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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 info@euronuclear.org
Internet www.euronuclear.org

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Session IV

Fuel back-end management

DEMONSTRATION OF THE EMPLACEMENT TECHNOLOGY FOR THE DIRECT DISPOSAL OF SPENT FUEL INTO DEEP VERTICAL BOREHOLES

W. FILBERT; W. BOLLINGERFEHR; J. WEHRMANN

DBE TECHNOLOGY GmbH, Eschenstraße 55, 31224 Peine

ABSTRACT

This paper outlines disposal techniques for spent fuel canisters in deep vertical boreholes starting from the 870m-level of a repository in salt. This is a new approach to minimize and optimize the efforts for transport, handling, and disposal of spent fuel rods compared to the reference concept of drift disposal of heavy self-shielding Pollux casks. The BSK 3 canister and necessary equipment for disposal operations, i.e. internal transfer cask, transport cart, borehole lock, and emplacement devices as well as the operational sequences are described. A test program for inactive tests in a surface facility is presented. The tests are necessary to demonstrate the reliability and safety of the emplacement system by means of a large number of demonstration tests and to draw conclusions and give recommendations for industrial application in the repository. The demonstration tests are followed by tests to eliminate operational disturbances and simulation tests. The work is performed in the context of the 6th European Framework Program.

1 Introduction

In Germany, the reference concept for the disposal of heat-generating radioactive waste considers the emplacement of canisters with vitrified waste and CSD-C (Colis Standard des Déchets Compactés) canisters in deep vertical boreholes drilled from the drifts of a repository mine in salt at a depth of 870 m. Spent fuel is to be disposed of in self-shielding POLLUX casks (weight up to 65 t) in horizontal drifts [1]. An optimized disposal concept anticipates the emplacement of unshielded canisters containing the fuel rods of 3 PWR or 9 BWR fuel assemblies [2] (Fig. 1) in boreholes with a diameter of 60 cm and a depth of up to 300 m.

This concept (called BSK 3-concept) provides the following optimization possibilities:

- A new steel canister of the same diameter (43 cm) as the standardized HLW canisters used for high-level waste and compacted technological waste from reprocessing abroad can be filled with the fuel rods of 3 PWR or 9 BWR fuel assemblies.
- The standardized canister diameter provides the possibility to use the same transfer and handling technique for both categories.
- The BSK 3 canister is tightly closed by welding and designed to withstand the petrostatic pressure at the emplacement level.
- The residual heat generation of a canister loaded with fuel rods burned up to 50 GWd/tHM will allow its emplacement in a salt repository already after about 3 to 7 years following reactor unloading of the fuel assemblies. This has been evidenced by thermal calculations.

Thus, the BSK 3-concept may considerably reduce the necessary effort in terms of time and costs. For this reason, a research program was launched to develop and test the necessary technical components and to transfer this emplacement technology into state-of-the-art technology.

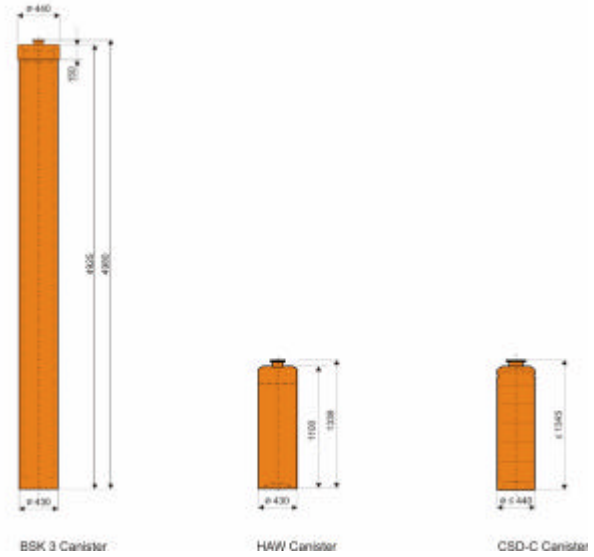

Fig. 1: BSK 3 canister, HLW canister, and CSD-C canister

The R&D work is part of the 6th Framework program of the European Union. The Integrated Project ESDRED deals with the development and demonstration of repository-relevant transport and emplacement techniques and is performed by a consortium of 13 partners from 9 European countries. Financial support is provided by waste management agencies and by the European Commission. The activities performed by DBE TECHNOLOGY GmbH are co-financed by the German Project Management Agency Karlsruhe, and the manufacturing of the components by the German nuclear industry represented by GNS.

2 Emplacement System for BSK 3 canisters

Fig. 2 shows an outline of the emplacement of a BSK 3 canister and the components involved.

The main components are:

- A BSK 3 canister (weight 5.2 t),
- A transfer cask for transport of BSK 3 canisters (weight loaded 50 t) within the testing facility,
- An emplacement device (weight 66 t),
- A borehole lock (weight 6 t) and
- A transport unit consisting of a transport cart and a battery driven mine-locomotive for rail-bound transport in the repository.

The internal transfer cask is loaded at the conditioning plant or at a hot cell at the repository. After the transfer cask has been closed and disconnected, it is placed onto a transport cart to be transported underground. The loaded transport cart is placed in the hoisting cage and subsequently transported through the shaft down to the repository level where it is parked in the underground bay station. Then, a battery driven mining locomotive brings the transport cart to one of the active disposal drifts (Fig. 2).

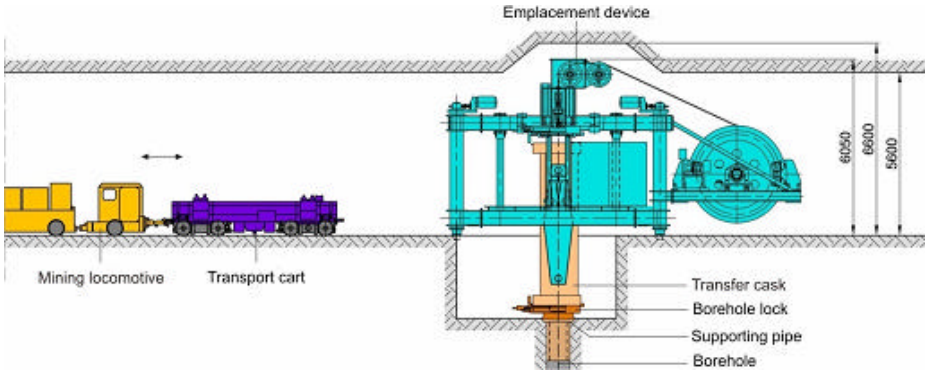

Fig. 2: System for the emplacement of spent fuel canisters into deep vertical boreholes

Once it reaches the disposal drift, the transport cart is attached to the emplacement device which is positioned over a disposal borehole. First, the lifting gantry of the emplacement device picks up the transfer cask, then the transport cart is removed. The cask is still in horizontal position over the borehole. Still within the emplacement device, the transfer cask is tilted to a vertical position and subsequently lowered and docked onto the borehole lock.

After the transfer cask has been docked, the shielding cover, which is part of the hoisting gear in the load portal above the raised transfer cask, is lowered and locked onto the upper end of the transfer cask. This shielding cover contains pulleys and ducts to guide the hoisting cables. After the lock slider at the upper end of the transfer cask has been opened, the canister grab, which is retracted inside the shielding cover, is lowered into the transfer cask where it locks onto the BSK 3 canister. Subsequently, the canister is elevated inside its transfer cask so that its lower lock slider, which is mechanically connected to the borehole lock slider, can be opened. After the lock sliders have been opened, the hoisting gear lowers the BSK 3 canister into the borehole to a position just above the backfill of crushed rock salt above the last emplaced canister, followed by an inching operation for the last few centimetres. Limit switches signal the release of the grab jaws, after which the grab is retracted into the shielding cover of the emplacement device. Undocking and return to the surface of the empty transfer cask is done in reverse order.

To provide sufficient radiation protection, the transfer cask consists of a thick-walled (445 mm) cylindrical body made of spheroidal graphite (SG) cast iron and two locks made of stainless steel. To provide sufficient neutron moderation, two concentric rows of polyethylene (PE) rods, equivalent to a 70-mm PE layer, are included in the cylindrical body. Both lids are equipped with 20-mm-thick PE disks. To facilitate its handling, the transfer cask is fitted with 4 cylindrical trunnions (Fig. 3).

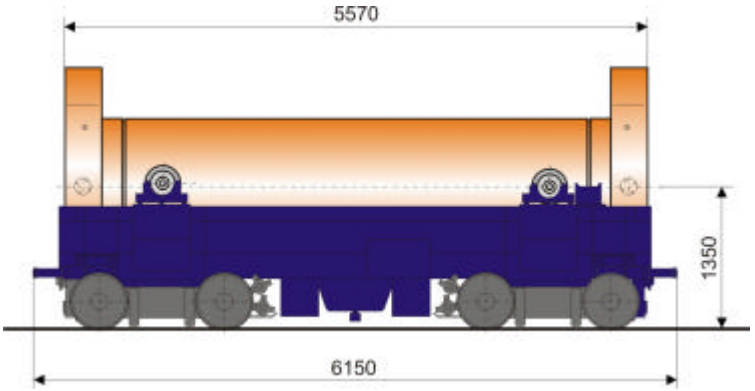

Fig. 3: Transport cart with the transfer cask

The borehole lock (Fig. 4) consists of a body and a flat slide latch, both made of stainless steel, as well as the equipment to guide and drive its slider. The upper part of the body is collar-shaped, supporting the transfer cask once it is in docked position. The upper part of this supporting collar is funnel-shaped to guide the transfer cask during its insertion. The lock slider is a solid cuboid steel body providing protection against radiation from the borehole. Once a transfer cask has been docked onto the borehole lock, its lock slider is mechanically locked to the transfer cask slider so that both sliders are operated simultaneously by the drive of the borehole lock.

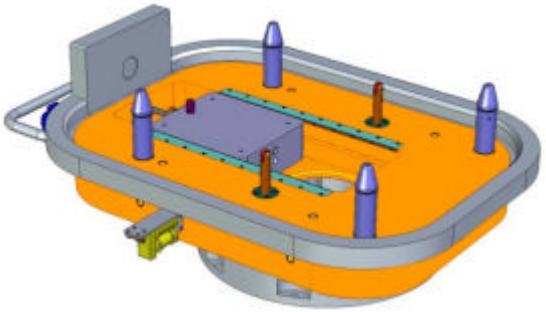

Fig. 4: Borehole lock

The emplacement device consists basically of the assembly units: lifting gantry, flap-frame with controls, swivel gear, canister lifting gear including hoist cable and lifting tackle, and shielding cover.

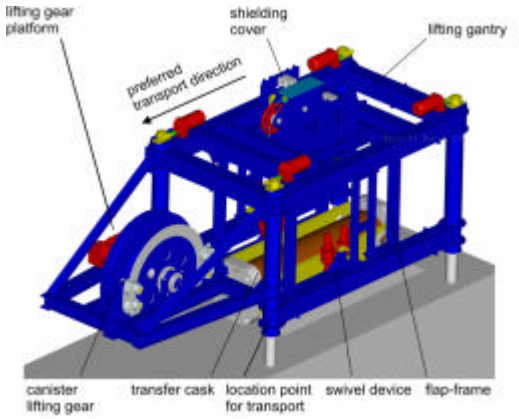

Fig. 5: Emplacement device

3 Test Program

As the safe function of the underground emplacement system cannot be shown by referencing to a comparable existing system, a full-scale test program, which focuses on three main points of examination, has been drawn up.

In a series of **demonstration tests**, the handling and sequences planned for the underground emplacement process are to be demonstrated, taking into account all the components which are relevant to the system's function and control. In combination with a special test programme, experimental data on the reliability of the underground emplacement process are to be obtained during 500 emplacement cycles.

Possible disturbances of operation are to be analysed with regard to their effects in order to plan corrective actions systematically. During the **tests on possible operational disturbances**, the corresponding corrective actions are to be recorded and improved in order to be able to describe detailed measures for an operating manual for the repository. In addition to this, data to estimate the radiation exposure of the operating personnel taking corrective actions in a repository are to be determined.

During the **simulation tests**, the components of the emplacement system, the emplacement device, the borehole lock, and the transfer cask are to be tested to ensure that they meet the requirements, and possible disturbances of operation are to be simulated.

For test purposes, an unused turbine hall of the E.ON power plant at Landesbergen (Lower Saxony) has been rented. Fig. 6 shows the situation at the test stand. A 10-m-long vertical steel metal casing will simulate the emplacement borehole. The BSK 3 canister will be lowered down by the grab of the emplacement device and - unlike in a real repository - removed again for further tests afterwards.

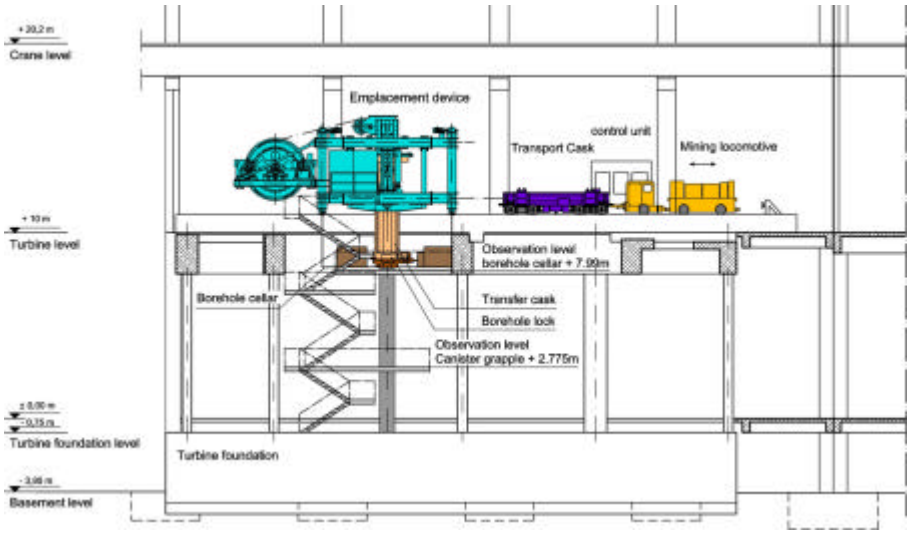

Fig. 6: Test stand

4 Status of Project and Outlook

All the components have been designed and evaluated by external experts. The components will be manufactured until June 2008. The construction work to prepare an appropriate test facility will be performed from February to April 2008. After the individual components have been delivered and accepted on site, the demonstration program - performed in two shifts - will commence in June and last until November 2008. At present, these tests are being prepared. The test results will be presented in December 2008.

5 References

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CORROSION OF SPENT ALUMINIUM-CLAD RESEARCH REACTOR FUEL – SYNERGISM IN THE ROLE OF STORAGE BASIN WATER PARAMETERS

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

ABSTRACT

Concerns about corrosion of aluminium clad research reactor (RR) fuels in wet storage lead to three IAEA supported projects. The activities within these projects consisted of exposing racks of test coupons of a variety of aluminium alloys in different configurations to spent fuel storage basins in 17 participating countries worldwide. The aluminium alloys were representative of typical RR cladding alloys, handling tools and storage racks. The coupons were evaluated after predetermined exposures times and the storage basin water parameters were monitored periodically during the projects. Pitting was the main form of corrosion and this was influenced by the conductivity and chloride ion content of the water, formation of galvanic couples and settled solid particles. Marked synergism was observed in the influence of these parameters on aluminium alloy coupon corrosion and this paper highlights this phenomenon.

1. Introduction

The International Atomic Energy Agency's (IAEA's) RR spent fuel data base (RRSFDB) shows that there are 62,027 spent fuel assemblies stored in various facilities around the world and another 24,338 assemblies in RR cores. Over 90% of these spent RR fuels are clad with aluminium or aluminium alloys and stored in wet basins. (Table I) The at-reactor storage is often in a different section of the reactor pool or in a separate pool within the reactor building, often referred to as the decay pool. In many countries the fuel has been in storage for periods of up to 50 years in different types of light water pools.

Type of storage	At-reactor	Away-from-reactor
Pool	154	55
Dry well	25	31
Vault	10	12
Other	18	6

Tab I: Spent fuel storage facilities

In the late 1990's, corrosion of aluminum-clad spent nuclear fuel stored in light-water filled basins became a concern and the IAEA implemented the coordinated research project (CRP) on "Corrosion of Research Reactor Aluminum-Clad Spent Fuel in Water.[1] A regional technical co-operation project for Latin America (RLA) titled 'Management of spent research reactor fuel', was also supported by the IAEA and carried out from 2001 to 2005. The CRP and the corrosion activities within the RLA consisted of exposing Al alloy coupons to spent fuel storage basins for predetermined periods followed by their examination. Seventeen countries participated in the two projects and the materials selected for testing were representative of RR cladding alloys handling tools, and storage racks. [1,2] These alloys included mainly, AA 1100, AA 6061 and SZAV-1. In addition to these alloys, many of the participants included site specific alloys in their studies. During the execution stage of the projects, the storage basin water parameters were monitored periodically. The influence of Al alloy composition, galvanic effects (Al alloy/stainless steel), crevices, water parameters, coupon orientation, alloy grain orientation and settled solids on the corrosion of the coupons were evaluated. Complementary laboratory tests were also carried out at some of the participating country's laboratories to determine the effect of specific parameters on the nature, extent and mechanism of Al alloy corrosion.

This paper presents the main observations of these projects with respect to the effect of spent fuel storage basin (SFSB) parameters such as conductivity, aggressive ions, galvanic coupling and settled solids on the corrosion of the coupons and the synergism in the effects of these parameters on Al alloy corrosion.

2. Materials and Methods

Aluminium alloy coupons of AA 1100 (or AA 1050), AA 6061 and SZAV-1 were assembled in stainless steel test racks with alumina separators as shown in figure 1. [1] The separators were used to avoid metallic contact between coupons and between the coupons and the rack. Site specific alloy coupons were also added to the racks. In the participating countries the coupons were exposed to spent fuel basins at two or more sites. The racks also contained Al alloy-Al alloy and Al alloy-stainless steel coupon couples to simulate crevices and bimetallic (galvanic) contacts. At many sites, coupons were exposed both vertically and horizontally to evaluate the effects of settled solids. [3-6] The coupons were exposed to the water in SFSBs for periods of up to 6 years. During this period the water parameters pH, conductivity, chloride content, temperature and other ions that were site specific were monitored periodically. Graphs of variations in the water parameters were plotted to help correlate coupon corrosion with water parameters. After pre-determined periods the racks were withdrawn from the basins and the coupons examined and the extent of corrosion evaluated following standard procedures outlined in CRP Test Protocol. [1]

Fig 1. A typical rack with test coupons.

3. Results and discussion

A large amount of data was generated about the corrosion of the aluminum coupons at the different sites. [1,2] Pitting was the main form of corrosion. The number, size and distribution of the pits on the coupons varied from one site to another. This data indicated that the factors that contributed to corrosion of aluminum alloys in the SFSBs were: (a) high water conductivity (100-200µS/cm); (b) aggressive ion concentration (Cl^-); (c) galvanic coupling between dissimilar metals (stainless steel/aluminum); (d) settled solids that are cathodic with respect to aluminum (Fe); (e) sludge (which contains significant amounts of Fe, Cl, and other ions); (f) scratches and imperfections in the surface oxide; (g) poor water circulation. Direct correlations between each of these parameters and pitting corrosion of Al alloy coupons were observed. [1-6] At many test sites, more than one parameter influenced pit formation on the coupons. Comparison of data from the different sites revealed synergism in the effect of these parameters on Al corrosion. That is, the combined effect of two or more of the parameters on Al corrosion was greater than the sum of the effects of individual parameters.

3.1. Synergism in the effects of specific parameters on Al coupon corrosion

Conductivity and chloride ions

Pitting is a localized form of corrosion and occurs on metals that form a layer of surface oxide. Halide ions, and specially chloride ions, are known to cause pitting of Al alloys. [7] Direct correlations between chloride content and extent of pitting has been observed at many sites. [1,2] It is also well known that a few chloride ions are sufficient to initiate and propagate a pit on Al surfaces exposed to neutral pH water. The chloride ions penetrate the surface oxides at imperfections and initiate pits. The growth of pits is an autocatalytic process and the chloride ions are the catalyst. [7] Laboratory data and Al coupon evaluation data have indicated that even though no pits formed in chloride ion free neutral pH waters with conductivity of 10-20 µS cm^{-1} and in distilled water with chloride ions in the ppm range, pitting was observed in waters with even lower conductivity (~2 µS cm^{-1}) and with some chloride ions. [5,6] This indicated synergism in the effects of conductivity and chloride ion content on pitting corrosion of Al. Since chloride ions contribute towards the ionic conductivity of aqueous systems, any synergism in the effects of either parameter, conductivity and chloride ions, with that of any other parameter is discussed as a conjoint effect of conductivity and chloride ions.

Galvanic coupling and conductivity/chloride ions

Proofs of synergism in the effects of galvanic coupling and conductivity/chloride ions was evident when the contact surfaces of Al alloy coupons in a crevice couple and in a galvanic couple (in the same rack) were compared. The surface of the Al alloy in the crevice couple was stained with Al oxide but had no pits. However, the surface of the same Al alloy in contact with the stainless steel (SS) coupon revealed many pits. (Fig. 2). Further, even though pits were observed on the Al alloy surface a short distance away from the contact region with SS, none were seen on the flip side of the same Al coupon or on the surface of the single coupon of the same alloy in the same rack. (Fig 2) This lent further proof of synergism between the effects of galvanic coupling and conductivity/chloride ions. Since corrosion is an electrochemical process involving anodes, cathodes and an electrolyte, increased pitting of the Al coupon surface in the vicinity of SS is due to the latter functioning as a large area cathode.

(a) (b)

Fig 2. Pits on the Al coupon surface at regions: (a) in contact with the stainless steel coupon and (b) further away from the contact region.

Settled solids and conductivity/ chloride ions

Evidence of the effect of settled solids on corrosion of Al coupons and synergism in the effects of settled solids and conductivity/chloride ions were observed at various sites among the participants of the three projects. Two examples highlight the synergistic effects of settled solids and conductivity/chloride ions. (1) Civil construction in the vicinity of the RA6 reactor in Argentina lead to increase in airborne dust and consequent increase in dissolved solids and conductivity of the reactor and decay pools as shown in Fig 3. The construction also lead to an increase in the amount of settled solids on the Al coupons. The Al coupons withdrawn and examined in early 2000 revealed pits in apparently very good quality water. The pits were observed on the top surface only, and more pits were observed on coupons higher up in the rack. Parts of the same coupon without any settled solid (bottom surfaces) did not reveal any pits even though the coupon was exposed to identical conductivity oscillations. (2) In laboratory tests, an Al surface with hematite particles was exposed for 20 days to a solution with 40 ppm of NaCl. This surface did not reveal pits but some stains. However, a similar surface coupled to stainless steel and exposed to the same solution revealed pits after only 7 days.

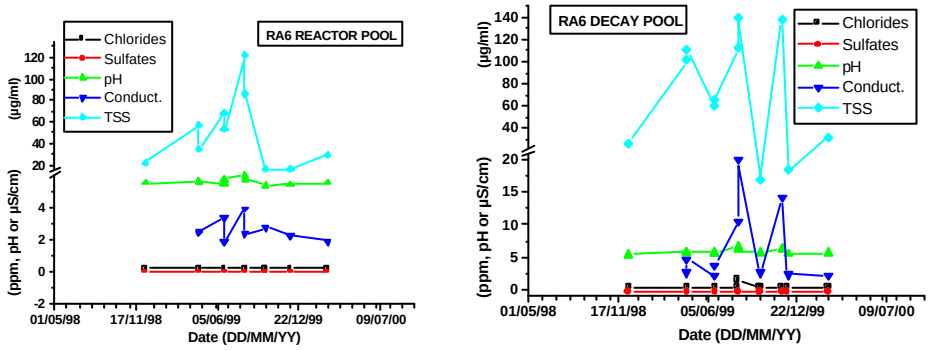
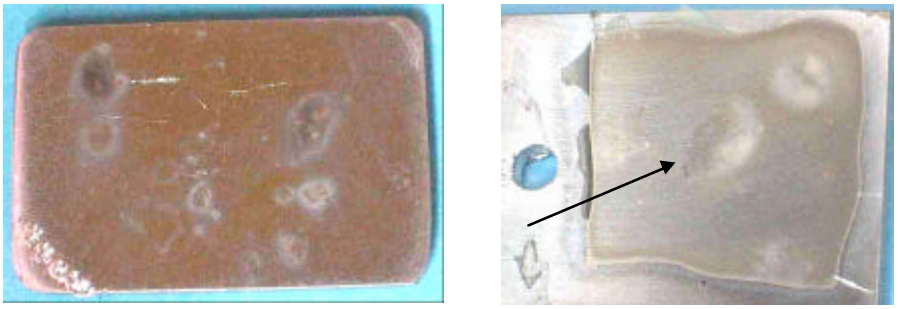


Fig 3. Variation of water parameters in the RA6 reactor pool (left) and decay pool.

Settled solid particle-induced corrosion of aluminium could be due to one or more of several reasons: (a) the nature of the solid and consequently the nature of products that could leach out; (b) formation of crevices under the solid and thereby crevice corrosion in the presence of aggressive ions; (c) the solids if conducting, could become the cathode and the cathodic reaction could cause a localized increase in pH and metal dissolution.



(a) (b)

Fig 4. Al alloy surfaces with hematite particles that were exposed to water with 40 ppm NaCl: (a) for 20 days – reveal stains but no pits; (b) for 7 days, but connected to SS – arrow reveals pits.

4. Concluding remarks

In spent fuel storage basins the four main parameters that lead to corrosion of the aluminium cladding of RR fuel are conductivity of the water, dissolved aggressive ions, galvanic coupling and settled solids. Separately these parameters do not cause significant pitting damage of the Al cladding. However, when two or more of these parameters are present or operate conjointly, there is synergism and the extent of corrosion of the Al cladding is much more than the sum of the effects of these parameters operating separately on corrosion.

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SPENT FUEL MANAGEMENT AT LVR-15 REACTOR

V. BROŽ, J. PODLAHA, J. RYCHECKÝ, M. HREHOR, P. NOVOSAD, M. MAREK, F. SVITÁK
Nuclear Research Institute Řež plc
Husinec-Řež, CP. 130, 25068 – Czech Republic

ABSTRACT

The system of the management of spent nuclear fuel (SNF) resulting from the operation of the LVR-15 research reactor is described in the paper. The Nuclear Research Institute Řež plc (NRI) jointed the Russian Research Reactor Fuel Return (RRRFR) programme under the US-Russian Global Threat Reduction Initiative (GTRI) initiative. The paper describes the experience gained during the preparatory works for the SNF shipment (facility equipment modification, cask licenses), preparation of shipment (SNF checking, repacking in a hot cell, loading into ŠKODA VPVR/M casks, drying, manipulation, completion of the transport documentation, transport of the casks to the SNF storage facility) and the shipment of SNF to RF. The paper also briefly describes a regulatory framework for these activities and legislative and methodological aspects of the return of vitrified waste back to the Czech Republic.

1. Introduction

The VVR-S reactor started its operation in 1957. The original EK-10 fuel was made up of rods of a 10 % enriched uranium dioxide-magnesium alloy in aluminium cladding. The fuel assembly (FA) consisted of 16 rods in an aluminium casing. The reactor was operated at 2 MW_{th} maximum output until 1969 when the power was increased to 4 MW_{th}. In 1974, the IRT-2M fuel with 80 % enrichment was introduced. This consisted of 4 or 3 concentric square tubes of uranium/aluminium alloy fuel/metal clad on either side with aluminium. The power output of the reactor was increased to 10 MW_{th}. In years 1988 – 1989 the reactor was reconstructed into the LVR-15 reactor. It was essentially a complete rebuild of the reactor vessel and internals, primary circuit, control room and ventilation system. In 1996, the IRT-2M fuel with 36 % enrichment used uranium dioxide was introduced. The maximum output of the LVR-15 research reactor is 10 MW_{th}.
SNF is removed from the reactor core to the at-reactor (AR) pool. The spent FA is loaded into the cask standing on the top of the reactor and then the basket with FA is slipped by the slide into the AR pool. Then SNF can be transferred to the away-from-reactor (AFR) pool with the cask. After two years of cooling, SNF can be transported to the AFR pool located in the High Level Waste Storage Facility (HLWSF).

2. History of spent fuel management

In the years 1969 – 1975, EK-10 SNF was transferred from the reactor site to the temporary store. SNF was held in dry storage drums. The SNF was then transferred to HLWSF between the years 1996-7 (see Fig. 1, Fig. 2). According to the period of storage, the character of construction materials of drums (carbon steel drum filled with concrete, carbon steel liner) and

the possible interaction with aluminium cladding, corrosion of the cladding had to be taken into consideration. It was decided to repack all EK-10 SNF into canisters.

Fig. 1. Temporary storage of drums with EK-10 SNF in the HLWSF

Fig. 2. View inside one EK-10 storage drum after plug removal

The new hot cell had been constructed in HLWSF and EK-10 SNF was repacked between the years 2006 and 2007 into stainless steel canisters, hermetically welded, put into a cask basket and then stored in a storage facility located close to the hot cell. Additionally, some leaked IRT-2M FAs were also repacked. The most of IRT-2M SNF was moved out of the initial AFR pool in the reactor building into the HLWSF pool between the years 1996 – 2003. The ŠKODA 1xIRTM transport cask for one FA was used.

3. Preparation for the shipment

The Czech Republic was included into the GTRI program in 2004. In 2005, the contract between US DOE and the NRI was signed. Preparation and implementation of the shipment of HEU SNF are very demanding and highly professional problems requiring cooperation of a number of organizations. With the significant technical and financial aid of the US Administration and the US DOE (total of approximate CZK 450 mil.), the Czech Republic shall become a pilot country, which will carry out such shipment from the NRI to the RF by means of special developed casks, which are compatible with the technology of research reactors of Russian design as well as the technology of the reprocessing plant in the RF.
Tender for such casks took place under the auspices of IAEA. Six famous manufacturers from the USA, RF, Germany, France and the Czech Republic participated therein. The ŠKODA JS a.s. was chosen as a supplier. Six ŠKODA VPVR/M casks were purchased by the NRI for shipment of LEU SNF; ten casks for shipment of HEU SNF were purchased by the US Administration (approximately USD 4 mil.) and by gift provided to the NRI provided that the NRI shall provide these as well as its casks for the RRRFR program. Once the shipment of SNF from the NRI is carried out, all 16 casks will be further used for return shipments of SNF from other countries to the RF by agreement between the NRI and the US DOE taking account of experience of the NRI from preparation and implementation of the transport from the Czech Republic to the RF.
In the following table, there is a list of all contracts, agreement and licenses necessary for SNF transport and reprocessing.

Tab 1. List of activities (documentation, contracts, licences)

	US DOE	Czech Rep.	RF	Slovakia	Ukraine	ESA
USA - RF GTRI Agreement	27-May-04		27-May-04			

NRI-DOE(NNSA) Contract	7-Mar-05	7-Mar-05				
CR-USA DipNote exchange	27-Apr-07	27-Apr-07				
CR-USA Min-to-Min Agreement	17-Sep-07	17-Sep-07				
Gov-to-Gov Agreement		04-Dec-94	04-Dec-94			
Gov-to-Gov Transport Agreement		14-Mar-98	14-Mar-98	14-Mar-98	14-Mar-98	
NRI – MAYAK Unified Project Contract		13-Jun-06	13-Jun-06			
Technical Conditions (Min Trans, Reg Body)		14-Sep-07	12-Oct-07	17-Sep-07	26-Oct-07	
NRI-TENEX Foreign Trade Contract		08-Aug-07	08-Aug-07			27-Aug-07
Package Design License		16-Feb-07	22-Dec-05	02-May-07	07-Aug-07	
Transport Permission		28-Jun-07	03-Aug-07	10-Aug-07	29-Oct-07	
CR-SK border physical protection exchange		27-Jan-04		27-Jan-04		
SK-UA border physical protection exchange				05-Oct-07	26-Oct-07	
UA-RF border physical protection exchange			12-Apr-96		12-Apr-96	
Agreement with Carrier		24-May-07	08-Aug-07	24-May-07	14-Aug-07	
Insurance for Nuclear Damage		28-Jun-07	08-Aug-07	28-Jun-07	28-Jun-07	
Czech Export License		28-Aug-07				
Russian Import License			30-Aug-07			
RF Governmental Decree			23-Nov-07			
Czech guarantee letter (RW return)		28-Aug-07				
Russian guarantee letter (nonproliferation)			21-Aug-07			
Transport technical documentation completed		17-Sep-07 (NRI+DMS)	17-Sep-07 (NRI+MAYAK)	17-Sep-07 (NRI+DMS)	17-Sep-07 (NRI+IZOTOP)	
Shipment	30 days before the scheduled shipping date MAYAK is informed about the					

4. ŠKODA VPVR/M transport and storage cask

The ŠKODA VPVR/M cask (see Fig. 3) is a type B(U) and S cask system designed and licensed for the transport and storage of SNF of Russian origin research reactors.

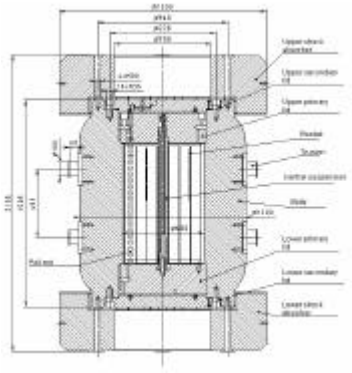

Fig. 3. Scheme of the VPVR/M cask.

Fig. 4. Manipulations with the cask

The VPVR/M cask loading procedure is divided into following activities:

- Cask transport to the SNF loading site, dismantling the cask (see Fig. 4)
- Transport of the cask to the SNF storage facility (pool, hot cell)

- Putting the basket inside the loading facility (pool, hot cell)
- Loading the SNF into the basket, basket retraction into the cask
- Cask flushing with hot air, desiccation of the cask, cask completion, helium leaking test
- Cask sealing by IAEA and EURATOM seals

The specially designed basket handling tool is used for lowering the basket from the cask into the storage pool (see Fig. 5, Fig. 6). The basket is filled manually with the FAs by a special manipulation rod. The crane and lift fixtures are equipped with a digital dynamometer that is used to monitor the weight of the basket during reinstallation into the cask. It prevents the disruption of the central suspension/hanger.

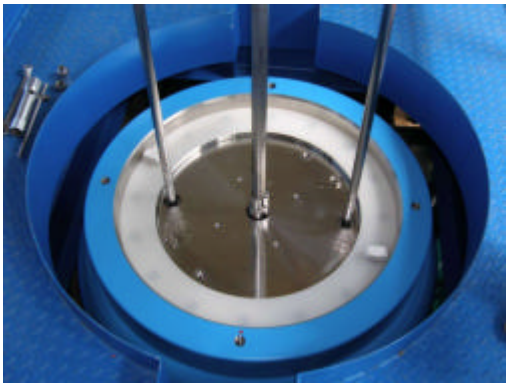

Fig. 5. Lowering of the basket

Fig. 6. Basket in the storage pool

5. Preparation for SNF loading

The VPVR/M cask underwent three demonstrations at the ŠKODA, NRI and Mayak facilities to verify that the design was acceptable technically, for handling and loading SNF at the research reactor facilities, and receipt and unloading at the Mayak facilities.
Preparation before the first cask loading included:
- Preparation of calculation and assemble data files for each FA and transmit to Mayak for acceptance (252 + 91 IRT-2M FAs, 206 canisters with EK-10 FAs / fuel rods).
- Preparation of documentation for loading and storage of the VPVR/M casks in NRI.
- Negotiations with the State Office for Nuclear Safety, IAEA and EURATOM about verification of the loading FAs and cask sealing by their inspectors were done.
- Preparation of fuel and cask handling equipment and facilities for operations.
- Sipping test and visual inspection of all FAs.
- Preparation for installation of ancillary equipment, cask manipulations training.

6. Loading SNF into the ŠKODA VPVR/M casks

The SNF loading was performed in 2007. A specially designed cask transport carriage, which moves by rail was used for loading operations at the reactor site. It serves for transferring the cask from the reactor hall to the reactor annex with the pool for SNF storage. Also, the shielding above the pool was used during the loading of the SNF protect the workers from radiation when the loaded basket was lifted out of the water and before it is completely inside the cask. Three casks were loaded at the reactor site with 91 FAs IRT-2M (36 %) and 10 FAs IRT-2M (80 %). The casks were then transported to HLWSF.
Six casks were loaded with 206 canisters with EK-10 FAs/fuel rods from the HLWSF hot cell. Seven casks were loaded with 242 IRT-2M (80 %) (235 FAs from the HLWSF pool and 7

repacked FAs from the hot cell). The casks were positioned onto the pool into the special platform. Loading the IRT-2M FAs was performed using a manual manipulation rod.

7. Transport of SNF

Before transport, the transportation documentation had to be prepared and assemble and all necessary licenses of transport had to be acquired. The transport of SNF were realised from the Czech Republic to the RF across transit countries Slovakia and Ukraine by combined rail and road transport. The transport was performed in December 2007. The VPVR/M casks were loaded from the HLWFS storage area into the ISO containers (see Fig. 7). The ISO containers were transported to the railroad station on trucks and then were transferred onto the railroad carriages (see Fig. 8). The physical protection and emergency preparedness were secured during the transport.

Fig. 7. ISO container with one VPVR/M cask

Fig. 8. Transfer of ISO containers onto the railroad carriages

8. Legislative and regulatory framework

The preparation of the transport of SNF to the RF included assuring compliance with a number of legislative and regulatory requirements contained in the Atomic Act and its implementing regulations.
The commitment on the return of waste from the reprocessing of SNF in the RF is included in the Amendment of the Russian-Czech Intergovernmental Agreement on Co-operation in Nuclear Energy of 15.4.1999 (Coll. No. 154/1999), referred to in the preamble of the Foreign Trade Contract (FTC) between the NRI and Tenex. A licence for re-import of vitrified waste back to the Czech Republic represents a challenge for both the NRI and the regulatory body, as it will be the first time when an application of this type will be dealt with. According to FTC such a licence will have to be issued in 2026. While there is enough time for planning such a return, setting down requirements for composition, physical parameters and properties of this waste was urgent as the vitrification of waste will be done rather soon - in 2009. Pursuant to this, it was necessary to address the return of waste in the FTC very carefully with an assumption of extrapolating existing “legislation” to the period around the year 2025.

9. Conclusions

All SNF have already been shipped to RF. Contract between the NRI and the US DOE for participations of the NRI in shipments from other countries (Custodian Agreement) will continue. The second shipment of the residue of HEU SNF from NRI (133 FAs) after changeover of the reactor operation to LEU fuel will be implemented in 2015.

10. Acknowledgement

We would like to thank to the staff of all organizations involved in the project: USA (DOE – NNSA), IAEA, EUROATOM, Russian Federation (Rosatom, Mayak, Tenex, Sosny), Ukraine (Izotop), Czech Republic (NRI, ŠKODA JS, DMS). Similarly, we would like to thank to the staff of the bodies of the state administration, ministries, and regulators in the Czech Republic and Russian Federation as well as in transit countries Slovakia and Ukraine.

PREPARING ANSTO'S FINAL HIFAR RESEARCH REACTOR MTR SPENT FUEL SHIPMENT

MICHAEL ANDERSON
Leader Fuel Management, ANSTO
New Illawarra Road, Lucas Heights
PMB1 Menai, NSW, 2234 Australia

LUBI DIMITROVSKI
Manager Waste Operations, ANSTO
New Illawarra Road, Lucas Heights
PMB1 Menai, NSW, 2234 Australia

KATHERINE LUCAS
Process Engineer, ANSTO
New Illawarra Road, Lucas Heights
PMB1 Menai, NSW, 2234 Australia

ABSTRACT

In her life of nearly 50 years, the HIFAR Reactor at ANSTO used 2280 Fuel Elements, of which 2122 uncompromised spent elements have been shipped overseas for reprocessing or final disposition. Following the closure of the HIFAR reactor, ANSTO is preparing for the shipment of the remaining spent elements, 19 of which are considered 'questionable' in terms of suitability for shipment without further preparation due to various levels of corrosion. Four of these elements had been subjected to destructive examination and separated into plates. This paper describes ANSTO's activities in the preparation of these elements for shipment, with particular emphasis on leach testing in storage ponds and methods of evaluation.

1. Introduction

Australia's HIFAR research reactor at Lucas Heights, NSW was a heavy water moderated, light water cooled high neutron flux reactor which operated from 1958 - 2007. The reactor was powered by 25 MTR type fuel elements and operated at 10 megawatts.

Following removal from the reactor, the spent fuel elements were initially wet stored for a minimum period of 21 months and then transferred to dry storage tubes, mostly under a nitrogen atmosphere. The storage and monitoring methods employed were generally highly successful in ensuring the integrity of the spent fuel elements (SFEs). Of the 2280 fuel elements used, 2261 were maintained in sound condition, with all cladding having total integrity and with no fission products being released. Of these sound elements, 2122 have been shipped to Scotland (Dounreay, 1963 and 1996), the USA (Savannah River, 1998 and 2006) and France (Cogema, 1999, 2001, 2003 and 2004). The remaining 139 sound elements are scheduled to be shipped to the United States in the next few years. There were, however, 19 'compromised' elements where there was some evidence of corrosion, four of which had been dismantled into separate plates for destructive examination. Until recently, these elements resided in dry storage to be dealt with at the end of the life of HIFAR. ANSTO's initial preference was to load these elements directly from dry storage into the casks, and hence containment or encapsulation of the individual elements was considered likely. However, in mid 2007 it was decided to employ the standard method of 'soaking' the 19 elements in the storage ponds to reduce the leach rate to an acceptable level for regular shipment.

2. Spent Fuel Elements

During HIFAR's operating life, a number of different MTR fuel assembly designs were employed, each of which was comprised of an enriched uranium-aluminium cast alloy fuel meat (95% volume Al) dispersed in aluminium metal within high purity aluminium cladding [3]. Each assembly consisted of curved rectangular fuel plates, with the fuel meat surrounded by a picture frame and metallurgically bonded between two sheets of Grade 1050 (99.5% Al) aluminium by hot rolling. Each fuel plate was tested at 600°C for 20 minutes [4] and examined for blisters after cooling to check the integrity of the bond between the meat and cladding. The cladding method prevented the release of uranium and fission products during burn-up, and ensured that in the event of a minor breach to the cladding, the only part of the fuel meat exposed to water was directly under the breach. Highly enriched uranium elements (up to 93% ^{235}U) were used until 2005 when HIFAR converted to using low enriched uranium elements (20% ^{235}U) for the final years of operation.

3. Compromised Spent Fuel Elements

The condition and handling/storage of the 19 'compromised' SFEs are described in the following table:

	Condition	Cause
4 SFEs EC 270 (Mk II), UED 609, 801 and 802 (Mk III),	42 individual plates with punched holes in some plates. Most plates in sound condition, some with minor pitting corrosion and ~10 with local and general corrosion	Assemblies separated into individual plates, discs punched from a selection of plates for through thickness metallographic examination (1967)
2 SFEs 45c 045 and 45c 048 (Mk IV)	Compromised structural integrity of some fuel tubes in 45c 045. Structural integrity of 45c 048 borderline. Localised corrosion damage to non-fuel bearing regions of the fuel tubes and some localised pitting corrosion has exposed fuel in some areas.	Corrosion damage from water ingress into dry storage facility through faulty seal
13 SFEs	Structurally sound. Some with visible pitting corrosion on cladding and/or minor corrosion nodules on plate surface. Others with more extensive corrosion and small areas of exposed fuel meat.	Exposure to moisture during dry storage

Tab 1. Condition of compromised SFEs

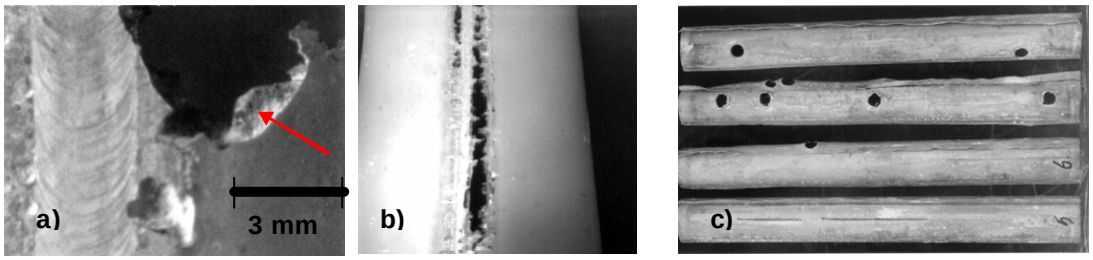

Fig 1. a) The corner of the fuel meat has been exposed at the top of a plate from assembly 45c 045. b) extensive pitting corrosion in the non-fuel section of assembly 45c 045, resulting in compromised structural integrity. c) Punched plates from UED 802.

The four dismantled elements were originally examined to determine the cause of abnormally high heavy water activities in the primary cooling circuit. They were sealed into airtight stainless steel containers in 1996.

Fuel elements SFE 45c 045 and 45c 048 were examined through a binocular microscope at up to 13 times magnification. The structural integrity of some fuel tubes of assembly SFE 45c 045 is compromised due to extensive pitting corrosion, and the element will need to be placed in a protective can for shipment. Only one or two small areas of fuel meat are exposed (Fig 1. a), as distinguished by the darker colour of the fuel meat against the lighter aluminium cladding. Given the metallurgical similarity of the meat and cladding, there is generally no preferential corrosion of the meat even when the cladding has been corroded through. With HIFAR type fuel, the evidence is that the non-fuel areas experience preferential galvanic attack and thereby protect the fuel meat from corrosion. SFE 45c 048 is also compromised with extensive corrosion to the cladding and some small areas of exposed fuel meat and will also be placed in a protective can for shipment.

The remaining 13 SFEs are structurally sound and considered suitable for shipment in standard NAC-LWT transport casks following leaching and SIP testing.

4. Leaching Method

All of the 19 'questionable' SFEs were subjected to leach testing in two 35 m^3 ponds of demineralised water with no other fuel present. The objectives were to assess the fission product release and if possible to confirm sufficient leaching of the fuel meat to form a passivation barrier layer that would prevent further leaching.

The SFEs were shared equally between the two ponds by revealed fuel meat surface area and placed in the ponds in a sequence from smallest to largest revealed fuel meat surface area. The 13 structurally sound SFEs were placed in the Building 23 pond and the 4 dissembled SFEs and 2 SFEs with questionable structural integrity were placed in the Building 41 pond.

Each pond was treated by an Ion Exchange (IX) system with 28 litres of mixed bed resin in a 3:1 ratio of anionic Amberlite IRN 78 and cationic Amberlite IRN 77. On the basis of a SFE with revealed fuel area of 3.23 cm^2, it was calculated that the expected leaching rate into the pond of ^{137}Cs would be 1055 Bq/hr. The required ion exchange flowrate to accommodate the absorption of ^{137}Cs was calculated to be 200 L/min per pond, however, the available capacity for each pond was only 10 L/min. The pond chemistry was tested on a regular basis to check that the pH and conductivity were at the required levels ($5<pH<8$ and $1<$conductivity<10 µSiemens/cm) and to determine the release of ^{137}Cs leached from the fuel.

5. Leach results

5.1 13 SFEs in B23 pond

Gamma spectrometry analysis of pond water for ^{137}Cs was commenced after the introduction of 6 SFEs with minor pitting on the 11/7/07 and 16/7/07. The first ^{137}Cs activity measured on 18/7/07 was 43 Bq/L, which decreased to 13 Bq/L over the week before the next addition of SFEs (Fig 1.). With the additions on 25/07/07, the ^{137}Cs activities rose to 21.8 Bq/L and then dropped to 14 Bq/L within a week. Following the additions on 7/08/07 and 9/08/07, the ^{137}Cs activities rose to 64.8 Bq/L and then dropped to 10 Bq/L within a fortnight. There were no samples taken between the additions on 15/9/07 and 26/9/07; however, it is a reasonable assumption that there would have been a similar pattern in the increase and decrease of pond activity following the addition made on 15/9/07. There was a large increase of ^{137}Cs activity in

the pond to 678 Bq/L following the final addition of heavily corroded or cut SFEs on 26/9/07, but again, the activities decreased appreciably over the following 10 weeks.

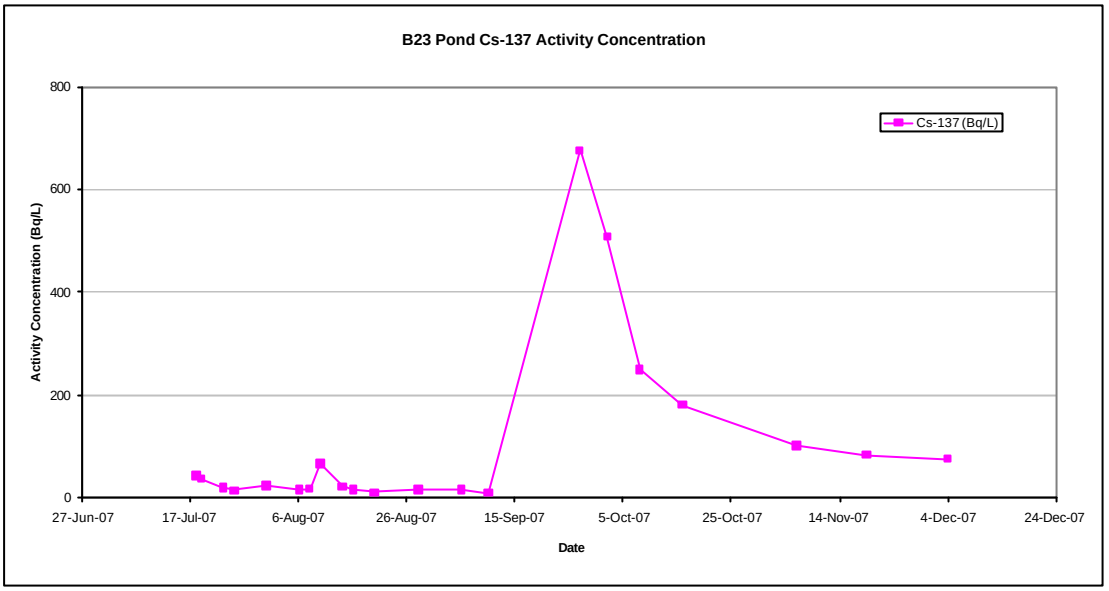

Fig 2. Cs-137 Activity Concentration in Building 23 pond

5.2 6 SFEs in B41 pond

Following introduction of 2 SFEs with unsound or borderline structural integrity on 3/10/07, the pond activity increased from normal background levels to 310 Bq/L ^{137}Cs and declined before the next addition (Fig 2.). Following the additions of the 4 dissembled SFEs from 9/10/07 – 15/10/07, there was a sharp rise of ^{137}Cs activity to 711 Bq/L. The activity was reduced by 55% within the first week and 8 weeks after the final SFE additions, the pond activity had decreased to 113 Bq/L.

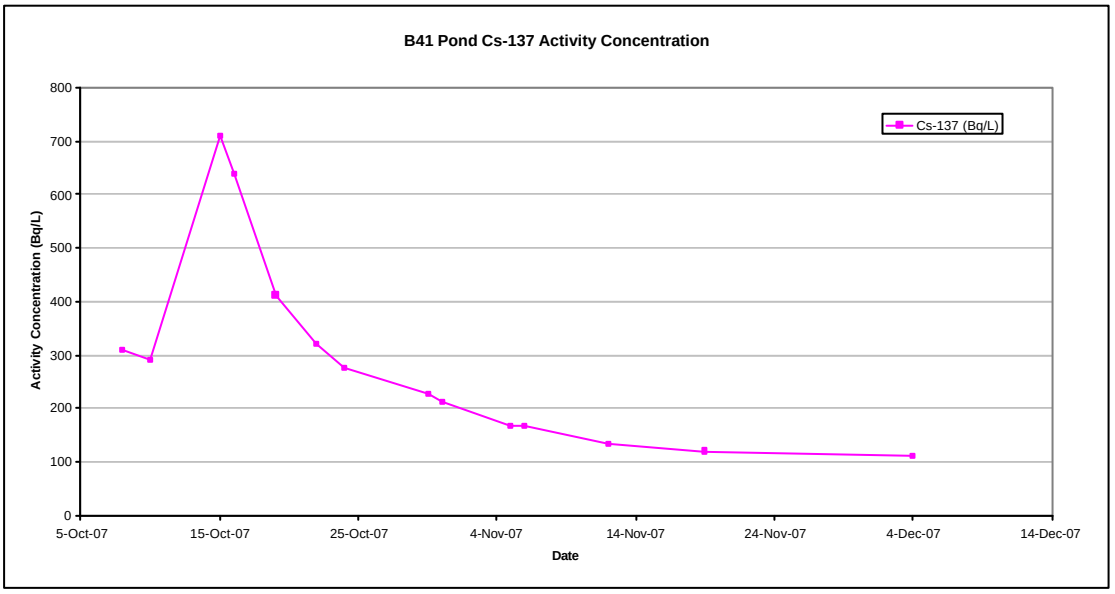

Fig 3. Cs-137 Activity Concentration in Building 41 pond

6. Discussion

There were only large increases of ^{137}Cs activity (~700 Bq/L) following the addition of SFEs with significant areas of exposed fuel meat. It is surprising that the pond activities were rapidly reduced, given that the actual IX flow rate was only 5% of the estimated rate required to accommodate the spent fuel leach rate. It can be seen from the gamma spectrometry results that the IX take up of ^{137}Cs was greater than the production.

It has been previously found [1] that when compromised hot rolled uranium-aluminium alloy SFEs are placed in an IX purified pond, leaching of ^{137}Cs and other fission products occurs initially and an impervious scale is then formed on the exposed fuel meat over ~ 5 years in successive wet and dry storage. This scale of up to 4 μm is a hard non-reactive surface film which acts as a passivation layer to inhibit further corrosion. The amount of fuel meat revealed, enrichment of fuel, time out of reactor and burn up percentage have an effect on the time it takes for this scale to form. The time of SFE immersion subjected to IX is an important factor - the longer the period, the more effective will be the passivation of the fuel. The quantity of water does not directly effect the scale formation; however it is critical that the water is kept free of aggressive chemistry. It is likely that the use of different IX resin compositions could have a substantial effect on the passivation process.

From the leach results, it is evident that with aging in storage, some passivation of the ANSTO SFEs has occurred and thus sealed the areas of exposed fuel meat.

7. Further Testing and Fuel Packaging for Shipment

A 12 hour SIP test for ^{137}Cs will be required to evaluate the leach rate of the 19 compromised SFEs in the absence of IX purification. It can then be determined if the quantity of leaching will be acceptable for shipment in a standard Fuel Shipment cask. The structurally unsound fuel elements will be placed within special containers inside the casks to capture any "fines."

8. Conclusions

The leach testing procedure employed shows that the fission products leached from the compromised SFEs are adequately taken up by the pond IX system. The results also indicate that some passivation of the exposed fuel meat has occurred during storage. Further leach testing (SIP) of the compromised SFEs will be required to determine if the leach rate will meet acceptance levels for shipment in a standard Fuel Shipment cask.

9. References

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[2] "Basis for Containment Analysis for Transportation of Aluminium Based Spent Nuclear Fuel", Westinghouse Savannah River Company WSRC-TR-98-00317, October 1998.

[3] R. Finlay, "Degraded Fuel to be shipped to SRS", ANSTO report, July 2006.

[4] A. B. Wilde, "MTR Fuel Fabrication Plant Specification for the Fabrication of MTR Mark 4/23 Fuel Elements with Electron Beam Welded Highly Enriched Uranium/Aluminium Alloy Fuel Tubes for the ANSTO Reactor HIFAR (DIDO), MTRS 9 (FELC)", UKAEA, June 1995.

INTERNATIONAL TOPICAL MEETING ON RESEARCH REACTOR FUEL MANAGEMENT (RRFM) - UNITED STATES FOREIGN RESEARCH REACTOR (FRR) SPENT NUCLEAR FUEL (SNF) ACCEPTANCE PROGRAM: 2008 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, now extended to expire May 12, 2016, to return research reactor fuel to the U.S. is in its twelfth year. This paper provides a brief update on the program, now part of the National Nuclear Security Administration (NNSA), and discusses program initiatives and future activities. The goal of the program continues to be recovery of U.S.-origin 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), which is part of NNSA's Office of Global Threat Reduction (GTR). After an initial discussion of program history, contract issues are discussed. Planning issues are then set out to incorporate lessons learned from recent shipments in order to help FRRs understand issues which can affect their SNF disposition project. 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 twelfth year of implementation, has completed forty one (41) shipments to date, safely and successfully, and another is expected to be completed in spring 2008. Twenty-seven countries have participated so far, returning a total of 8,078 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. Thirty two (32) of the 41 shipments contained aluminum-based spent nuclear fuel from research reactors and were placed into storage at the Savannah River Site (SRS) in South Carolina. Two shipments have been forwarded on to the Y-12 National Security Complex, since the fuel was fresh or slightly irradiated and eligible for receipt at that facility. The remaining seven (7) shipments consisted of Training, Research, Isotope-General Atomics (TRIGA) type fuel and were placed into storage at the Idaho National

Laboratory (INL). The most recent shipment was completed without incident, arriving at SRS on December 11, 2007. During the remaining calendar year (January - December 2008), the program is planning to receive as many as five shipments of SNF from various locations.

3. Contractual Requirements

3.1 Contract Extensions

Research reactors which have converted to LEU fuel will need a contract extension to authorize shipments after May, 2009. Reactor Operators who intend to ship after May 2009 and do not have a modified contract should contact the Acceptance Program office to negotiate the extension of the DOE-FRR 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 maintained. One important article which has been misunderstood in the past involves compliance with U.S. government regulations restricting public disclosure of any shipping plans, shipment information, or individual details comprising such plans or information. Compliance with this article is an important obligation to ensure security for all shipment activity. Any press release made prior to the material reaching the storage site, even after the ship reaches international waters on the way to the United States, is a violation of the contract which makes the security of the shipment more vulnerable. Premature release of shipment related information also violates the U.S. 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 this critical and agreed-to process.

4. Focus on Early Planning

The FRR SNF Acceptance Program focuses on the planning and implementation of research reactor spent fuel shipments to the United States in support of worldwide nuclear nonproliferation efforts, while allowing 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 programmatic importance.

4.1 Shipment Scheduling

It is always important that NNSA clearly understands the intentions of all Reactor Operators 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 adequate time for the receiving site 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, when required. Budget limitations have been known to challenge implementation of shipping plans for our customers and NNSA. Similarly, the Department of Energy receiving facilities also face increasing challenges in providing resources to receive material, particularly when reactor operator's 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 extension was granted for the benefit of FRRs needing justifiable relief. Some other FRRs are cancelling or rescheduling shipments in order to defer costs. This was not the intent of the program extension. These delays impact DOE's ability to maintain a regular schedule of operations and adequate resources for the receipt facility. The FRRs are strongly encouraged to continue shipping as early as possible and maintain original schedules where possible. Deferring shipments when spent fuel is available for shipping could result in changes to DOE's ability to support the receipt of fuel when a shipment is desired by the customer. Also, as the Acceptance Program approaches the end of the policy period, a large amount of shipments are expected. DOE may be required to exercise its authority to limit receipts to specific customers with the greatest need.

4.2 Insurance Issues

Insurance issues have been a recurring 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 occurs 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 determine 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. Recently, we have experienced better results for some customers with aggressive coordination. It is also important to be conscious of this potential problem and budget for any added cost that cannot be mitigated.

4.3 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 license 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. Because there are limited NRC resources for review of cask licenses, our customers need 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, 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 or changes from past shipments such as a change in fuel type or fuel condition. 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 Source Recovery

The recent shipment of SNF from Argentina and Brazil provided an opportunity for an ISO container with Sources from South America by the U.S. Radiological Remove division of GTR to be transported to the U.S. on the same vessel used to ship SNF. This is an excellent opportunity for the customer or other organizations in the customer's country or surrounding countries to dispose of unwanted radioactive sealed sources, particularly sources that can not be transported by air transport.

The U.S. Department of Energy (DOE) sponsors a program to recover excess and unwanted radioactive sealed sources presenting disposal difficulties. Traditionally, the program dealt largely with americium-241 and plutonium sources. Owing to heightened concerns about terrorist threats to steal radioactive material for use in a dirty bomb, the DOE is moving aggressively to include other isotopes of concern. The DOE is currently emphasizing larger excess sources containing cobalt-60 and cesium-137, such as medical irradiators. The DOE is also considering a campaign to manage large numbers of small obsolete sources, examples of which are cesium-137 brachytherapy sources, and various radium-226, americium-241, and other sources. To be considered, institutions must register their material with Los Alamos National Laboratory.

Reactor operators and other Stakeholders should consider this opportunity and also communicate with and assist in any coordination within their country or region. To learn more and register online, please visit osrp.lanl.gov

5.2 Material Disposition

The DOE's Office of Environmental Management (DOE-EM), the previous organization to manage the FRR SNF Acceptance Program, is making strides to further disposition the repatriated spent nuclear fuel. DOE-EM 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's Acceptance Program staff remains committed to working with staff in other program offices within DOE to assist in smooth transition 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)

European Nuclear Society
Rue de la Loi 57
1040 Brussels, Belgium
Telephone +32 2 505 30 54
Fax + 32 2 502 39 02
rrfm2008@euronuclear.org
www.euronuclear.org