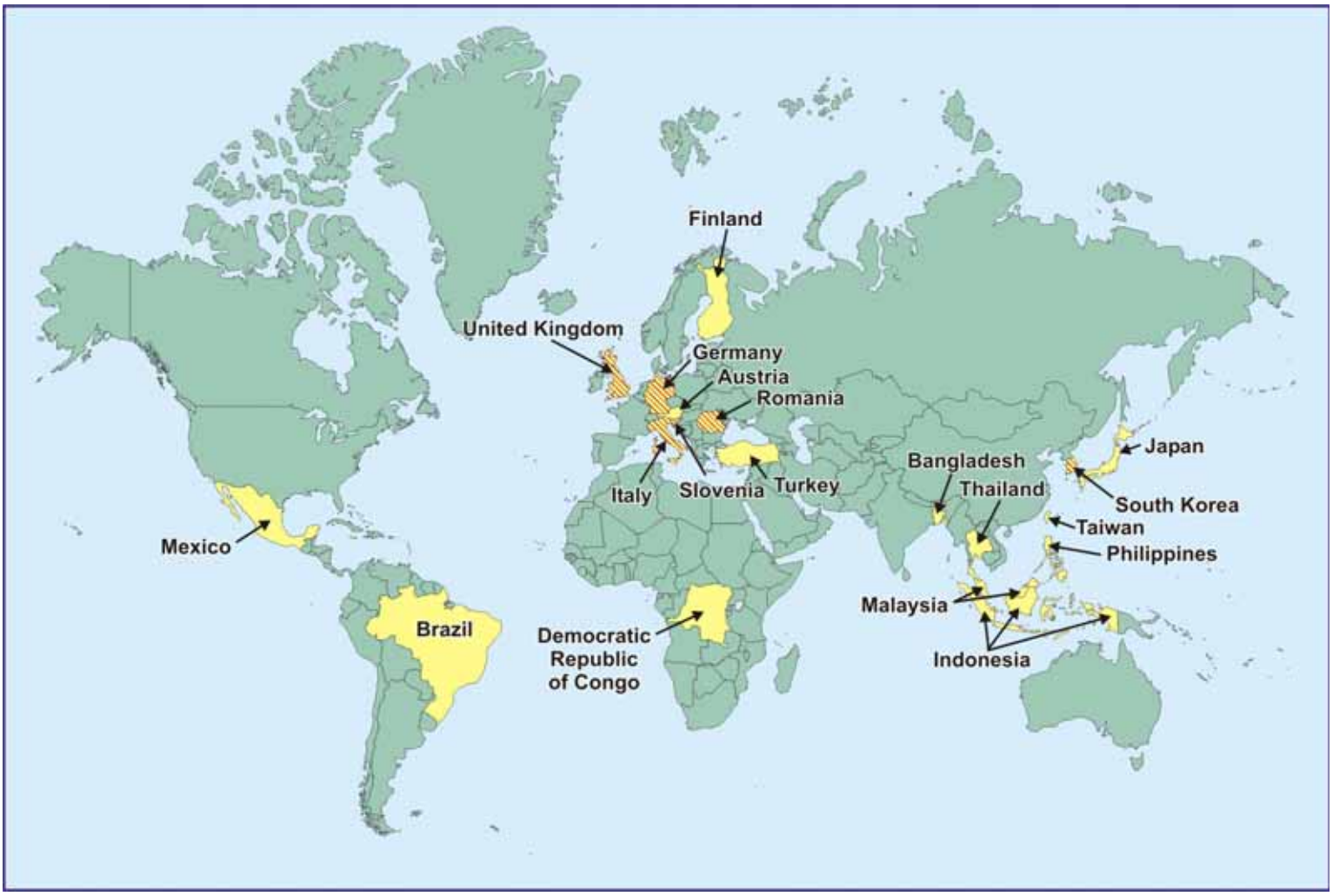


FIGURE 1

INL SNF RECEIPT PREPARATATION FLOW DIAGRAM

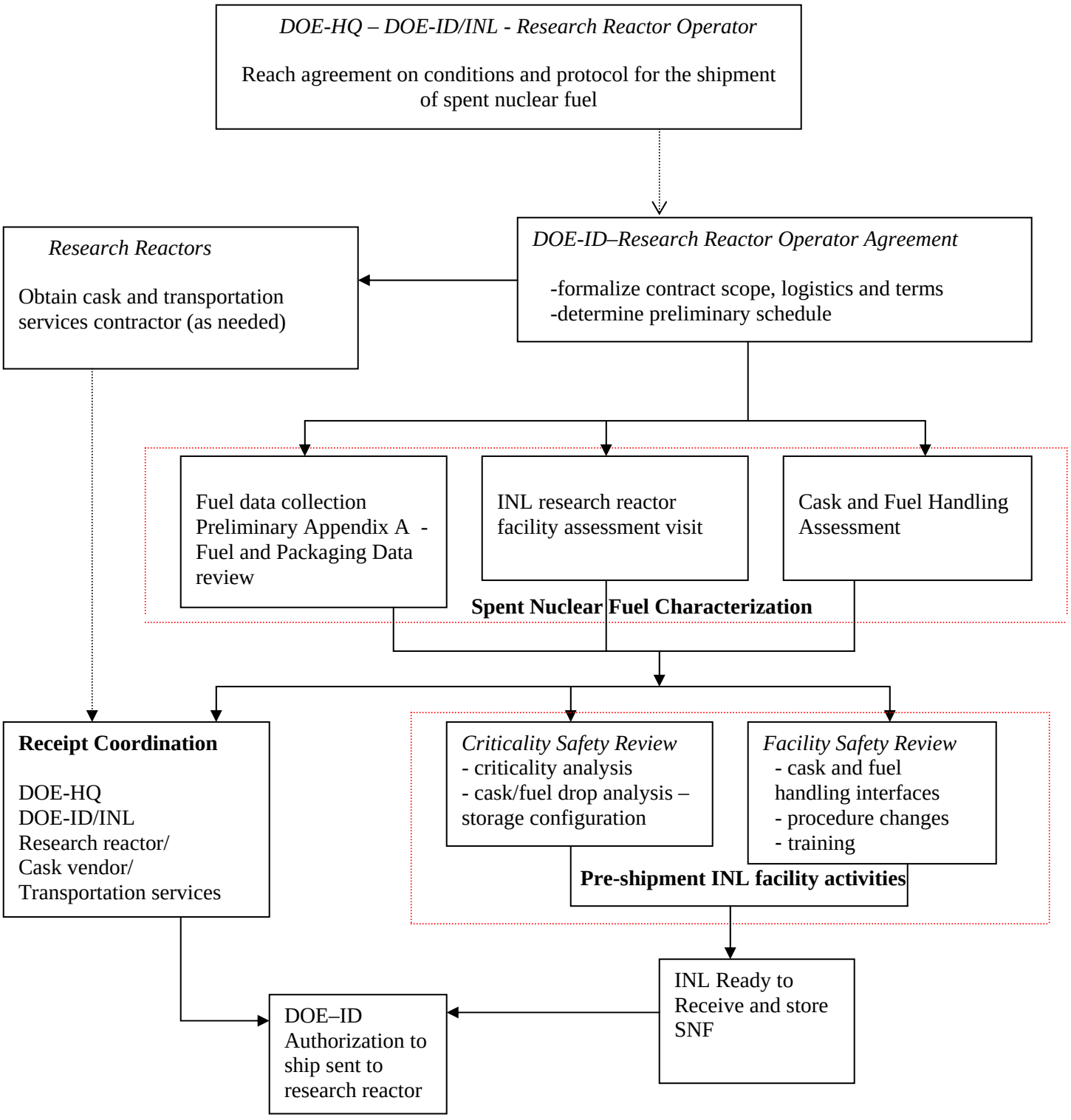


Figure 2

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INVESTIGATIONS TO THE BEHAVIOUR OF RESEARCH REACTOR FUEL ELEMENTS IN REPOSITORY RELEVANT AQUATIC PHASES

H. BRÜCHER, H. CURTIUS
Institute of Energy Research (Safety Research and Reactor Technology, ISR)
Forschungszentrum Jülich GmbH, 52425 Jülich - Germany

ABSTRACT

At the Institute of Energy Research the behaviour of UAl_x-Al and U_3Si_2-Al dispersed fuel elements in repository relevant aquatic phases is under investigation. As repository relevant aquatic phases a salt brine ($MgCl_2$-rich brine 2 solution), granite water (Äspö-type) and clay pore water (Mont-Terri-type) are used. Both fuel types have similar corrosion rates in these repository relevant aquatic phases. Due to the high amount of chloride the highest corrosion rates were obtained in salt brine The corrosion products obtained by a complete corrosion of an UAl_x-Al fuel sample in the salt brine were separated and treated under various geochemical conditions. Mobile and immobile radionuclides were determined. Furthermore one identified crystalline phase component, a Mg-Al-Cl-hydrotalcite was synthesized and sorption investigations with different repository relevant radionuclides were performed in brine 2 solution. Selenium in the anionic form as selenite and cationic radionuclide species as the trivalent americium, the trivalent europium and the tetravalent thorium were shown to be retarded.

1. Introduction

At present three research reactors, the FRM-II-reactor (München), the FRG-I-reactor (Geesthacht) and the BER-II-reactor (Berlin) with a thermal output higher than 5 MW are in operation in Germany. In these reactors dispersed U_3Si_2-Al-fuel is used. In general, three possible back-end options for irradiated research reactor fuel elements exist in Germany [1]: first, irradiated research reactor fuel elements of USA origin can be sent back when these fuels were taken out of the reactor not later than May 2006. This affects basically the dispersed metallic UAl_x-Al-fuel, which was used in the FRJ-II-reactor (Jülich) till May 2006 . Secondly, reprocessing in Great Britain or France is possible and it will performed with respect to economical reasons. Thirdly, dry interim storage and later on, direct disposal in deep geological formations has been taken into account too. As deep geological formations salt, granite and clay are considered. In granite and clay formations pore water is always present whereas in a salt repository a water ingress and subsequent formation of salt brines is considered as accident scenario.
At the Institute of Energy Research (Safety Research and Reactor Technology, ISR) investigations to the behaviour of research reactor fuel elements in these repository relevant aquatic phases are performed. In November 2003 a third project, funded by the federal Ministry of Economics and Technology had started and three main work packages were established:

- Leaching tests with non-irradiated and irradiated dispersed UAl_x-Al- and U_3Si_2-Al-fuel types in granite water (Äspö-type), clay pore water (Mont-Terri-type) and in $MgCl_2$-rich salt brine (brine 2) in order to study their corrosion behaviour
- Treatment of the secondary phases, formed by corrosion of an dispersed UAl_x-Al-fuel type in brine 2 under various geochemical conditions in order to distinguish between mobile and immobile radionuclides
- Synthesis of an identified crystalline phase, a Mg-Al-Cl-hydrotalcite, in order to study the retardation of repository relevant radionuclides.

The results from these work packages are presented in this paper [2].

2. Experimental set-up

Leaching experiments with the non-irradiated UAl_x-Al- and U_3Si_2-Al-fuel elements samples were performed in glass autoclaves at 90°C in 400 ml of $MgCl_2$ brine solution, in clay pore water (Mont-Terri-type) and in granite water (Äpö-type) in presence of 10 g $FeCl_2$ (iron it the basic material of the fuel container) under anaerobic conditions. The dimension of the UAl_x-Al-samples was 28 x 20 x 1.5 mm with an amount of 0.120 g U total (U-235-enrichment: 89%). The dimension of the U_3Si_2-Al-sample was 40 x 20 x 1.36 mm with an amount of 1.44 g U total (U-235-enrichment: 0.21%). Similar leaching experiments were performed with irradiated UAl_x-Al- and U_3Si_2-Al-fuel elements samples. The dimension of the UAl_x-Al-sample was 40 x 20 x 1.36 mm with an amount of 0.25 g U total (U-235-enrichment: 80%). The dimension of the U_3Si_2 sample was 40 x 20 x 1.36 mm with an amount of 1.704 g U total (U-235-enrichment: 19,75%).
In order to distinguish between mobile and immobile radionuclides, trapped by the corrosion products, the geochemical conditions were varied. Corrosion products used for these studies were obtained by complete corrosion of an irradiated UAl_x-Al-fuel element sample (40 x 20 x 1.36 mm) in $MgCl_2$-rich brine in the presence of iron under anaerobic conditions at 90 °C. In order to study the influence of dilution, aliquots were treated with “fresh brine 2” solution. The influence of the ionic strength was studied by the use of different concentrated brine 2 solutions. The fraction of inventory was determined by complete dissolution of some aliquots in 8 M HNO_3 solution. All samples obtained were analysed radiometrically.
The experimental set-up for the syntheses of the Mg-Al-Cl-hydrotalcite is described in [4]. The radioactive selenium, americium, europium and thorium solutions used were prepared from standard stock solutions. Sorption experiments were performed according to the batch-technique. All samples were stored in glass tubes with occasional shaking under argon-atmosphere for two days. Then all samples were filtered (450 nm) and the pH was measured. Aliquots of the solutions were analysed radiometrically. The solids were washed, dried and analysed by XRD and FT-IR. Blank experiments were performed too, indicating that the sorption on the glass walls was negligible.

3. Results and Discussion

Topic 1
First, the corrosion behaviour of non-irradiated dispersed UAl_x-Al- and U_3Si_2-Al-fuel elements specimens in repository relevant solutions was investigated. In Figure 1 and Figure 2 the hydrogen production for both fuel-types in brine 2 and in clay pore water (Mont-Terri-type) are presented. The functions of the hydrogen formation in granite water (Äspö-type) are comparable to the clay pore water.

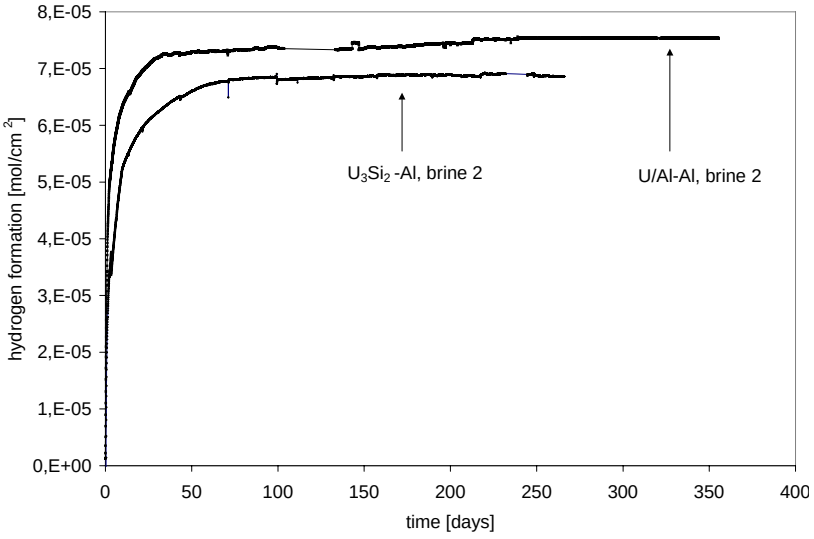

Fig. 1 Hydrogen formation in brine 2 solution

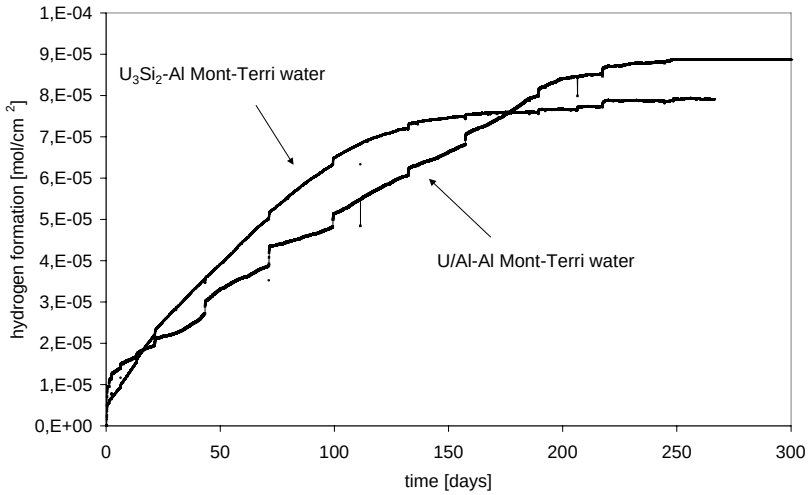

Fig. 2 Hydrogen formation in Mont-Terri water

In brine 2 the hydrogen formation is terminated after 100 days. This indicates the complete corrosion of the fuel. In granite and clay pore water the corrosion processes are terminated after 250 days. Nevertheless in view of the time periods of final disposal the assumption can be drawn, that in these repository relevant aquatic phases both fuel types corroded instantaneously. The detected maximal hydrogen formation normalized to the fuel surface was between 7.0E-05 and 8.5 E-05 mol/cm^2. Furthermore in brine 2 higher corrosion rates were obtained. In brine 2 more chloride is present which causes a low pH value. Under these conditions the aluminium dissolution and the formation of local elements, which accelerate the corrosion, are favoured. The matrix elements, aluminium, uranium, and silicium were detected only in the secondary phases formed. Similar experiments were performed with irradiated fuel elements. First results will be obtained in the next months.

Topic 2

The secondary phases obtained by complete anaerobic corrosion of an irradiated UAl_x-Al-fuel element in brine 2 were treated under various geochemical conditions particularly under aerobic conditions.

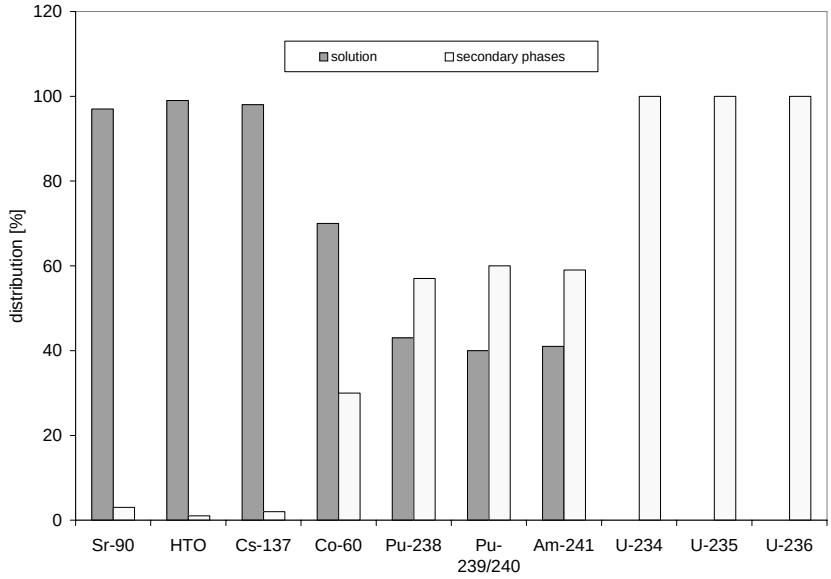

Fig. 3 Radionuclide distribution between solution and secondary phases after complete corrosion of an UAl_x-Al fuel element sample in $MgCl_2$-rich brine

First the radionuclide inventory was determined and a radionuclide distribution between secondary phases and solution was obtained (Figure 3). The main mobile radionuclides are Cs, Sr, and Tritium, while the radionuclides Pu, Am, and U are immobile.
Then aliquots of the secondary phases were taken and treated with "fresh" brine 2 in order to study the dilution effects. Other aliquots were treated with different concentrated brine 2 solutions in order to study the influence of the ionic strength. The results indicate clearly, that under aerobic conditions Am and U must be regarded mobile as well. This can be explained mainly by the change to aerobic conditions. Oxygen causes the oxidation of U-IV to U-VI and these U-VI components are lightly soluble. The higher solubility of Am-III can be explained by the formation of lightly soluble Am-III-carbonyl-complexes. The behaviour of Pu is different. It seems that Pu already formed polymeric oxy or hydroxy-complexes which are nearly insoluble.

Topic 3
In view of the actinides the secondary phases have a significant influence on radionuclide retardation. After the complete corrosion of an UAl_x-Al-fuel element sample in brine 2 a Mg-Al-Cl-hydrotalcite was identified as a crystalline phase-component in the secondary phases formed [3]. A synthesis was performed and the obtained solid was analysed. From these results the formula of the Mg-Al-Cl-hydrotalcite can be derived as: $Mg_3Al(OH)_8Cl_{0.88}$ $(CO_3^{2-})_{0.063}$ $2.4 \cdot H_2O$. Due to the structure of this Mg-Al-Cl-hydrotalcite an anion exchange in the interlayer should be possible. We performed sorption experiments according to the batch technique and for the monovalent anionic species, iodine [4] and pertechnetate no retardation in brine 2 solution was achieved, but for the divalent anionic specie selenite, a retardation was gained.
For cationic species two reactions are feasible: sorption and incorporation. Results from sorption experiments did show that the trivalent cationic species, Am and Eu had a similar sorption behaviour. Then the sorption data were fitted to the Dubinin-Radushkevich (D-R) equation [5] and the mean energy of sorption was calculated. The mean energy of sorption is the free energy change when one mole of ion is transferred to the surface of the solid from infinity in the solution. The energy calculated for Am was 11.8 kJ/mol and for Eu a value of 11.18 kJ/mol was determined. Both values were in the range for ion exchange reactions, i.e., 8-16 kJ/mole. For thorium, a tetravalent cationic specie, the D-R-Plots with two different hydrotalcite concentrations are shown in Figure 4.

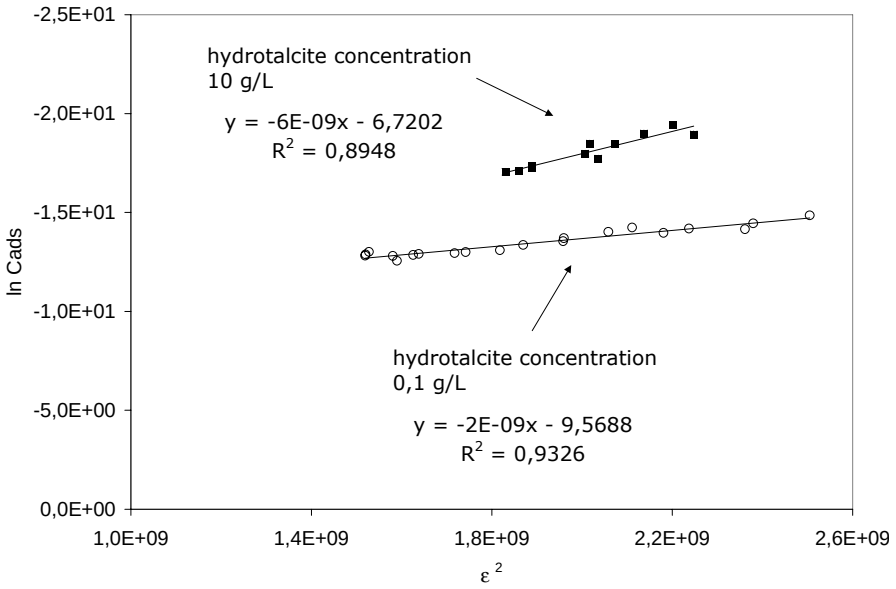

Fig.4 D-R-plots for the sorption of thorium onto the Mg-Al-Cl-hydrotalcite

For a hydrotalcite concentration of 10 g/L the mean energy of sorption was 9.13 kJ/mol and a value of 15.8 kJ/mol was obtained for the hydrotalcite concentration of 0.1 g/L. In both cases sorption occurs via an ion exchange mechanism.

4. Conclusions and outlook

From the results of the present work the following main points can be summarised:

- Metallic UAl_x-Al dispersed fuel as well as the U_3Si_2-Al dispersed fuel corrodes rapidly in salt brine (brine 2), in granite water (Äspö-type) and in clay pore water (Mont-Terri-type) instantaneously. The matrix elements aluminium, uranium and silicium are quantitatively found in the secondary phases formed by corrosion. The hydrogen formation created in these repository relevant aquatic phases was determined to be in the range of 7.0E-05 and 8.5E-05 mol/cm^2.

- Secondary phases, formed by corrosion of an irradiated UAl_x-Al-fuel element in salt brine, were separated and treated under various geochemical conditions. Mobile radionuclides are Cs, Sr and Tritium. The remobilisation of uranium and americium is caused by the aerobic conditions which accelerate the formation of soluble U-VI compounds and soluble Am-III-complexes. The actinide Pu is not affected and we assume the formation of polymeric oxy or hydroxy-complexes which are slightly soluble.

- In the leaching experiments using salt brine, one crystalline phase component of the corrosion products, a Mg-Al-Cl-hydrotalcite was synthesised and it was shown that selenium, americium, europium and thorium were retained by means of ion exchange reactions in brine 2 solution. Under final disposal conditions in a salt repository these elements will be retarded by the Mg-Al-Cl-hydrotalcite. No retardation was given for iodine and pertechnetate in brine 2 solution. Towards the hydrotalcite the affinity of chloride as competition anion is higher.

In our future work leaching experiments with irradiated fuel elements in hot cell facilities will be continued and after complete corrosion the radionuclide distribution between solution and secondary phases will be determined for both fuel types in the three repository relevant aquatic phases. Secondary phases obtained after complete corrosion of the non-irradiated dispersed fuel elements will be analysed for elemental composition. Furthermore the identification of all present crystalline phases will be aspired. Besides sorption, radionuclide retardation may be achieved by incorporation. The incorporation of various radionuclides in the lattice structure of the Mg-Al-Cl-hydrotalcite will be another main topic of our future work, because this radionuclide retardation can be regarded as an irreversible process.

5. Acknowlegment

This work is funded by the BMWi (Förderkennzeichen: 02E9803).

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CORROSION OF SPENT RESEARCH REACTOR FUEL: THE ROLE OF SETTLED SOLIDS.

L.V. RAMANATHAN
Materials Science and Technology Center, Instituto de Pesquisas Energéticas e Nucleares (IPEN)
Av. Prof. Lineu prestes 2242, Cidade Universitaria, São Paulo - Brazil

R.E. HADDAD
Materials Dept.,Constituyentes Atomic Centre, National Atomic Energy Commission (CNEA)
Av. Gral Paz 1499, (B1650KNA) San Martín, Buenos Aires – Argentina

P. ADELFANG
Nuclear Fuel Cycle and Materials Section, International Atomic Energy Agency (IAEA)
Wagramer Strasse 5, P.O. Box 100, A-1400, Vienna - Austria.

ABSTRACT

Reactor components or fuels that have remained immersed in a research reactor pool or in a spent fuel storage basin often reveal solids on their surfaces. The sources of these solids are many and include air borne dust, corrosion products and precipitated salts. Results of the IAEA coordinated research project (CRP) “Corrosion of Al-clad Spent Fuel in Water” and the Regional Project for Latin America (RLA) revealed that settled solids contribute to corrosion of Al coupons. Further studies carried out at different sites worldwide to determine the rates of settling of solids, their composition and influence on corrosion of aluminium alloys revealed that the solids consisted mainly of oxides of aluminium, iron, silicon and calcium. Other constituents in the settled solids were site-specific. Short term laboratory tests in which solids with specific composition were positioned on aluminium substrates in relatively pure water revealed their role in initiating pitting and/or crevice type of corrosion.

1. Introduction

Reactor components or fuel assemblies that have remained immersed in a research reactor pool or in a spent fuel storage basin for a reasonable length of time have loose solids on their surfaces. The quantity and nature of these solids vary. These solids settle on all surfaces inside pools or basins. Most of these solids eventually settle at the bottom of the pool or basin and are the main constituent of sludge. The origin or sources of these solids are mainly air borne dust, corrosion products and precipitated salts. Airborne dust in the reactor hall or in the spent fuel basin (SFB) hall settles on reactor pool or SFB surfaces. The dust (fine solid particles) on the pool/basin surface floats until wetted by the water. Surface skimmers in reactor pools remove most of the floating dust. However, depending on properties of the solid, mainly density, it settles at the bottom of the pool or on any surface that it encounters as it descends through the pool or basin water. Solids settle faster in stagnant regions of the pool. Fine solid particles also have a tendency to agglomerate to form larger particles which settle faster. This tendency varies with the composition and density of the particles. The corrosion products on metallic surfaces in contact with flowing water are easily detached and carried in the circulating water, often returning, depending on its size, to the reactor pool or basin. Another source for solids is precipitated solids. Reactor hall operations such as: (a) movement of cranes; (b) opening of doors; (c) immersion of inadequately dusted components in the reactor pool; (d) shifting of immersed components are other sources of settled solids. In away-from-reactor fuel storage facilities the above mentioned scenes plays out to a greater extent.

Experimental work carried out in the IAEA coordinated research project (CRP) “Corrosion of Aluminium-clad spent fuel in water” and the Regional Project for Latin America (RLA) included the exposure of horizontal and vertical coupons of aluminium (Al) alloys to spent fuel basins in many countries. [1] Comparison of the corrosion behaviour of these coupons revealed that horizontal coupons corroded considerably more than vertical coupons. The top surfaces of horizontal coupons corroded more than the bottom surfaces of the same coupon. These observations lent proof to the dominant role of settled solids on the corrosion of Al coupons and by extension on the corrosion of Al-clad spent research reactor fuel. [2] Further studies were undertaken to determine the: (a) amount of settled solids and the rate of settling of these solids at the various sites; (b) composition and/or constituents of the solids; (c) role of specific settled solids on the nature and extent of corrosion of Al alloys. This paper presents details of these studies.

2. Methods and materials

In this study, a stainless steel or aluminium recipient (henceforth referred to as the collector) was used to collect settled solids at 11 different sites where the two IAEA projects were being carried out. This collector was positioned in the vicinity of racks with corrosion test coupons. The collectors remained in the research reactor (RR) or SFB for periods ranging from 4-6 months. After withdrawal of the collector, the water was filtered; the solids collected on a filter paper were dried at 100º C for 24 hours and weighed. The amount of solids and the rates of settling were determined. Representative specimens of the solids were examined in the SEM and/or analysed using one or more of the following techniques: (a) energy dispersive spectroscopy (EDS), (b) x-ray diffraction (XRD), (c) x-ray fluorescence (XRF), (d) mass spectrometry. Fig. 1 shows a collector and settled solids in one of the collectors.

(a) (b)

Fig. 1. (a) AISI 304 stainless steel collector used at CChen, Chile. (b) Settled solids in a collector

3. Results and discussion

The rates of sedimentation of the solids at different sites and their composition and/or constituents are shown in Table 1. The solids consisted mainly of oxides of aluminium, iron, silicon and calcium. This indicated that the sources of the solids were construction debris, atmospheric dust and corrosion products of aluminium alloy and steel components in the pool or basin. Other constituents in the settled solids were site-specific.

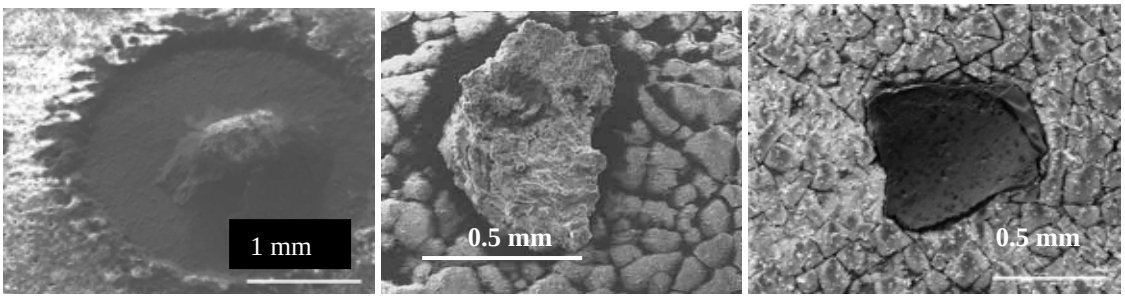
Test site	Rate of sedimentation ($\mu g/cm^2$/month)	Composition or constituent of settled solid
RECH-1, Chile.	58.6	Al; SiO_2 (quartz);SiO_2 (cristobalite) $(Ca,Na)(Si,Al)_4O_8$ (anortite)
RECH-2, Chile.	1.5	SiO_2 (quartz)
CDTN, Brazil.	18.1	Amorphous $CaCO_3$ (calcite); Fe_3O_4 (magnetite), SiO_2 (quartz); $CaMg(CO_3)_2$ (dolomite)α-$Al_2O_3.3H_2O$ (gibbsite), Fe_2O_3 (hematite) $Mg_3Si_4O_{10}.(OH)_2$ (talc)
ININ, México.	17.666	Iron oxides (hematite, magnetite), aluminosilicates, sodium and calcium carbonates
România.		Fe, Zn and Cd
SF pool, Thailand.	195	Fe, Al, SiO_2, Cu, Ba; Light particles of plant tissue, bio-mass floating in water with some fine dust.
Reactor pool, Thailand.	141	Fine light brown dust. Main component is iron.
Cooling pool, Kazakhstan.	36.000	Deionizing column resin
Water storage, Kazakhstan.	12.000	Deionizing column resin
RA3 Decay Pool, Argentina	19.5	Silicon oxide and combined oxides of Si, Al and Fe
IPEN, Brazil.	178.2	56.79 Al_2O_3, 21.04 SiO_2, !4.93 Fe_2O_3, 2.35 CaO, 1.6 Cr_2O_3, 0.76 TiO_2, 0.6 NiO (wt%)

Tab. 1. Rates of sedimentation of solids and their composition or constituents.

3.1. Effect of settled solids on corrosion of aluminium alloys

Aluminium and its alloys normally corrode in aqueous solutions with low or high pH. Pitting is the main form of corrosion and aluminium alloys are prone to pitting corrosion in solutions with not only chloride ions but also other anions like sulphates and nitrates. Aluminium alloys do not generally pit in environments totally free of aggressive ions. Nevertheless, Al alloys have been observed to pit in very low conductivity water in the presence of solids on the surface. This solid particle-induced corrosion of aluminium could be due to one or more of several reasons: 1) The alkaline nature of the solid or leaching of alkaline products from the solid; 2) Formation of crevices under the solid and consequent crevice corrosion in the presence of aggressive ions; 3) The solids if conducting, become the cathode for the cathodic reaction, result in localized pH increase and consequent metal dissolution.

Reason (1) is considered to be operative if the solid is a concrete particle. Fig 2a reveals the surface of an AA 6061 alloy with a concrete particle and exposed for 40 days to nuclear grade demineralised water. [3] Aluminium alloys are also susceptible to crevice corrosion, which occurs in the presence of aggressive ions. [4] In pure water, this form of corrosion sometimes occurs in the presence of aggressive ions originating from the sediment itself. Figs 2b and 2c show corrosion of AA 6061 alloy with hematite and magnetite particles on the surface. These figures reveal that the crevice corrosion mechanism could be operating at the microscopic level. The alloy surface around the hematite particle is corroded, where as the surface around the magnetite particle is unaffected. [5]



(a) (b) (c)

Fig. 2: Corrosion of AA 6061 alloy exposed for 40 days to nuclear grade demineralised water: (a) with a concrete particle; (b) with a hematite particle; (c) with a magnetite particle.

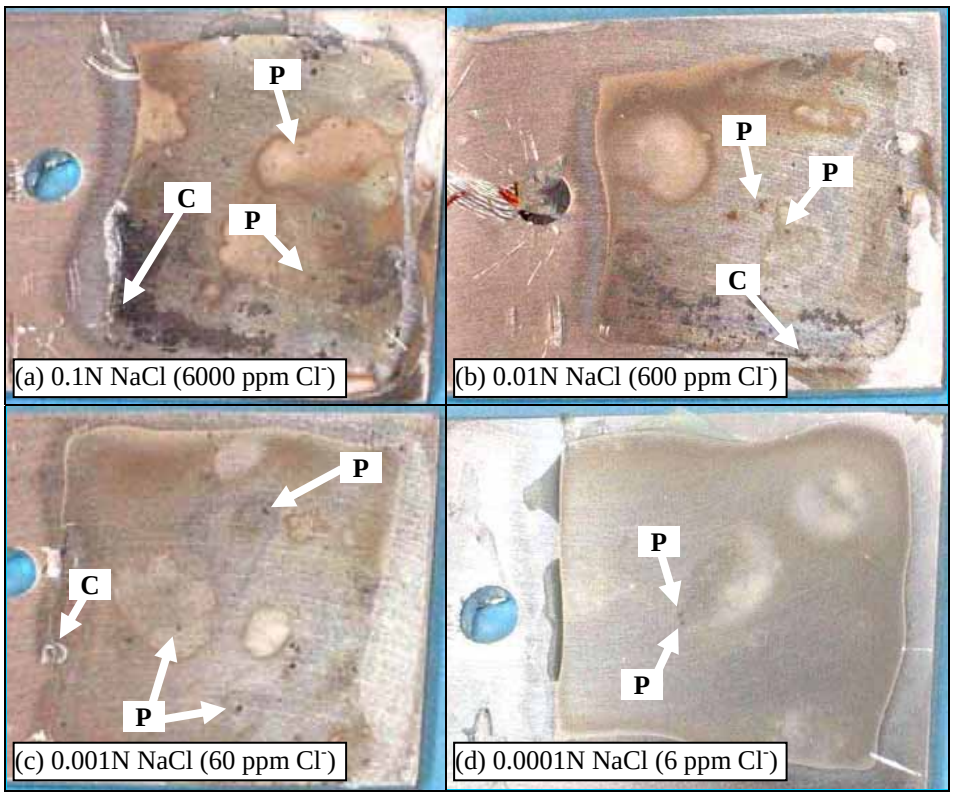


Fig. 3: Appearance of AA 6061 sample surfaces with hematite sediments after 7 days of exposure to NaCl solutions of different concentrations. Pitting is indicated with 'P' and crevice corrosion with a 'C'. The stains produced by the sediments are still visible.

The sediment induced pitting produced in the laboratory resembles, to some extent, observations made on Al surfaces exposed to SFB waters. However the pits were not as deep as those observed on certain spent fuels stored for extended periods. To evaluate the short term influence of sediments on the corrosion of Al, immersion tests were carried out in waters similar to that in SFBs. In these tests, particles of hematite, magnetite and glass were deposited on aluminium surfaces exposed to sodium chloride solutions of various concentrations.

3.2. Effect of short term exposure of deposited solid particles on Al alloy corrosion

The Al surface with hematite particles that was exposed for 60 days to a solution with 40 ppm of chlorides did not reveal pits but some stains. However, a similar surface coupled to stainless steel revealed pits after only 7 days. The results also indicated the marked effect of chloride ion concentration as shown in Fig 3. In waters with high chloride ion concentrations, both pitting and crevice corrosion was observed. Pits formed both under the deposits and on the free surface. In the solution with 0.001N NaCl, very few pits formed at regions away from sediments. In 0.0001N NaCl all the pits were under the particles. The Al surface with magnetite particles and exposed for 7 days to 0.001N NaCl (60 ppm chloride) revealed pits only around particles that had transformed from magnetite to hematite. Glass pieces were used to simulate inert particles. The Al surface with glass pieces was exposed for 7 days to 0.001N NaCl and it revealed pits that could be observed through the glass.

4. General discussion

Although these tests were of short duration, the results demonstrate that spent fuel elements that are stored for extended periods can undergo these forms of environment assisted degradation. It is quite probable that corrosion occurs in what is considered to be innocuous environments and sediments could trigger the onset of pitting. The results also demonstrate the deleterious synergistic effect of galvanic contacts and sediments. These conditions facilitate the onset of corrosion processes, which in more dilute environments are prone to occur beneath the deposits. Similar conditions are encountered in some spent fuel storage sites in which fuel bundles are positioned inside steel structures and without proper electrical isolation.

5. Recommendations

The composition of the settled solids at a specific site helps indicate the possible source. Knowing the source, steps can be taken to reduce or eliminate it. The actions are usually site-specific. However, some general recommendations can be made to reduce or eliminate airborne dust from settling on RR pool or SFB water surface. These include:

1. Increase in efficiency and/or frequency of water circulation through a filter.
2. Use of a skimmer and filter system.
3. Increase in water flow in the vicinity of stored spent fuel.
4. Use of stainless steel doors or lids for away-from-reactor storage basins.
5. Use of adequate corrosion protection schemes, if plain carbon steel doors or lids are used.
6. Installation of double doors, in case of a dusty atmosphere just outside a SFB.
7. Vacuuming of all surfaces in reactor hall and inside the pool, in the case of in-reactor basins.
8. Improved air circulation and pumping of filtered air into the reactor hall.
9. Reduction in water turbulence in the SFB.

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INTERNATIONAL TOPICAL MEETING ON RESEARCH REACTOR FUEL MANAGEMENT (RRFM) - UNITED STATES FOREIGN RESEARCH REACTOR (FRR) SPENT NUCLEAR FUEL (SNF) ACCEPTANCE PROGRAM: 2007 UPDATE

C. E. Messick, J. L. Taylor
Foreign Research Reactor Spent Nuclear Fuel Acceptance Program
U.S. Department of Energy, National Nuclear Security Administration
Office of Global Threat Reduction, Washington, D.C. 20585—United States of America

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 eleventh year of implementation, has completed 37 shipments to date, safely and successfully, and another is expected to be completed soon. Twenty-seven countries have participated so far, returning a total of 7,620 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 nine (29) of the 37 shipments contained aluminum-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 seven (7) shipments were placed into storage at the Idaho National Laboratory (INL). The most recent shipment

was completed without incident, arriving at SRS on February 6, 2007. During the remaining calendar year (January - December 2007), the program is planning to receive as many as five shipments of SNF from various locations.

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.

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. 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 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.

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. The GTR Acceptance Program staff will be happy to answer questions about scheduling or clarify what type of information is needed to facilitate receipt of fuel.

As requested by many FRRs the program was extended to allow additional time for further development of 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. These delays are hurting DOE's ability to continue normal planning and to maintain adequate resources for the receipt facility. The FRRs are strongly encouraged to continue shipping as early as possible and maintain original schedules and plans as closely as possible. Deferring shipments when spent fuel is available for shipping could result in changes designed to improve shipping decisions.

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. We 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 optimize 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 aluminum 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 aluminum 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.

7. 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)

AUSTRALIAN NUCLEAR SCIENCE AND TECHNOLOGY ORGANISATION (ANSTO) AND NAC INTERNATIONAL

AREA: FUEL BACK-END MANAGEMENT

PREPARATION AND PERFORMANCE OF THE LARGEST EVER SHIPMENT OF IRRADIATED HEU FUEL ELEMENTS UNDER THE FRR PROGRAM: THE 2006 ANSTO HIFAR SPENT FUEL TRANSPORT FROM SYDNEY TO THE UNITED STATES

Kaye Hart, Lubi Dimitrovski and Michael Anderson (ANSTO)
Catherine Anne and Jamie Adam (NAC International)

RRFM 2007
Lyon, France

ABSTRACT

Just over six years after signing the contract between ANSTO and INVAP for the design and construction of a new 20 MW multipurpose research reactor, Australia's new research reactor, OPAL, went critical for the first time on August 12, 2006. The HIFAR reactor continued to operate in parallel - until officially shut down on 30 January 2007 in order to ensure continuity of operation during the transition.

The management of spent fuel remains a very important aspect of the operation of research reactors for ANSTO. For disposition of UK-origin spent fuel arising from the operation of the HIFAR reactor, ANSTO initially elected to ship the irradiated fuel assemblies to the UKAEA in Dounreay, Scotland. With the closure of Dounreay, alternatives were evaluated and reprocessing of the spent fuel at the La Hague reprocessing plant was selected as the option for the disposition of ANSTO's UK origin spent fuel. Between 1999 and 2004, a total of 1288 fuel assemblies were sent in four shipments to the La Hague reprocessing plant. For the remaining HIFAR fuel assemblies containing U.S. origin uranium, ANSTO decided to exercise its option to return the fuel assemblies to the U.S. under the Foreign Research Reactor Spent Nuclear Fuel (FRR SNF) Acceptance program. This paper will describe the organization and performance of the latest and largest shipment of HEU fuel elements under the FRR program from the HIFAR reactor in Sydney to the U.S. DOE's Savannah River Site (SRS) in South Carolina.

PAPER

Introduction: Background HIFAR

HIFAR was Australia's only multi-purpose research reactor for the past 50-years and has operated safely and effectively over this time. Over the past five years ANSTO has built a replacement research reactor – OPAL – which is now ready to take over from HIFAR in supplying neutrons for research, for industry, and for making nuclear medicine. There is, therefore, no further need for HIFAR and, accordingly, it was shut down on 30 January 2007 and will now undergo a staged decommissioning process.

HIFAR was originally built to test materials for use in future power reactors. The idea was to test materials in HIFAR by subjecting them to an intense neutron flux; a relatively quick assessment could be made of their suitability for use in a power reactor. With the decision not to pursue a power reactor program in Australia, there was a gradual change in how the reactor is used over the years.

The construction of HIFAR commenced in February 1956; it first went critical on 26 January 1958, and routine full power operation commenced in January 1960.

HIFAR is a copy of the DIDO reactor in the United Kingdom, which was, in turn, modelled after the CP5 reactor built near Chicago.

	CP5	**DIDO**	**HIFAR**
First went critical	February 1954	November 1956	26 January 1958
Thermal power	5 MW	Originally 10 MW Finally 25.5 MW Shutdown 3/90	10 MW
Peak thermal neutron flux ($n/cm^2/s$)	10^{14}	2.3×10^{14}	1.4×10^{14}

The HIFAR reactor had 25 fuel elements containing enriched uranium encased in aluminium. Every four weeks, three of these fuel elements were replaced and are deemed "spent fuel". HIFAR uses around 37 fuel elements annually.

HIFAR's spent fuel elements are each 600 mm long and 100 mm in diameter.

Some fuel used in HIFAR was manufactured using uranium enriched in the U. S. Such fuel, when spent, can be repatriated to the U. S. under the Foreign Research Reactor (FRR) acceptance program, where the U.S. Government takes ownership of it. The U.S. Government is then responsible for the safe storage and disposition of this material. No waste arising from the storage or handling of this spent fuel is returned to Australia from the U. S.

ANSTO also operated an Argonaut type 100 kW reactor (MOATA) from 1961 to 1995 when it was finally shut down. All of the fuel plates (HEU) have been removed, re-configured into shipment assemblies and returned to the US with the current 2006 spent fuel shipment.

Since commencement of the HIFAR operations there have been eight spent fuel shipments overseas totalling 2,142 Spent Fuel Elements (SFEs), as summarised below:

- 1963 Dounreay (UK) 150 SFEs
- 1996 Dounreay (UK) 114 SFEs
- 1998 US (Savannah River Site) 240 SFEs
- 1999 COGEMA (France) 308 SFEs
- 2001 COGEMA (France) 360 SFEs
- 2003 COGEMA (France) 344 SFEs
- 2004 COGEMA (France) 276 SFEs
- 2006 US (Savannah River Site) 330 SFEs

Shipment Preparation

The initial discussions between ANSTO and NAC International (US company specialized in nuclear materials transport, spent fuel management technology and nuclear fuel cycle consulting) took place in 2004. By mid 2005, an initial contract was in place. NAC would provide three NAC-LWT casks capable of transporting 126 fuel elements, as well as designing, licensing and manufacturing two new sets of baskets. In mid 2006, the contract was amended to include four additional casks (corresponding to a total of 204 fuel elements in the same shipment campaign) and also to integrate the transportation services for the seven empty and loaded casks to and from the Port of Sydney, Australia. The shipment consisted of five NAC-LWT casks containing 42 fuel elements each and two TN 7/2 casks each containing 60 fuel elements each representing a total of 330 fuel elements. This was the largest shipment of HEU fuel elements removed from one single facility under the FRR program to date. While the TN 7/2 casks owned by Nuclear Cargo Services had been used during numerous occasions at ANSTO, the NAC-LWT cask had not previously been used at the ANSTO facility.

<u>Licensing and Basket manufacturing effort</u>

Three types of fuel elements were part of the shipment (Type 1, Type 2 and Type 3).

The Type 1 and Type 2 fuel types are cylindrical elements while the Type 3 fuel consists of flat plates.

The TN 7/2 casks were used to ship the Type 1 fuel elements for which they were already licensed. The NAC-LWT casks were used to transport the three different fuel types. The NAC-LWT cask was already licensed for the Mark IV fuel, as NAC had previously shipped such fuel from the Danish research reactor of Risø. However, the NAC-LWT casks had not been licensed as an authorized package for the Type 2 fuel elements. Additionally, the Type 2 fuel elements could not fit into the existing DIDO baskets as the fuel is 4.5 mm (0.18”) larger in diameter than the Type 1 fuel. Consequently, NAC obtained a license for and manufactured a new basket type to accommodate the Type 2 fuel elements. In order to maintain a maximum capacity of 42

fuel elements per cask, NAC designed a new basket with seven cells (six baskets/cask). This new basket design, with the same capacity as the original DIDO basket, was possible in part because of the low heat load of the fuel which had been cooling for more than 20 years.

Regulatory approval was also necessary for the Type 3 fuel plates, which were re-assembled by ANSTO from their original Reactor Fuel Element configuration of 12 plates with 15mm spacers into 14 Shipment Elements with spacers of 1.5mm to reduce the K effective to below 0.5 prior to packaging.

A request to amend the NAC-LWT certificate of compliance was submitted by NAC to the U.S. Nuclear Regulatory Commission (NRC) in April 2006. Upon completion of its review, the NRC issued a revised NAC-LWT certificate of compliance in August 2006. In order to apply to the Australian Maritime Safety Authority (AMSA) and the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA) for an Australian validation of the package, NAC received a U.S. Department of Transportation (DOT) certificate in August 2006. The Australian validations were issued in October 2006.

Project Preparation

A project kick-off meeting was held between ANSTO and NAC at the Lucas Height site in July 2006 to initiate the project. During this meeting, the ANSTO and NAC project teams assessed the site layout and infrastructure, selected a cask loading method, reviewed the cask documentation and agreed on a project schedule.

All maintenance of the NAC-LWT casks, packing, and preparations for shipment of all support equipment and parts were performed at NAC's storage and maintenance facility located at Wampum, Pennsylvania.

The two German-based TN 7/2 casks were already at Savannah River Site (SRS), which simplified their inclusion in the outbound shipment. Recently, operators of liner ships have become more reluctant to accept radioactive cargo. Consequently, it was decided to ship all seven casks and three additional 20' ISO containers of equipment together on one ship in order to facilitate the booking acceptance for the cargo using a liner-ship.

Site Preparation

Organizations utilizing the German TN 7/2 or the French TN-MTR casks will be well aware of the preparations required before and after loading the casks with spent fuel whilst submerged in a pond. The NAC-LWT cask presents a completely different scenario; however it is one that is easy to operate despite the larger volume of support equipment. Loading the baskets in the pond and retrieving them into the Dry Transfer System (DTS) is no different from the principles employed of withdrawing fuel from a pond using a specified cask and depositing it in dry storage.

Given that not many operators of research reactors have pools sufficiently deep to receive the NAC-LWT cask nor, indeed, buildings of sufficient headroom and crane capacity (23 tones) to conduct this exercise inside, this is best done outside by deploying a 100 tones mobile crane.

Preparatory steps:

- Availability of a suitable working area for all the containers which has sufficient space to employ a side loading device to lift the container from the ground on to a transporting trailer.
- Availability of a reinforced concreted area to receive each cask at the loading area which is flat, level and of sufficient area to receive the container.
- The base plate area for the NAC-LWT to be mounted is flat and level concrete-base.

At ANSTO all seven casks were received, inspected, surveyed and unloaded in the presence of NCS and NAC technicians.

Cask Receipt/Inspection/Unloading Area

Cask Survey prior to setting up casks

NAC-LWT Cask Loading Area

TN 7/2 Cask Loading Preparation Area

The full complement of ten ISO containers, including the seven casks and three ancillary equipment containers, were received in a dedicated area where they underwent comprehensive smear swipe testing to check for potential contamination.

Cask Loading Operation

As stated above, the TN 7/2 casks had previously been used at ANSTO. Consequently, the same wet loading method was repeated. It took a total of ten working days to load and test the two TN 7/2 casks.

The reactor pool at ANSTO is too shallow to permit a wet loading of the tall NAC-LWT cask. It was therefore decided at an early stage to perform the cask loading operations using the NAC Dry Transfer System (DTS) with the pool adapter (shielded tube placed across the top of the pool). This concept was approved by the Australian Regulatory agency (ARPANSA). The DTS is a transfer cask used to move one loaded basket from the pool into the cask. Each cask accommodates six baskets.

The photos below provide a pictorial overview of the cask loading operation. This method was new to ANSTO. However, it is a well-proven process which has been used in more than 20 facilities worldwide.

NAC provided technical guidance to ANSTO during the cask loading operation which took place from the beginning of October until mid-November, 2006.
The time taken to load and test the five NAC-LWT casks was less than four weeks.

Shipment

The convoy of seven loaded casks with attendant three containers of equipment was transported under high security to a nearby port. After a long maritime voyage, the casks were delivered safely to the DOE site at Savannah River in South Carolina.

Conclusion

This shipment is the largest spent nuclear fuel shipment ever performed from a single facility under the FRR acceptance program (330 fuel elements in total representing approximately 30kg of U235). It was a very comprehensive project including design, licensing, basket manufacturing, usage of seven casks, technical assistance for cask loading operations and transportation activities. Adding to the complexity of the project was the numerous entities involved in all the different tasks and jurisdictions. The successful shipment provides a useful precedent for the shipment of the remaining HIFAR spent fuel will be shipped to SRS in the coming years.

THE EXPERIENCE OF SHIPPING SPENT NUCLEAR FUEL FROM UZBEKISTAN TO THE RUSSIAN FEDERATION

Dr. B. Yuldashev and Dr. U. Salikhbaev
Institute of Nuclear Physics
Ulugbek, Tashkent 702132 - Uzbekistan

And

Dr. I. Bolshinsky, INL and J. Thomas, SRS
Research Reactor Fuel Return Program
1000 Independence Avenue, SW, Washington, DC 20585 – USA

And

V. Tyazhkorob
Production Association Mayak
Lenina Prospekt 31 456790 Ozersk, Cheliabinskaya Oblast – Russian Federation

And

Y. Golyapo
KATEP-AE
Lisa Chaikina 4, Cottage 5, 480020 Almaty – Kazakhstan

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 will describe 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.

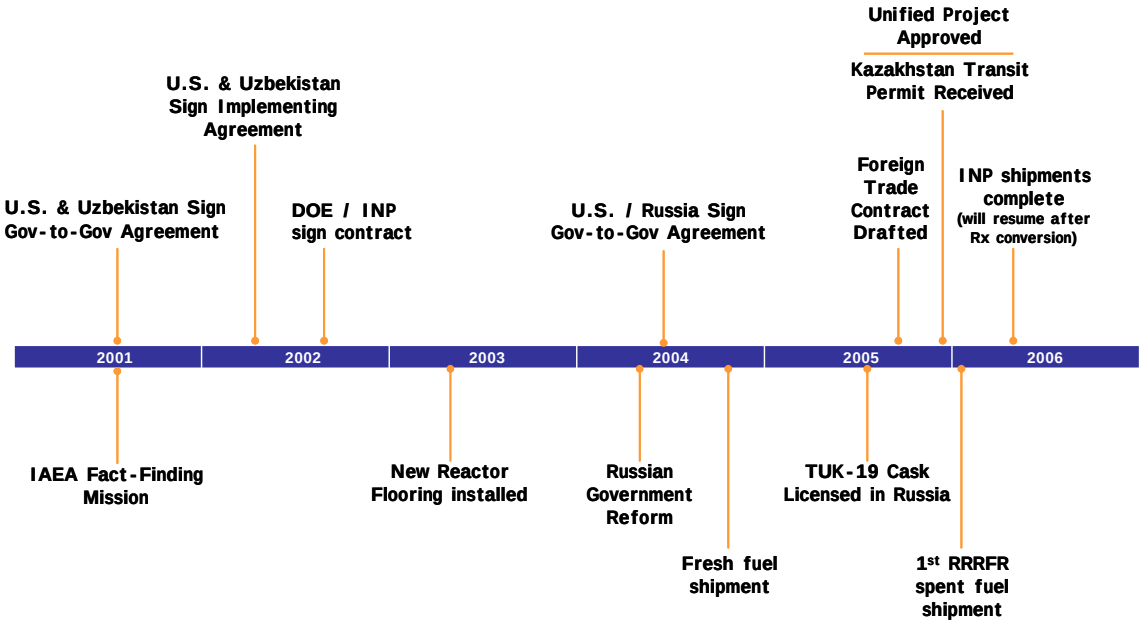
Introduction

The research water-water reactor of the Institute of Nuclear Physics of Uzbekistan Academy of Sciences (Tashkent, Uzbekistan) is operating since 1959. From the beginning till 1977 the reactor was working at thermal power 2 MWt using high enriched fuel. In 1977-1978 after upgrading the reactor started to operate at power 10 MWt which helped to increase significantly

the flux of neutrons and, as result, to start mass production of isotopes for needs of medicine, science and technique. In former Soviet Union the Institute of Nuclear Physics (INP) was one of the leading facilities producing reactor isotopes at the commercial level. Nowadays the INP is producing isotopes like I-125, I-131, P-32, P-33, S-35, Au-198, Ir-192, Sr-89, Re/W- and Tc-99M generators and other ready to use preparations for internal needs as well as for export. In addition, the reactor is used for neutron activation analysis, material sciences, ennobling the precious and semiprecious stones and for studies in nuclear physics.

For the last fifteen years reactor was operating, in average, more than 5000 hours a year using from 1978 till 1997 the 90%-enrichment fuel and then, from the middle of 1997 till the present time, the 36%-enrichment Russian made IRT-3M type fuel. Such an intensive use of reactor, in turn, created a problem of handling, in particular, of spent fuel the storage facility for which is limited. The last shipment of spent fuel back to Russia was done in 1991 and due to known reasons (decrees of Russian government in 90's on not accepting highly radioactive waste, in particular) the INP was not able to send spent fuel and had to keep it building two new additional storage facilities. In result, by the beginning of 2006 more than 300 spent fuel assemblies were stored at facility and, because of limited space, the problem of further operating of reactor became serious.

However, 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 changed dramatically since the last shipments 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 below highlights the major milestones of the project.

This paper will not attempt to discuss all of the activities completed over the past two years but will 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 who were involved will also be identified along with a brief description of their

[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.

responsibilities. The details of the shipments (i.e. cask loading, logistics) will be described followed by the lessons learned and future plans of the reactor.

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.

Table 1: Project Team

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.

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. 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 enter into a 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 (to be discussed later), which was another critical path activity. With the government-to-government agreement in place, INP was permitted to contract with Mayak for the return of the spent fuel. This contract called the Foreign Trade Contract and discussed later in this paper, specifically addressed the following issues:

1. The scope of services to be provided by the Russian Federation. Services included temporary storage of SNF, SNF processing, interim storage, and radioactive waste return.
2. The contract defined the owner of the SNF after their importation and the owner of the reprocessing products after SNF reprocessing.
3. The contract included confirmations 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.

Unified Project

A Unified Project was required for the importation of spent nuclear fuel into the Russian Federation per Russian Law [1]. The Unified Project 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 provided 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 titled '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, a duration of 15 months is recommended to develop and obtain SEER approval of the Unified Project.

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 (right) had been allowed to expire in 2000 due to its inactivity. The transportation license, which is shipment specific and issued for each shipment campaign, required 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.

The Kazakhstan license for the design and transportation 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 that VNIPIET 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 of the TUK-19 licensing activities and 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.

Kazakhstan Transit Requirements

As with the TUK-19 cask license, KATEP coordinated all of the transit activities for this shipment. This included the following main activities:

- Development of '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
- Signing of the 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.

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:

- New reactor hall flooring was installed to increase safety and help prevent the spread of contamination.
- New reactor hall lighting and remote 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 new remote cameras improved the loading operations (right), making loading quicker and safer.

- 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.

The reactor staff and support organizations received extensive training on the operations and procedures of every aspect of the fuel shipments. Many practice exercises were performed on: fuel loading; cask handling and loading; cask preparation; criticality and personnel safety; radiological safety; and security.

Shipment Details

The shipment consisted of the transport of 252 IRT-3M spent fuel assemblies enriched to 36% and 90% ^{235}U. The IRT-3M assembly is shown in the left picture below. Several months prior to the shipment, the fuel assemblies were inspected by Mayak experts. All of the spent fuel 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, four shipments were needed to return the 252 spent fuel assemblies to Mayak.

The shipment process was identical for each shipment. 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 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 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 while 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.

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.

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 early October. 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 guideline document and intended for use as a reference tool. Sample passport and receipt documents, agreements, and contracts will be shared at this workshop. This experience demonstrated the need to identify all of the 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.

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.

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POSSIBILITY OF A PARTIAL HEU-LEU TRIGA FUEL SHIPMENT

M. VILLA, T. STUMMER R. KHAN, H. BÖCK
Vienna University of Technology, Atominstitut
Stadionallee 2, 1020 Vienna – Austria

ABSTRACT

The TRIGA reactor Vienna operates since March 1962 and was initially started up with 66 TRIGA fuel elements type 102 (Al-cladding, LEU). In the following years additional spare fuel elements of type 104 (SST cladding, LEU) were purchased which were added to the core. Therefore the core was composed of two types of fuel elements. Later on in 1974 fuel elements type 110 (SST cladding 70% enriched) were acquired and placed into the core resulting in a complete mixed core with 3 different fuel element types. As the reactor power of 250 kW is rather low only a very few fuel elements had to removed from the core in the past 45 years, mainly because of mechanical damage during fuel handling. In fact in all those years only 8 fuel elements are removed permanently from the core and stored in a dry spent fuel storage pit. This paper describes MCNP modelling of the TRIGA Vienna core if FLIP elements are replaced by standard fuel elements

1. Introduction

As a result of the US spent fuel return program the HEU fuel elements are the main target of DOE. For the TRIGA reactor Vienna it is however necessary to continue operation as long as possible. Recently possible solutions were worked out and will be discussed in more detail in this paper:

1. Austria could possibly obtain 15 fresh TRIGA elements from the TRIGA Slovenia and could thus remove the HEU from the operating core.
2. Further a fuel shipment is planned from Romania in 2008 which could be used for the Austrian HEU and the 8 LEU spent fuel elements if these fuel elements are shipped with proper timing from Vienna to Pitest. It all depends on careful planning and excellent cooperation with the involved parties.

2. The fuel history

The 250 kW TRIGA Mark-II reactor operates since March 1962 at the Atominstitut, Vienna, Austria. Its main tasks are nuclear education and training in the fields of neutron- and solid state physics, nuclear technology, reactor safety, radiochemistry, radiation protection, dosimetry, low temperature physics and fusion research. Criticality of a typical TRIGA Mark II reactor is usually achieved with about 57 standard TRIGA fuel elements (about 2 kg ^{235}U). To allow higher power operation (100 kW) more fuel elements and several graphite reflector elements are usually added in the outer ring of the core. This results in an operating core of about 66 fuel elements and a core excess reactivity of about 2 $ depending on specific license of such a reactor. Since the first criticality further fuel elements were purchased and added to the core. During the past decades the fuel supplier (General Atomics) has changed the fuel specifications several times. Since the development of the original TRIGA type reactors in the mid-fifties, several new types of TRIGA fuel elements have been developed. A significant change in the design was replacing the aluminium cladding material to stainless steel (SST).

Therefore, many TRIGA reactors are operated with a mixed core using different fuel elements in the core. Among the SST elements several different types of fuel exist. Most widely used fuel elements are TRIGA elements with <20% enrichment (Standard SST) and in some cases fuel elements with approximately 70% enrichment (called FLIP = **F**uel **L**ifetime **I**mprovement **P**rogram). Details about these different fuel types are given in Table 1. Recently, General Atomics has designed and certified a new low enriched fuel with high uranium loading (as high as 20 wt-%) to replace the FLIP fuel. Gradually, FLIP research reactors are planned to be converted to low enriched uranium (LEU) fuel due to worldwide nuclear proliferation concerns. This new LEU fuel has higher uranium loading to give a long fuel lifetime without high enrichment. Additionally, General Atomics has produced a TRIGA fuel with a smaller diameter for higher (>3MW) power operations.

Fuel element type	102	104	110 (FLIP)
Fuel moderator material	U-$ZrH_{1.0}$	U-$ZrH_{1.65}$	U-$ZrH_{1,65}$
Uranium content (wt%)	8.5	8.5	8.5
Enrichment (%)	20	20	70
Average ^{235}U content (g)	38	38	136
Burnable poison	SmO_3-disk	Mo-disk	Erbium 1.6 wt%
Diameter of fuel meat	35.8 mm	36.3 mm	36.3 mm
Length of fuel meat	35.6 mm	38.1 mm	38.1 mm
Graphite reflector length	10.2 mm	8.73 mm	8.81 mm
Cladding material	Al-1100F	304 SS	304 SS
Cladding thickness	0.76 mm	0.51 mm	0.51 mm

Tab 1: Specifications of fuel elements used in the core of the TRIGA reactor Vienna

At present nine Flip elements are installed in the Vienna TRIGA core. Eight of them are since 1972 in the C ring of the core, one is near the neutron source of the reactor in the F- ring

Since the elements are in the core more than 34 years, table 2 gives more detailed information about the nine high enriched elements.

Fuel Number	Fuel Typ	Burn up (MWd)	Uranium weight (g)	U-235 weight (g)
7301	Typ 110	5,3	194	135
7302	Typ 110	4,3	195	136
7303	Typ 110	5,3	194	135
7304	Typ 110	4,3	194	135
7305	Typ 110	4,3	194	135
7306	Typ 110	3,5	194	135
7307	Typ 110	3,98	194	135
7308	Typ 110	3.4	194	135
7309	Typ 110	3.3	194	135

Tab 2: History of the Flip elements at the TRIGA reactor Vienna

3. Present Situation

During the past RRFM meetings the possibilities to remove and return these nine FLIP elements were discussed with NNSA and DOE representatives. Presently the situation is as follows:

The TRIGA reactor Vienna is used intensively not only by the Vienna University of Technology and other Austrian universities for education and training purposes but also by the IAEA because it is now the only nuclear facility close to the IAEA headquarters. The previous 10 MW ASTRA reactor had been ultimately shut down in summer 1999, thus remaining the only nuclear facility in Austria. Removing these nine FLIP elements from the TRIGA core would seriously endanger the future operation of this reactor. Although some standard TRIGA fuel elements are on stock which will not be of sufficient worth to extend the reactor operation life till May 2016.

Contacts have been established with the Josef Stefan Institute/Slovenia as information was obtained that some fresh TRIGA fuel elements are available, however the latest information received was that they will be offered to CERCA in a compensation deal. Otherwise the plan existed to obtain about 9-10 fuel elements from Slovenia and to remove the FLIP elements from the TRIGA core for interim storage until a reasonable shipment to the US is available such as it is planned from Pitest/Romania in 2008. However it seems that the CERCA deal is not yet settled a in February 2007 information was received form IJS that all fuel is still in Ljubljana.

4. On-going Activities

Since fall 2006 Monte Carlo calculations started at the Atominstitute incorporated in one Diploma work and one PhD work to calculate the flux density changes when replacing the 9 FLIP element but standard 104 TRIGA fuel elements. As the Vienna TRIGA core is mixture of old Al-clad fuel elements (Type 102), SST clad elements from various periods and FLIP elements these calculations are rather complicated especially as it turned out that even among one series of fuel elements considerable variations in composition (i.e. H-content of FLIP elements) are present. The Monte Carlo method is used to simulate a statistical process such as interaction of nuclear particles with material and is particularly useful for complex problems which need more approximations (homogenisation and multi group cross-section treatment, etc.) when modelled by deterministic computer codes.

To model the TRIGA MARK II reactor, Vienna, the general purpose 3-D Monte Carlo N-Particle code MCNP was chosen because of its generalized geometry, continuous energy cross sections modelling capability and long history of use in reactor calculations. All fresh fuel, control rod, and other elements (e.g., neutron source, graphite dummy elements) models were prepared in order to be able to simulate any possible fresh core.

4.1 Methodology

A full core MCNP model up to the tank wall was created for the TRIGA Mark II Vienna using detailed core geometry and isotopic composition of each element in the core materials (Ref). All three types of fuel elements (102, 104, 110) control elements, the surrounding reflector and beam tubes were modelled in detail. In this frame work, two main cases were considered for simulation. In first case, core incorporates only two types of fuel i.e. 102 and

104. Three control rods i.e. regulating, safety and transient rod were kept fixed at position 500 (full out) to obtain the total excess reactivity of the core.

In second case, the nine 104-type fuel elements at positions C1, C2, C4, C5, C7, C8, C9, C11 and F27 were replaced by FLIP elements (as used in the actual core). All other conditions i.e. control rod and other elements position were kept unchanged. The top (XY) and vertical (YZ) view of the core simulation is in Fig.1.

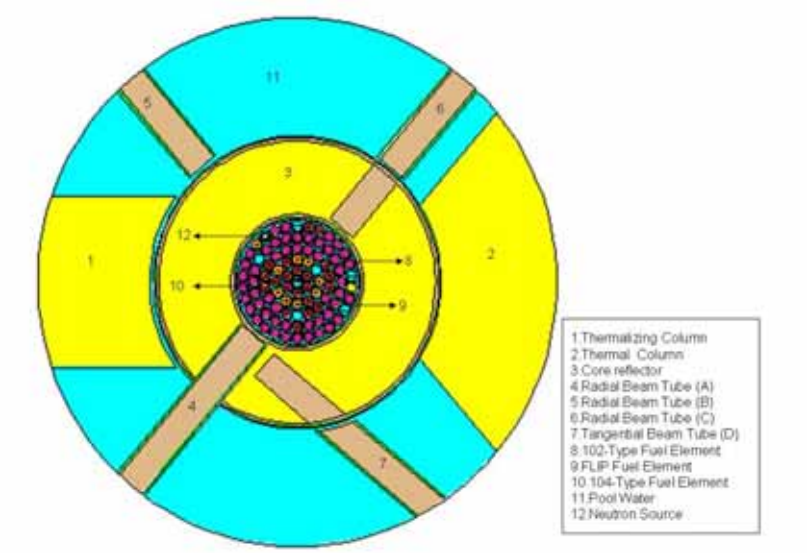

Fig 1. Horizontal cross section of the TRIGA Mark II core in Vienna

Using flux tallies, the radial and axial distribution of the fluxes were plotted for both case described above and are shown in Figs. 2 and 3 at the end of the paper.

In this Monte Carlo simulation, with the help of KCODE card, the effective neutron multiplication factors for both cases have been calculated. The simulation results give us more reliable information about the rod worth compared to the experimental results.

4.2 Reactivity Worth Difference between 102, 104 and 110 fuel elements.

Generally speaking there are three important characteristics for a fuel element in MCNP if the geometry is constant and well known: U235 mass, H content and poison mass. Concerning the poison the influence is small due to the low power and all results will be related to a non-Xenon-poisoned core, the other two factors are significantly different between fuel elements and the overall average can differ from the values given in the supplier's data sheets. While the U235 mass varies for about +/-0.5g (at an average of 38g per fuel element) the core average value can easily be obtained using the data from the shipping documents, however this is not the case with the H content. The H/Zr ratio is given for some of the elements and i.e. varies from 1.57 to 1.63 for the 9 FLIP elements. Even relative minor shifts in the ratio have a significant effect on keff simulations: In example changing the H-content from 1.6 to 1.65 (+0.0484 wt%) for the 104 elements (22 out of 82) results in a keff change of +1.12%.

While operating the reactor at a power level of 10W and replacing a 104 rod with a 110 rod at core position C1 produces a difference in control rod position and gives therefore an estimate of the reactivity worth difference between the two rods. In this case the difference is about 0.267 $ per rod or a total of 2.4 $ for the whole core (rel. error about 5-10% due to the

accuracy of the rod calibration curve). The simulated values are 0.262 $ and 2.36 $ respectively. If replacing a 104 rod with a 102 fuel element in position F2 with similar burn-up the reactivity change is extremely small with -0,02$ and in the order of magnitude of the measurement precision.

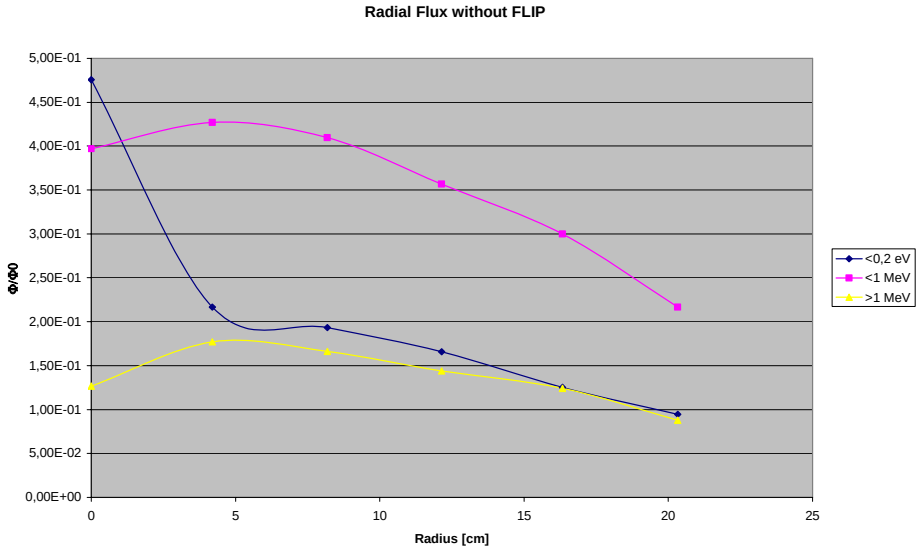

Fig. 2: Relative energy dependent radial flux without FLIP integrated over the fuel meat height from the core centre to the outer ring (Normalisation: Total flux at r=0 :=1).

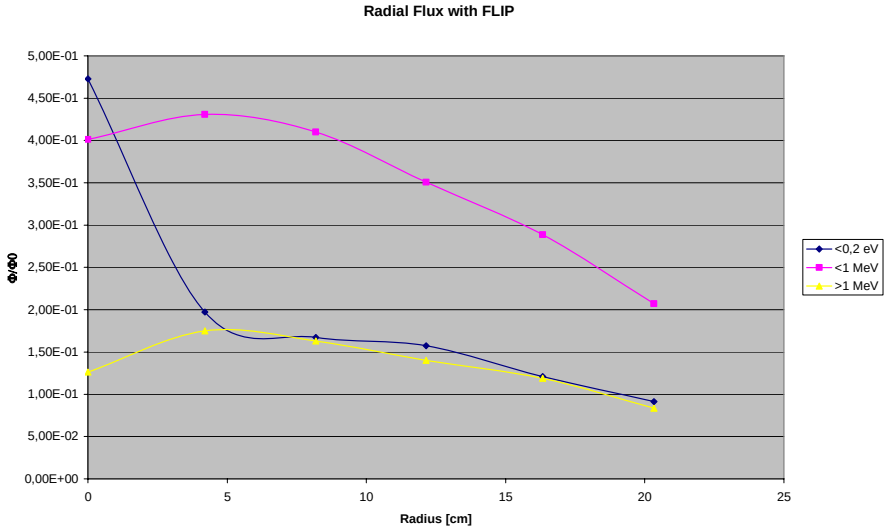

Fig. 3: Relative energy dependent radial flux with FLIP integrated over the fuel meat height from the core centre to the outer ring (Normalisation: Total flux at r=0 :=1). Note the relatively lower thermal flux around the FLIP elements (r=5-10cm) compared to the 104 elements at the same position in the previous diagram.

5. Summary and conclusions

Calculations show that there is no major difference in core performance between a TRIGA core with 9 FLIP elements and TRIGA core using 20% enriched standard fuel elements except in long time burn-up behaviour. They main problem with the TRIGA reactor Vienna reactor is that there are only 8 spent fuel elements stored and a partial fuel shipment of these stored elements together with 9 FLIP elements is extremely expensive, further when initiating such an effort a high risk of a governmental decision to shut down the reactor completely is imminent.

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European Nuclear Society
Rue de la Loi 57
1040 Brussels, Belgium
Telephone +32 2 505 30 54
Fax + 32 2 502 39 02
rrfm2007@euronuclear.org
www.euronuclear.org

Layout and Design: Marion Brünglinghaus, ENS