IAEA-TECDOC-1080

Procedures and techniques for the management of experimental fuels from research and test reactors

Proceedings of an Advisory Group meeting held in Vienna, 1–4 December 1997

April 1999

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The originating Section of this publication in the IAEA was:

Nuclear Fuel Cycle and Materials Section
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100
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PROCEDURES AND TECHNIQUES FOR THE MANAGEMENT OF EXPERIMENTAL FUELS FROM RESEARCH AND TEST REACTORS
IAEA, VIENNA, 1999
IAEA-TECDOC-1080
ISSN 1011–4289

Printed by the IAEA in Austria
April 1999

FOREWORD

Almost all countries that have undertaken fuel development programmes for power reactors, research reactors and military reactors have experimental and exotic fuels, either stored at the original research reactor where they were tested or at some away-from-reactor storage facility. These historic spent fuel liabilities cannot follow the standard treatment recognized for modern power reactor fuels. They include experimental and exotic fuels ranging from liquids to coated spheres and in configurations ranging from full test assemblies to post irradiation examination specimens set in resin.

To obtain an overall picture of the extent of the problem of managing these fuels, an Advisory Group Meeting on Procedures and Techniques for the Management of Experimental and Exotic Fuels from Research and Test Reactors was convened in Vienna, 1–4 December 1997. This document contains the proceedings of the meeting and an expert evaluation of the overall situation in countries which participated in the meeting.

The IAEA wishes to thank all of the participants in the meeting for their contributions to this document. The IAEA officer responsible for the organization of the meeting and for the compilation of this document was I.G. Ritchie of the Division of Nuclear Fuel Cycle and Waste Technology.

EDITORIAL NOTE

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SUMMARY OF THE ADVISORY GROUP MEETING

1. INTRODUCTION

The IAEA's programme on the safe handling and storage of spent fuels from research and test reactors has been designed to collect, evaluate and disseminate information on the safe and reliable handling, management and storage, or preparation for shipment, of spent fuel from research and test reactors. As a first step in this programme, a database on spent fuel from research and test reactors has been compiled and is continually maintained. This database contains information on spent fuel inventories, types of fuel stored, their enrichment, country of origin, dimensions, types of storage, and concerns of the owner/operators. Of the more than 56,000 fuel assemblies at 210 facilities for which information has been received, about 18% are of non-standard types. These pose special problems for continued safe storage and eventual final disposition.

The IAEA held an Advisory Group Meeting (AGM) on the Management and Storage of Spent Nuclear Fuel from Research and Test Reactors, in Vienna from 1 to 4 November 1994 which dealt exclusively with standard fuel types (IAEA-TECDOC-900). Also, a Workshop/Technical Committee Meeting on Procedures for the Safe Management and Storage of Failed Fuels from Research and Test Reactors was held in Budapest from 29 to 31 October 1996, which dealt exclusively with failed standard fuel types. At these meetings, experts from facilities all over the world frequently mentioned the fact that there were other fuels, usually of an experimental nature and in various quantities, stored at their facilities which posed special problems. These had not been addressed adequately in any of the previous IAEA programmes. To address this issue, an AGM on Procedures and Techniques for the Management of Experimental and Exotic Fuels was convened from 1 to 4 December 1997 in Vienna. Nine nominated experts and one observer attended the meeting from nine countries. This report comprises their expert evaluation of the overall situation in the countries they represent and a summary of their deliberations, conclusions and recommendations.

2. DEFINITION OF EXPERIMENTAL FUELS

For the purposes of this meeting, experimental fuels are defined as materials containing fissile atoms that have been used in experiments to develop nuclear fuel and materials that are no longer fully contained or whose containment is suspect.

Examples of experimental fuels are:

1. Non-standard oxides	Enrichments above 5% for uranium and above 6% for plutonium. This fuel type also includes the U/Pu fuel.
2. Graphite matrix fuels	Fissile materials incorporated in graphite to form the main driver fuel. This includes AVR, THTR, Fort St. Vrain and Dragon type fuels.
3. Metal alloys	Uranium and plutonium alloys including those stabilized by additives such as Mo, Nb and Zr.

4. Thorium	The full scope of the thorium fuel cycle is included. Fuels include U/Th and Pu/Th.
5. Carbides/Nitrides	Generally fuels from fast reactor development, but may eventually include MTR high-density fuels.
6. ^{233}U	From tests on materials recovered from the Th fuel cycle.
7. Na-bonded	Metallic fuel types from the fast reactor development programme.
8. Failed fuels	Fuels that have failed in reactor or in storage.
9. Specimens	Post-irradiation examination (PIE) specimens mounted in resins, including any PIE remnants.

3. SCOPE OF PROBLEM

During more than forty years of operation of research and test reactors, there have been a large number of fuel types and fuel designs which have been irradiated. Similar programmes have been undertaken in many countries, resulting in small amounts of spent fuel that cannot be considered as standard. In addition to this material, in some cases, the method of storage of standard fuels has resulted in fuel failure creating additional problems for final disposition. The types of fuel identified by the participating experts are listed in Tables I–III. These tables also show which of the participating countries have the forms of fuels and their current methods of treatment.

4. OBJECTIVES OF MEETING

One important goal of the meeting was to evaluate the overall situation of the management, storage practice, problems, and concerns with experimental and exotic fuels from research and test reactors. This included evaluation of the types of fuel, amounts, enrichments, forms and packaging, if any, of these fuels in the countries represented at the meeting. In doing so, the participants were able to exchange information and to benefit from each other's experiences to date in coping with the special problems associated with the safe management of experimental and exotic fuels. An equally important goal of the meeting was to examine the management and disposition plans for these fuels in the participating countries.

Nine "country" reports were presented and they are reprinted in their entirety in this publication. In disseminating this IAEA-TECDOC to research reactor owner/operators and facilities where spent research reactor fuels are stored throughout the world, it is hoped that the whole research reactor community will benefit from the information herein. Furthermore, it is hoped that facility operators will be prompted to plan very carefully their future storage and final disposal of these problematic fuels.

Table I: INVENTORY OF NON-STANDARD FUELS

	Belgium	**Canada**	**China**	**France**	**Germany**	**India**	**Russia**	**UK**	**USA**
Non-standard oxides	✔	✔	✔	✔	✔		✔	✔	✔
Silicides		✔		✔	✔				✔
Graphite matrix fuels		✔			✔		✔	✔	✔
Metal alloys	✔	✔		✔	✔		✔	✔	✔
Thorium		✔			✔			✔	✔
Carbides/Nitrides		✔		✔	✔	✔	✔	✔	✔
U^{233}					✔	✔		✔	✔
Na-bonded								✔	✔
Failed fuels	✔	✔	✔	✔	✔		✔	✔	✔
Specimens	✔	✔	✔	✔	✔	✔	✔	✔	✔

Table II. DISPOSITION OPTIONS FOR NON-STANDARD FUELS BY COUNTRY

	Belgium	**Canada**	**China**	**France**	**Germany**	**India**	**Russia**	**UK**	**USA**
Current reprocessing capability				✔		✔	✔	✔	✔
Plans for home reprocessing			✔	✔		✔		✔	
Reprocess by others	✔			✔	✔			✔	
Interim wet Storage	✔	✔	✔	✔	✔	✔	✔	✔	✔
Interim dry storage	✔	✔	✔	✔	✔	✔	✔	✔	✔
Active repository program	✔	✔		✔	✔		✔	✔	✔
Repository open <25 years				✔	✔		✔		✔
Return to country of origin					✔				
Foreign return program (accept from others with no waste return)		DD*							✔

*DD: Decision deferred.

Table III. DISPOSITION OPTIONS BY FUEL TYPE

	Non-standard oxides	Graphite matrix fuels	Metal alloys	Thorium	Carbides/ nitrides	U^{233}	Na-bonded	Failed fuels	Specimens
Demonstrated reprocessing capability (commercially available)	✔		✔		✔			✔	✔
Could be reprocessed	✔	✔	✔	✔	✔	✔	✔	✔	✔
Interim wet storage	✔		✔	✔	✔	✔	✔	✔	✔
Interim dry storage	✔	✔	✔	✔	✔	✔	✔	✔	✔
Direct disposal probability	Medium	High	Medium	Medium	Low	Medium	Low	Medium	Medium

5. PAST AND CURRENT MANAGEMENT PRACTICES

Reprocessing is a proven technology for dealing with many types of research reactor spent fuel. Although even for some standard types, e.g. TRIGA fuel, it has only been carried out on a demonstration scale. Some countries have taken advantage of this option for standard MTR fuels, but it has not been used in many cases for small quantities of experimental spent fuels. In addition, politically or economically it may be difficult to justify, especially for the small quantities of research reactor fuel involved. Transportation of the fuel to the reprocessing plant and the return of waste, even in those few cases where this option is available, is prohibitively expensive for most research reactor facilities. Moreover, in many cases, especially in countries with a research reactor but no nuclear power programme, the infrastructure to take back high level waste from reprocessing is not available. This situation is aggravated by the continual ratcheting of transportation standards, not for safety reasons, but for public relations concerns.

Thus, it is hardly surprising that interim storage has been in the past and still is the management strategy of choice at the overwhelming majority of research reactor facilities. It is a short term, relatively low cost option that keeps treatment strategies open, but delays positive action. The purpose of safe interim storage is to maintain the integrity and the retrievability of the fuel without further degradation for a well-defined or undefined period.

To avoid any degradation during storage, one should ensure the compatibility of the fuel or external container materials with the external and internal wet or dry environment. This requires a detailed knowledge of all potential corrosion mechanisms.

Failed and unpackaged, and declad fuels should normally be canned to maintain the quality of the storage environment or to satisfy any licensing/environmental requirements, unless there are mitigating reasons to nullify this action. Appropriate monitoring of the storage environment should also be provided with remediation provisions available in case of detected outer containment failure. Several countries reported failures of canister seals in cases where fuel was canned dry and the cans were then placed in water. In these cases, accelerated corrosion and rapid degradation of the fuel have been noted. Since monitoring of canned fuel is very difficult, placing canned fuel into wet basins should be carefully considered.

Direct disposal has been investigated for standard fuels (mainly from power reactors) and is being investigated further. However, a disposal facility license could be jeopardized if many different fuel types are added to the base case. For the most advanced studies of geological repositories, the problems associated with the much higher levels of enrichment of research reactor fuels compared with power reactor fuels have not been studied in detail. This effectively means that without further long and expensive studies, some form of pre-dilution treatment of research reactor spent fuel may be required before they can be considered for final disposition in a geological repository designed for power reactor fuel.

6. CONCERNS AND MITIGATION

The diversity of research reactor fuels and storage practices in the past has resulted in a unique set of concerns associated with the long-term management of these fuels. Some of these concerns together with possible mitigation strategies to address these concerns are presented below.

Concern	Mitigation
1. Lightly irradiated fuel and/or fuel that has been discharged from research reactors for longer periods may have radiation levels less than 1 Gy/h at 1 metre in air. Physical security and transportation of these fuels may require stringent physical protection measures in accordance with Category I recommendations in INFCIRC/225/Rev.4.	• Establish and maintain accurate inventory. • Ensure that the design of interim storage facilities for enriched fuels meets the requirements for the categorization appropriate to the physical protection of the nuclear materials as per INFCIRC/225/Rev. 4. • If reprocessing is an option, reprocess this material before its physical protection categorization needs to be upgraded.
2. Loss of historical data and/or quality of data.	• Establish spent nuclear fuel (SNF) databases of uniform design. • Collect and protect existing sources of data. • Adopt minimum uniform quality assurance requirements. • If necessary, undertake sampling and NDA.
3. Funding is not always adequate to support interim management and ultimate disposition.	• Identify problem scope and prioritize. • Ensure that decision-makers are aware of consequences of past practices. • Identify the risks of inadequate precautions.
4. Small quantities of fuels that are difficult to direct dispose may not be economical to reprocess commercially.	• Collect data on these fuels from all sources. • Evaluate direct disposal potential for these fuels. • Evaluate central treatment of these fuels.
5. What to do with the fissile material returned after reprocessing.	• Reuse of HEU in research reactor fuel. • Civil reuse of Pu. • Blend-down and civil use of uranium.
6. Ultimate disposition of PIE debris and specimens (epoxy).	• Removal of epoxy with existing treatment technologies followed by reprocessing. • Disposal in high integrity can.
7. Acceptability of enriched fuels for direct disposal.	• Continued research on direct disposal. • Ensure that repository design criteria will accommodate HEU and Pu residues. • Dilution to LEU (power reactor equivalent).

Concern	Mitigation
8. Transportation costs and cask availability of some materials precludes many of the management options available.	• Establishment of an accurate SNF inventory will allow the private sector to design transport systems. • Discuss individual countries SNF disposition strategies through appropriate forums. • Continue to deal with public concerns over transportation.

7. FUTURE OPTIONS

(1) Regional repositories

For efficient, economical and safe storage of spent fuels and radioactive waste, regional or international repositories may be the preferred option, especially for the use of Member States with limited nuclear power programmes. Such repositories can only be realized if public relations issues are addressed as a priority. Discussions on this issue should be coupled with discussions related to environmental effects from other waste streams. For example, one country may offer to "trade" hazardous waste treatment and disposal services for radioactive waste disposal from another.

(2) Group like fuels

The potential for treatment of many fuels will be improved if like fuels are collected together for treatment. The possible swapping of materials between countries to achieve this end should not be ruled out. Again, this underscores the importance of an accurate inventory.

(3) Storage parameters

The identification of the parameters for best practice management and storage, both wet and dry, remains a priority issue.

8. CONCLUSIONS AND RECOMMENDATIONS

(1) All countries that have undertaken nuclear fuel development programmes have historic liabilities that cannot follow the standard treatment routes recognized for power reactor fuels.

(2) The choice of disposal, interim storage or chemical processing is dependent on a country's overall nuclear strategy. For countries with limited nuclear programmes there is a recognized difficulty with the funding of national facilities. The promotion of regional facilities for nuclear fuel and waste disposal is a potential solution, but the political problems associated with this should not be underestimated.

(3) To avoid further increases in the non-standard fuel inventory there should be a full consideration of the fuel back-end/waste management strategy before any new fuel development programme is permitted to continue.

(4) While some research reactor fuels can be safely stored for long periods in water basins, all research reactor operators should evaluate moving all fuels to dry storage if final disposition plans are uncertain. Assured dry storage has proven to be a safe and reliable

method for the long-term storage of fuels, already demonstrated for 25-30 years. Dry storage has also proved to be compatible with almost all research reactor fuels and is less costly than wet storage.

(5) Problems with dry storage have always been attributable to poor design. No international guidelines currently exist that define good practice for dry storage of experimental fuels.

(6) If new dry storage systems are developed, consideration should be given to integration of safeguards technology to lower overall cost.

(7) Each country should compile, complete and maintain a detailed inventory of all research and test reactor fuels. Ideally, this activity should be co-ordinated so that exchange of information can take place in electronic form in a standard database format, with a reasonable (4-year) time-scale for completion.

(8) In preparation of the research reactor fuel inventory, each country should perform a systematic evaluation of environmental, safety and health vulnerabilities associated with their current spent fuel storage and options for ultimate disposition. The fuel treatment option chosen should be planned from the information on the "current status" taking into account the following items as an aid to project prioritization:

- Safety — are there any substantial concerns?
- Vulnerability — identify and mitigate environmental, health and safety vulnerabilities.
- Availability of suitable technology.
- Funding — what other projects may compete for limited funding? If funding is earmarked for decommissioning, the option chosen must be planned in phase with decommissioning.
- Life cycle costs — should be considered along with annual costs.
- Consistency with policy — are any new waste forms created?
- Decommissioning — the treatment should not create further materials' problems for decommissioning.
- Non-proliferation — the treatment should take into account continued safeguards and include adequate physical security.

(9) The increasing cost of transportation and political pressure associated with nuclear material transport will limit the disposition options available to research reactor operators.

MANAGEMENT OF EXPERIMENTAL AND EXOTIC FUELS AT SCK/CEN, MOL, BELGIUM

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Abstract

SCK•CEN owns two experimental reactors BR2 and BR3 which produced a large amount of experimental and exotic fuels during the last 35 years. The spent fuel inventories for each reactor are briefly discussed and the present storage conditions identified. The present plans for the management of the various types of fuels are presented.

1. INTRODUCTION

SCK•CEN, Mol, Belgium. owns two reactors BR2 and BR3 which produced a large amount of experimental and exotic fuels during the last 35 years.

It is the purpose of this short presentation to discuss the fuel inventories, present conditions of storage and plans for possible conditioning and transfer of these fuels.

The information presented here reflects only an operator's point of view, looking for safe and controlled storage and for possible options for disposal.

2. BR3 PLANT

2.1. Plant description

BR3 was a Pressurised Water Reactor PWR, the first installed in Europe, with a capacity of 11 MWe (40.9 MW thermal). It was operated from 1962 to 1987.

The utilisation focused on the training of operators for the Belgian utilities and on the testing of new fuels on behalf of research centres and nuclear fuel suppliers.

The plant is now being decommissioned. After primary circuit decontamination and dismantling of the reactor internals, the decommissioning program foresees as next step the removal of the reactor pressure vessel.

2.2. Fuel description

Driver fuel and test fuels were similar to PWR fuels, made of UO_2 or UO_2/PuO_2 (MOX) with stainless steel and later Zircaloy cladding. The exotic character of the fuels stems from the large variety of enrichments used (up to 8.6 % ^{235}U and 10% fissile Pu, and the specifications of the fuel and cladding materials.

2.3. Interim storage on site

At the final shutdown in 1987, the fuel inventory in the storage pool amounted to 200 fuel assemblies containing about 2000 fuel pins (~ 2 t heavy metal).

Since this time, the fuel was maintained in the storage pool under demineralized water. No specific problems have arisen.

The decommissioning programme and pressure from the safety authorities led to the decision to remove BR3 fuel from the plant site.

2.4. Options for fuel disposal

Different options were considered :

- reprocessing,
- dry storage on site in "home-made" containers,
- dry storage in a TN24 container,
- dry storage on site following the NUHOMS concept,
- dry storage at Belgoprocess in Castor containers.

The reprocessing option was finally rejected due to the overall cost of the operation -transport, reprocessing, and subsequent waste management.

Dry storage on site with "home-made" containers was abandoned due to the expected licensing difficulties.

Dry storage in a TN24 was technically difficult.

The option selected was the GNS Castor container. The project foresees the purchase of 8 Castor 11ST containers for storage in a dedicated building on the Belgoprocess site. The fuel elements will be directly loaded into a basket placed in the container. Ruptured fuel pins (or segments - remnants - after post-irradiation examination) will be properly conditioned in sealed bottles. Interim storage at the Belgoprocess site is for 30 to 40 years. Final disposal - in clay - will only be possible after conditioning of the fuel elements in "standard bottles".

A formal decision in favour of this last option is to be taken in the near future.

3. THE MTR BR2 REACTOR

3.1. The BR2 facility

BR2 is an MTR reactor built in the late fifties. It started operation in 1963. After an extensive 21 months of refurbishment, it restarted in April 1997 for another projected 15 years of operation. The facility is equipped with a large storage channel and with hot cells.

3.2. Inventory of Experimental and Exotic fuels at BR2

For about 35 years the BR2 reactor was utilised in the framework of national and international irradiation programmes for various types of fuels representative of both thermal and fast reactors. This led to the storage at BR2 of a large number of irradiated exotic and experimental fuels.

Another source of stored fuels originates from post-irradiation - destructive and non destructive - examination contracts performed by the specialised Hot Laboratories (LHMA) at SCK•CEN.

After examination, all these fuels were transferred to the BR2 site for storage.

About 1200 fuel pins - intact and remnants from destructive post-irradiation examination - are now in wet storage at BR2, with ~ 420 pins from BR3, ~ 530 pins from thermal reactors and ~ 250 pins from fast reactors.

3.3. Storage conditions

All experimental and exotic fuels, intact and remnants, are stored under water in the storage channel of BR2. This storage channel is 50 m long, 3 m wide and 8 m deep. It is divided into compartments which can be isolated. One of these compartments is devoted to the storage of experimental and exotic fuels. The storage channel is filled with demineralized water which is continuously purified by a mixed ion-exchange bed with a flow rate of about 15 m^3/hr. It is foreseen in the near future to upgrade this purification circuit with an increased flow rate of 60 m^3/hr and include anion and cation beds, filters and UV lamps.

Fuel pins are stored in aluminium tubes which are located in storage racks (aluminium and stainless steel). Remnants from fuel pins were first loaded in a dry aluminium tube (first containment) which was inserted in another dry aluminium tube (second containment). This second containment is placed in a (wet) tube for positioning in the storage rack.

Recent examination of the storage conditions for remnants revealed significant corrosion of the first and second containment tubes: the second containment was found to contain heavily contaminated water, but only the first containment is supposed to be filled with water.

3.4. Future plans for storage and disposal

In the near future, intact BR3 fuel pins may be transferred back to BR3 for loading in Castor casks. Remnants from BR3 pins will be conditioned for loading in these casks.

Thermal reactor fuel pins and their remnants may also be conditioned before loading in casks.

No plans for transfer exist yet for the fast reactor fuel pins.

Conditioning of the remnants in storage at BR2 is now being studied. Conditioning will certainly involve the replacement of the second containment by stainless steel tubes sealed with metal gaskets. Provisions for conditioning shall take into account the required specifications for interim dry storage.

MANAGEMENT OF EXPERIMENTAL AND EXOTIC FUELS BY ATOMIC ENERGY OF CANADA Ltd

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Abstract

Atomic Energy of Canada (AECL) has been engaged in nuclear research and development at its Chalk River Laboratories since the 1940's. During this time, a wide variety of irradiated research reactor and experimental fuels have been stored in a variety of storage facilities. Some of these fuels are of unique composition and configuration, and some fuels have been degraded as a result of research activities. In preparing decommissioning plans for these storage facilities, AECL has developed a strategy that identifies how each type of fuel will be dispositioned in the future. The goal of this strategy is to ensure that the fuels are maintained in a safe stable state until a repository for these fuels becomes available. This paper describes the current storage facilities, options considered for long-term fuel management, and the strategy selected to manage these fuels.

1. INTRODUCTION

Atomic Energy of Canada (AECL) has been generating spent research reactor fuel since the late 1940's. These fuels have originated from the research and isotope production activities performed by research reactors located at the Chalk River Laboratories (CRL) in Ontario, and the Whiteshell Laboratories (WL) in Manitoba. The missions of these reactors have evolved over time, from the production of plutonium in the late 1940's and early 1950's, to the current role in developing CANDU© reactor fuels and the production of medical isotopes. A wide variety of fuel types have been accumulated over time, and are currently being stored in a variety of storage facilities. AECL has initiated a program to evaluate the current storage of these fuels, and recommend and implement a strategy to bring the inventory of fuels into a stable state for the long term.

The purpose of this paper is to describe AECL's program to address the long term storage and disposal issues associated with its research reactor and experimental fuels. The paper will outline the program steps that were taken to assess current storage conditions, evaluate storage and disposal options, and develop the strategy for long-term spent fuel management.

2.0. CURRENT STORAGE OF RESEARCH REACTOR FUELS

Spent research reactor fuels have been accumulated from operating the National Research Experimental Reactor (NRX) and National Research Universal (NRU) Reactors at CRL in Ontario, and the organically cooled Whiteshell Research Reactor (WR-1) in Manitoba. AECL's inventory is diverse, and many fuel types, enrichments and configurations are represented. In all, there are approximately 75 fuel types and configurations currently in storage. As improvements in fuel performance were made and the missions of the reactors evolved, several types of driver fuels were developed and used in the research reactors, and a variety of fuels originated from research reactor loop experiments. The AECL inventory is summarised by major fuel types in Table I.

Table I: Spent Fuel Inventory

Fuel Type	*Rod, Bundles or Elements*	*kg HM*	*Comments*
Metal Fuels	193	9.537	Early Al clad NRX driver fuels
U Oxide (long)	433	7.829	Al clad driver & FN rods
U/Al dispersions	966	525	Al clad NRX & NRU driver fuels
U Silicides	1,200[1]	3,000[1]	NRU driver fuels
U Oxides (short)	1,788	6,428	CANDU development & PIE debris
U Oxides (bundle)	650	7,500	WR-1 driver fuels
U Carbide	1,060	11,000	WR-1 driver fuels
Other	543	3,813	MOX, special alloys & PIE debris
Total	***6,893***	***49,632***	

The metal fuels represent some of the oldest fuels in the inventory, and consist of natural uranium metal and thorium metal fuel rods. The uranium metal fuels were used as the original driver rods in both the NRX and NRU research reactors, and were originally re-processed for plutonium production. The thorium metal fuels were used in the research in thorium fuel cycles. Both fuel types are typically solid cylindrical rods clad in aluminium, and are approximately 3.35 m in length. Some of the uranium metal fuels were fabricated as 5 flat fuel elements encapsulated in an aluminium flow tube, also approximately 3.35 m in length. Metal fuels are vulnerable to corrosion. Uranium metal can form pyrophoric uranium hydride when it corrodes in an oxygen deficient and hydrogen rich atmosphere. The uranium aluminium fuels are also NRX & NRU driver fuels of various enrichments. They are less vulnerable to corrosion than the uranium metal fuels.

The uranium oxide "long" rods were primarily NRX driver rods. These fuels were either solid or annular cylindrical pellets clad in aluminium and are also approximately 3.35 m long. Some of these oxide rods were also used to accommodate experimental irradiations. The "short" uranium oxide fuels are experimental fuels used in the development of the CANDU© reactor fuels, generally consisting of bundles of zirconium clad elements 50 cm long. These fuels are typically stored in small sealed fuel cans, and are often partially disassembled, cut or altered as a result of post irradiation examinations. Further, there are zirconium clad uranium oxide bundles from the WR-1 reactor. These bundles are intact.

The uranium silicide fuels (U_3Si/Al) are the current NRU driver fuels, used in the production of medical radioisotopes and neutrons for condensed matter and material research. The carbide fuels were used in the WR-1 reactor, and are slightly enriched uranium carbide, Zr/Nb clad bundles, approximately 0.5 m in length.

Finally, there is a relatively large group of miscellaneous fuels. For example, these include:

- non-standard oxides of various enrichments, cladding materials (e.g. stainless steel), and unique configurations;
- uranium graphite rods;
- non-standard alloys such as uranium zirconium;
- thorium and uranium mixed oxides.

These fuels are stored in dry storage facilities after initial cooling in rod bays. These facilities are tile holes and concrete canisters. The canisters are above-ground concrete cylinders that are marketed by AECL world-wide. The WR-1 oxide and carbide driver fuel bundles are stored in concrete canisters.

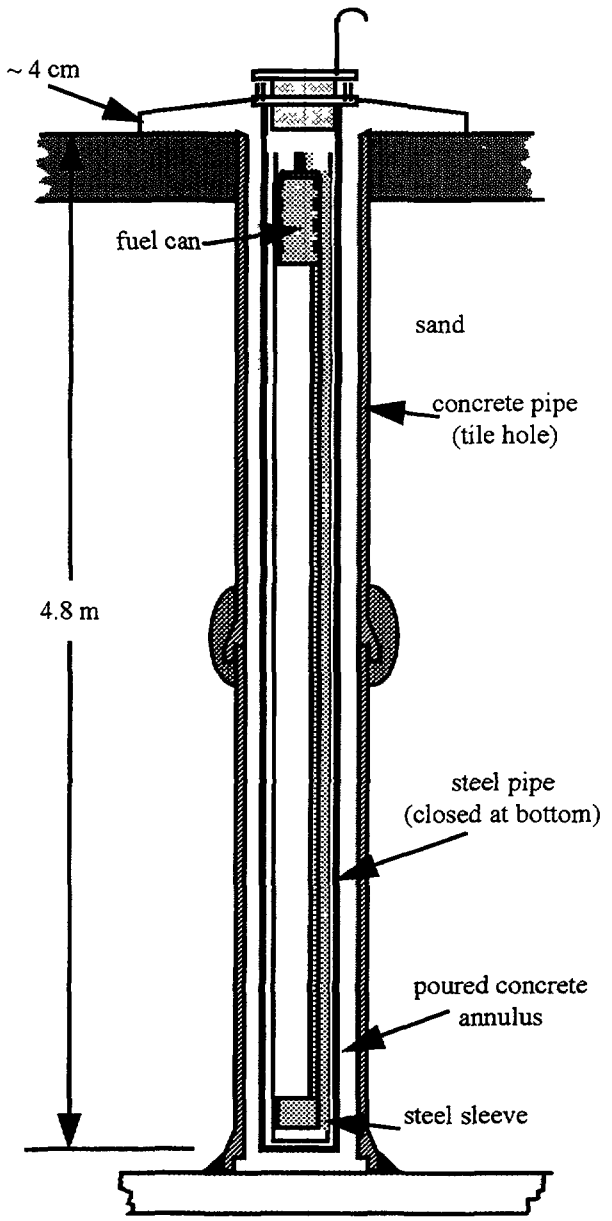

Figure 1.: Long Fuel Can in "IFE" Type Tile Holes

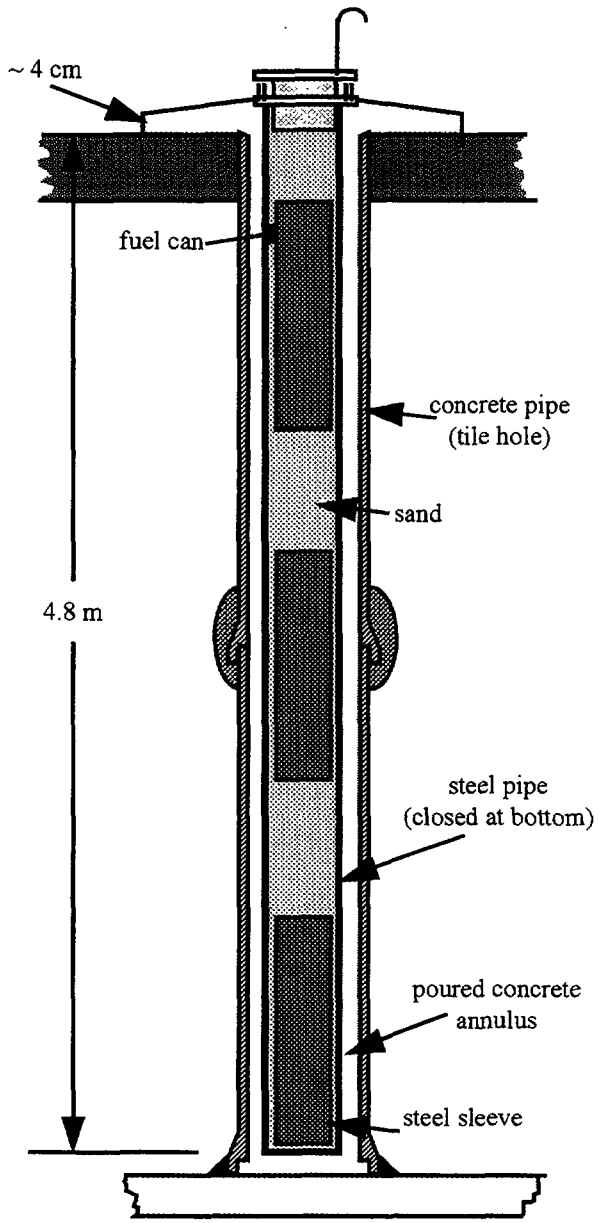

Figure 2.: Short Fuel Can in "IMD" Type Tile Holes

The CRL fuels are stored in tile holes. These are below grade storage cylinders, comprised of a steel liner enveloped within a concrete cylinder. Intact research reactor driver fuel elements are held in long mild steel fuel cans, as shown in Figure 1, and emplaced in "irradiated fuel element type" (IFE) tile holes. Defected fuel rods are kept in long closed fuel cans, also in IFE-type tile holes. The experimental fuel bundles and elements are stored in short closed mild steel fuel cans, as shown in Figure 2, and emplaced in "irradiated debris type" (IMD) tile holes, some of which have been back-filled with sand.

The tile holes have undergone many changes since they were first implemented in the early 1960's. The more recent versions of the tile hole design are effective in storing current fuel inventories. However, early vintage IFE-type tile holes have been found to be less effective at storing some fuel inventories, most notably uranium metal fuels, compromised or

disrupted experimental fuels and mixed inventories of non-standard oxides and enriched fuels. Further, there is a concern that in the older IMD-type tile holes, unabated mild steel corrosion of the fuel cans could lead to problems in retrieving fuels in the future.

3.0. PROGRAM APPROACH

The Tile Hole Remediation Program was established in 1995 with a mandate to prepare and implement cost effective decommissioning plans for the AECL fuel storage facilities. The following underlying principles were used in developing the decommissioning plans:

- direct resources to activities that deal with areas where there is the greatest potential for near-term health and safety impact, and/or that will significantly minimise business risks;
- ensure facilities are in a safe and stable state for the long term; and
- implement final program elements when demonstrated technologies and experience exists.

In developing the decommissioning plans, the program preferentially considered proven technologies over unproved or novel approaches, and used external expertise whenever appropriate. The approach taken to develop a spent research reactor fuel management strategy can be summarised in three steps:

1. identify and quantify hazards associated with various fuel types;
2. perform a systematic evaluation of options for dispositioning various fuel types; and
3. prepare decommissioning plans and cost estimates based on recommended dispositioning option(s).

A very significant effort has been made in preparing a comprehensive inventory of fuel types and geometries. Confidence in the fuel inventories of fuels in early vintage tile holes -- where the documentation may not be as complete as is currently provided for -- is extremely important, since most of the more exotic and failed fuels originate from early experimental programs and have been stored for the longest period of time. An assessment of each fuel type was conducted to determine the potential hazards of each fuel type under various storage conditions. Tactical sampling coupled with ongoing monitoring and surveillance of the tile hole storage facilities was performed to establish actual storage conditions. The actual hazards to be expected under long term storage conditions were then determined using conservative, but illustrative parameters.

A team was assembled to identify all credible pathways for moving the fuel from current storage to final disposition (i.e. the point where the fuels do not represent an ongoing liability). Final disposition was determined to be either disposal of the fuel or transfer of ownership to another responsible authority (e.g. the United States under the terms of the nuclear non-proliferation policy). Alternative interim storage conditions for each fuel type were also considered, such as continued storage in tile holes or repackaging and storage in concrete canisters, and evaluated against reprocessing the fuels into a vitrified waste form based on the hazard assessment discussed above. The principle options considered were:

- maintain current storage in tile holes until a repository becomes available;
- modify existing tile holes and maintain storage;

- condition and repackage the fuels and move to concrete canisters,
- process fuels into a stable waste form, and
- return fuel to U S under the nuclear non-proliferation policy

From this, a framework to evaluate all options was developed, simply stated, a decision tree Costs for the various alternative strategies in the decision tree were provided from conceptual engineering studies performed by internal and/or external experts Treatment, storage and disposal costs and cash-flow requirements to disposition each fuel type were then determined An analysis of options and a recommendation was forwarded to management, that will form the basis of decommissioning plans for the tile hole facilities

4 0 EVALUATION RESULTS & STRATEGY

Conceptual studies examining research reactor fuel final disposition options concluded that it is technically feasible to dispose of all AECL fuel types directly into a deep geological repository However, there are significant issues to be resolved before direct disposal can be achieved, such as the costs and regulatory constraints required to dispose enriched fuels The study has assumed a reference date of 2050 for when a repository would become available It was concluded that most fuels in the inventory could safely be stored in the current tile hole facilities until this date with an appropriate level of monitoring and surveillance. New storage facilities would have to be provided if a repository able to accept the research reactor fuels will not become available by 2050

The studies have also concluded that some fuels stored in early vintage tile holes would require future treatment and repackaging in hot cells, as a prerequisite to long-term dry storage until a repository becomes available Processing fuels into a more stable vitrified waste-form prior to storage was considered economically unattractive, in particular the high costs associated with transport to existing facilities in Europe Further, conditioning or packaging of the fuels in anticipation of unspecified future repository requirements was rejected, also for economic reasons

Return of enriched fuel to the United States is still being considered, relative to the following issues·

- not all of the enriched fuels in the current inventory or future arisings are eligible for return to the U S., therefore, the issues relating to the direct disposal of enriched fuels will still have to be addressed, and
- the large expenditures required to return the fuels to the U S would preclude other important decommissioning work planned by AECL

It is planned to begin the retrieval, repackaging and transfer of fuels from early tile holes into concrete canisters early in the next millennium If by 2015 the probability that a repository will be available by 2050 is low, then the decision to move the remaining fuel inventory to long-term storage in canisters will be evaluated

5 0 CONCLUSIONS

AECL has invested considerable effort in developing a sound strategy to manage its spent research and experimental fuels This effort has been directed at the smaller proportion of fuels that represents the greatest potential hazard, and at ensuring these fuels are

maintained in a safe stable state for storage until final disposal can be achieved. The strategy was developed using a combination of internal and external resources with demonstrated expertise in fuel management and experience with proven technologies.

In using proven technologies and existing expertise, the implementation of costly developmental programs and risk is avoided. Using external contractors and vendors is also useful in developing a strategy, as they often have more experience in many of these technologies, can often bring proven technologies from other industries, and allows the facility owner to focus on developing a sound and cost effective strategy rather than becoming preoccupied with technical issues. Finally, using external expertise develops a good understanding of contractor capabilities that will be valuable in evaluating proposals for decommissioning work during project implementation.

ACKNOWLEDGEMENTS

I would like to thank K. A. Simpson, A. T. Jeffs and L. Johnson for their ongoing contributions to the program. I would also like to thank D. R. Champ for his review of this paper.

STORAGE AND MANAGEMENT OF SPENT FUELS AT RESEARCH REACTORS AT THE CHINA INSTITUTE OF ATOMIC ENERGY

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Abstract

Research reactors at the China Institute of Atomic Energy and their interim spent fuel storage are described. Mitigation measures for the limited storage space which must accommodate continuously increasing amounts of spent fuel are also described. The overall status of research reactor spent fuel in China is outlined.

1. INTRODUCTION

China Institute of Atomic Energy (CIAE) owns several research reactors and critical facilities, some of which are the oldest in China. A certain amount of spent fuel has been accumulated from their operation over more than thirty years. Table 1 lists the main reactors at CIAE. They have been operating for several months every year and have therefore been accumulating spent fuel continuously except for the Miniature Neutron Source Reactor MNSR which has not had its core reloaded yet due to its very low power level. The Heavy Water Research Reactor (HWRR) went critical in 1958 with a core of 2% enriched metallic uranium fuel, cooled and moderated by D_2O at a power of 10 MW. The reactor was reconstructed between 1978 and 1982. The current modified core uses 3% enriched UO_2 and has a rated power of 15MW (see HWRR-III in Table 1). Other related information is also given in Table 1.

2. STORAGE OF SPENT FUEL

Table 2 gives information related to the storage of spent fuel from CIAE reactors. Initial constructions of HWRR and SPR interim storage facilities were designed to meet all requirements foreseen at that time. However, the situation has changed after long term operation and the spent fuel storage space provided for these two reactors has proved to be insufficient. The following measures were taken to solve the problem.

a) Maintain and monitor water chemistry conditions to ensure that the stored spent fuel keeps its cladding integrity for the whole of the interim storage period. The essential parameters controlled include pH and conductivity of the water in the storage pool.

b) Re-racking of the spent fuel. Two methods have been used. The first is to make more dense store simply by creating additional positions among the regular storage cells. The second was to create a two-tier store. Fig 1 shows a storage pattern for SPR where the storage racks in the pool are composed of a top layer and a bottom layer thus doubling the storage capacity.

Table 1

Name of Reactor	HWRR		SPR	MNSR
	I	III		
Date of Critical (year)	1958	1983	1964	1984
Reactor Type	D_2O	D_2O	H_2O Pool	Tank in Pool
Power (kW)	10,000	15,000	3500	27
Fuel Enrichment (U-235%)	2	3	10	90
Fuel Material	Metallic	UO_2	85%UO_2-15%Mg	UAl_4-Al
Height of Core (mm)	1,243	1,000	500	230
Diameter of Uranium (mm)	Tubular	6.0	7.0	4.3
Clad Material, Thickness (mm)	Al, 1.0	Zr, 1.0	Al, 1.5	Al, 0.6
U-235 Mass in a Rod (g)	80	9	8	2.88
Number of Rods in Assembly	1	12	16	1
Number of Assemblies in Core	84	72	43	350
U-235 Mass in Assembly (g)	80	108	128	2.88

c) transfer part of the spent fuel to an interim storage facility to ensure that the on-site storage always has enough spare room to accommodate a complete core load of fuel. This satisfies maintenance requirements as stipulated in the operational procedure (i.e. space for an emergency core unload).

Figure. 2 shows the layout of the HWRR storage pool while Figures 3 to 5 give the storage pattern of HWRR as of October, 1996. As indicated in Table 2, 180 spent assemblies had already been moved to a dry interim storage facility at the site of a pilot reprocessing plant. Therefore, more than 100 rack positions on site are available to receive new spent fuel.

Table 2 Spent Fuel Storage as of October 1996

Name of Reactor	HWRR(III)	SPR
Size of Storage Pool (m)	12×5.5	4×2.5
Depth of Pool Water (m)	5.9	5.8
Depth of Shielding Water (m)	3.5	3.8
Dose Rate above the Pool (μR/s)	<10	<10
Number of Storage Racks	293	280
Number of Racks Occupied	115	200
Rate of Spent Assemblies Produced per year	20	6.3
Number of Spent Assemblies Transferred	180	0

3. THE PILOT REPROCESSING PLANT

To finally resolve spent fuel disposition, reprocessing has been selected as the main solution for most of the research reactor spent fuel in China. A pilot reprocessing plant is being designed with start up planned for the year 2000. The design capacity for reprocessing is currently specified as follows:

for spent fuel from power reactors 300 kg UO_2/day

for spent fuel with higher enrichments 0.8 kg ^{235}U/day

4. SPENT FUEL FROM THE OTHER RESEARCH REACTORS IN CHINA

Though the above description is specifically for CIAE, it can be used to get a general picture about the spent fuel issues for the whole country.

The operating research and test reactors with significant power to produce spent fuel are the following:

- Reactors of SPR-type, including SPR (CIAE), SPRR-300 and the Tsinghua Reactor
- HWRR
- Material test reactors HFETR and MJTR
- Nuclear Heating Reactor
- Xi'an Pulsed Reactor
- Miniature Neutron Source Reactor (there are four in China).

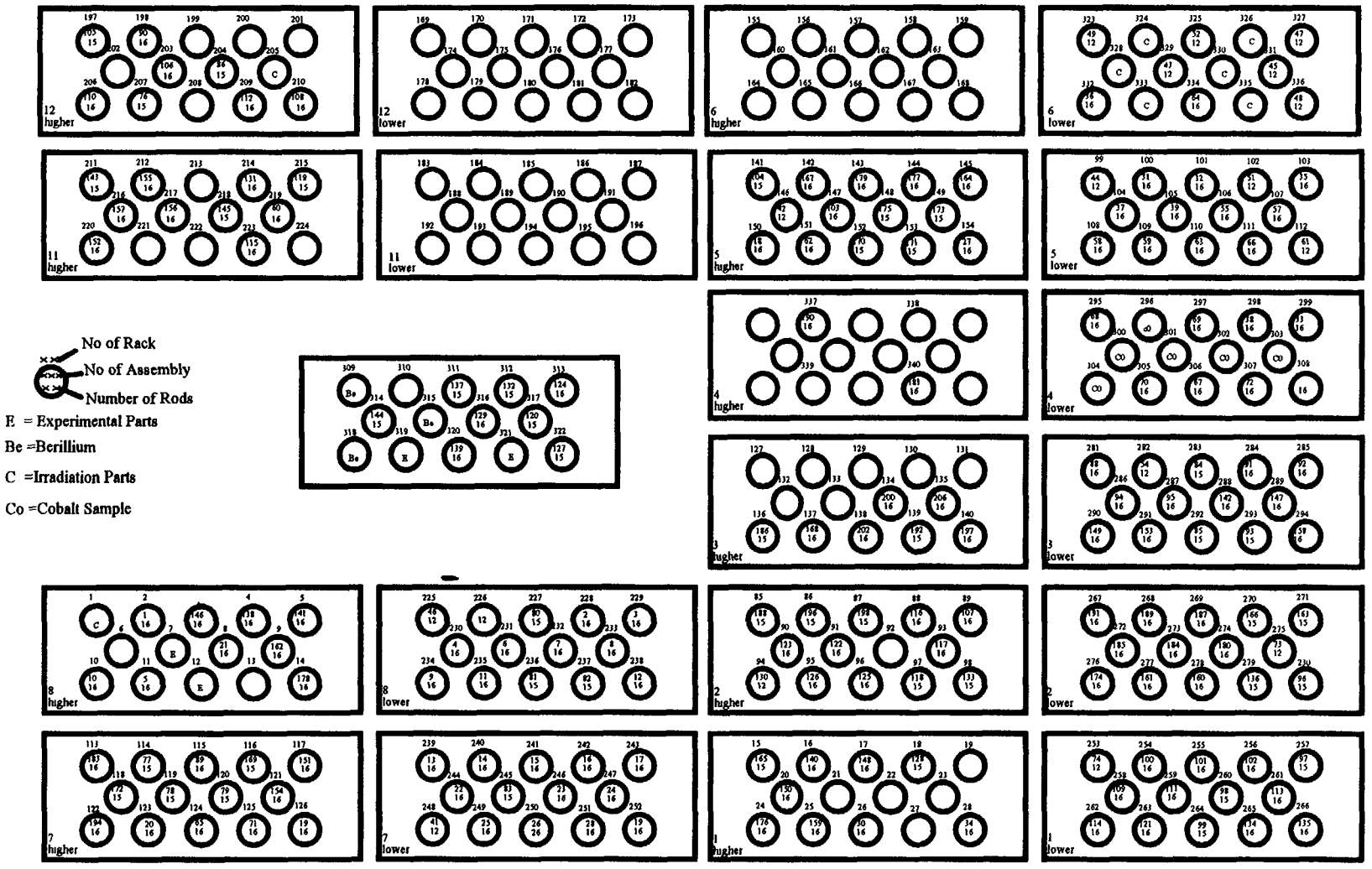

Fig. 1 Store Pattern of SPR, May of 1996

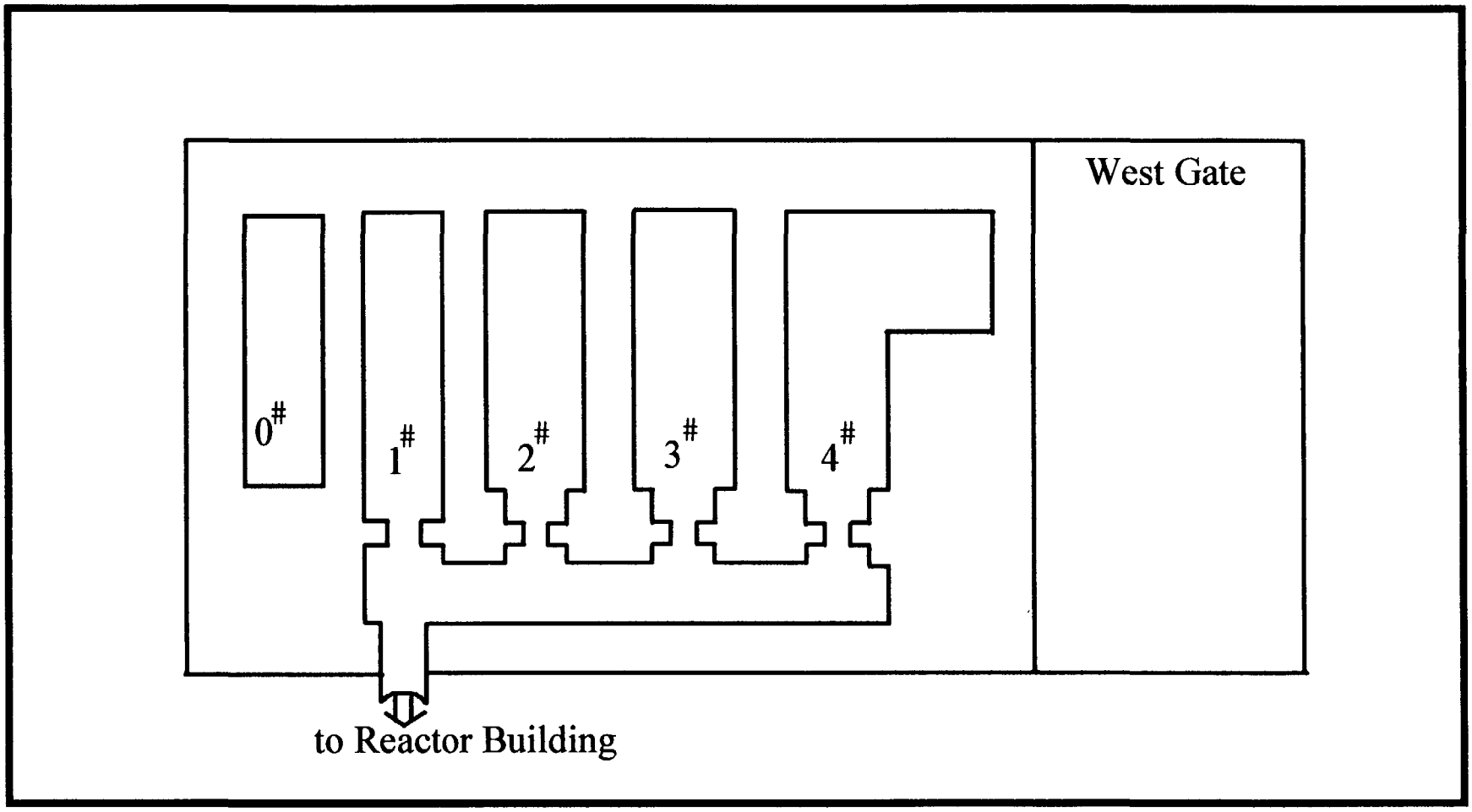

Fig. 2 Storage Pool of HWRR

Pool No. 1 (Total 22)

6	6	5	4	# 4	# 5	# 6	10
	3	* 2	1	# 1	# 2	# 2	
5	A073 860630 6	A111 860630 5	A122 860630 4	A142 860630 4	A002 850424 5	3059 900913 6	9
	A262 881221 3	A236 900913 2	A259 900913 1	A114 860630 1	A123 860630 2	A119 860630 3	
4	A059 850424 6	A196 860929 5	A174 871030 4	A268 900913 4	3174 900913 5	3394 900913 6	8
	A169 871031 3	A042 841211 2	A017 841211 1	1	2	A178 861030 3	
3	6	5	4	4	5	6	7
	3	2	1	1	2	3	
2	6	5	4				
	3	2	1				
1	6	5	4				
	3	2	1				
si 1							
si 2							

Note:
1. # Other Core Component.
2. * J-02 Irradiation Rig.

Fig. 3 Store Pattern of HWRR, Pool No.1, Oct. of 1996

Pool No.2 (Total 97)

7	A265 9301058	3272 9301057	3163 9301056	3157 9301055		A150 8602205	A152 8602206	A154 8602207	A153 8602208	13
	3371 9301054	3166 9301053	3008 9301052	3129 9301051		A147 8602204	A158 8602203	A159 8602202	A149 8602201	
6	3200 9301058	A264 9301057	2079 9301056	3329 9301055		A112 8606305	A113 8606306	A131 8606307	A163 8606308	12
	3022 9112204	3365 9112203	A271 9112202	3081 9112201		A166 8606301	A167 8606302	A128 8606303	A049 8412114	
5	3158 9112208	3033 9112207	3271 9112206	3055 9112205		A171 8606305	A172 8606306	A162 8606307	A211 8610158	11
	3268 9112204	3181 9112203	3255 9112202	3171 9112201		A148 8602201	A168 8606302	A133 8606303	A134 8606304	
4	3250 9112208	A269 9112207	A085 9112206	3027 9112205		A156 8602205	A157 8602206	A151 8602207	A146 8602208	10
	A137 8606304	A088 8507273	3164 9009132	A135 8606301		A132 8602201	A126 8602202	A130 8602203	A155 8602204	
3	A082 8507278	A008 8811307	A069 8504246	A071 8504245		A098 8510305	A078 8510306	A108 8510307	A090 8510308	9
	A056 8504244	A072 8504243	A070 8504242	A068 8504241		A086 8510301	A110 8510302	A105 8510303	A087 8510304	
2	A083 8507277	A097 8507276	A102 8507275	A084 8507274		A014 8504244	A075 8507275	A106 8510306	A077 8510307	8
	A109 8510303	A080 8510302	A076 8510301				A255 8911111	A254 8704202	A092 8507273	
1	A261 9009133	A103 9009132	A235 9009131							

Fig. 4 Store Pattern of HWRR, Pool No.2, Oct. of 1996

Pool No.3 (Total 59)

7	3155 950109 8	7	3011 950109 6	2020 950109 5		3378 950109 5	3152 950109 6	3140 950109 7	3132 950109 8	13
	3176 950109 4	3002 950109 3	2013 950109 2	2047 950109 1		005 950111 1	3340 950805 2	3072 951009 3	3377 951009 4	
6	3287 950109 8	3089 950109 7	3234 950109 6	2017 950109 5		5	6	7	8	12
	3006 950109 4	2007 950109 3	3379 950109 2	2092 940621 1		1	2	3	4	
5	2088 940621 8	2063 940621 7	2072 940621 6	3353 940621 5		5	6	7	8	11
	3131 940621 4	3153 940621 3	2093 940621 2	182 940621 1		1	2	3	4	
4	3010 940621 8	2068 940621 7	2067 940621 6	2077 940621 5		5	6	7	8	10
	2044 940621 4	3139 940621 3	2046 940621 2	3099 931026 1		1	2	3	4	
3	2097 931026 8	006 931026 7	2055 931026 6	2085 931026 5		5	6	7	8	9
	2084 931026 4	2082 931026 3	2010 931026 2	2053 931026 1		1	2	3	4	
2	3069 931026 8	2089 931026 7	2016 931026 6	3070 930105 5		5	6	7	8	8
	3156 930105 4	3098 930105 3	3003 930105 2	A270 930105 1		1	2	3	4	
1	3080 930105 8	3097 930105 7	3070 930105 6	3147 930105 5						

Fig. 5 Store Pattern of HWRR, Pool No.3, Oct. of 1996

Many of the reactors mentioned above are operated at a very low power or used on a campaign basis. The test reactor MJTR actually uses fuel discharged from HFETR. Therefore, the only contributor to spent fuel in addition to the SPR and HWRR reactors is HFETR, which has already produced several hundred spent fuel assemblies.

PROCEDURES AND TECHNIQUES FOR THE MANAGEMENT OF EXPERIMENTAL AND EXOTIC FUEL FROM RESEARCH AND TEST REACTORS IN FRANCE

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Abstract

Since 1995, the Commissariat a l'Energie Atomique CEA has implemented a three point fuel management plan. One of the three points, the removal and the reprocessing of a wide range of spent fuels stored in different facilities in Saclay, Grenoble and Cadarache nuclear research centers, is described. The COGEMA Group has developed and implemented a comprehensive set of solutions for the management of research reactor and exotic fuels. It includes transport casks, storage casks or interim storage facilities at the reactor site, or at a centralized interim storage, as well as spent fuel reprocessing, material recycling and waste conditioning.

1. CEA SPENT FUEL

1.1. Reprocessing and storage management

As a result of its programs, the French Commissariat a l'Energie Atomique (CEA) has produced irradiated fuels. These fuels come from either abandoned reactor types or laboratory experiments (fuel examinations, critical experiments, special fuel element designs) or the experimental reactors operated in Cadarache, Saclay and Grenoble. These fuels were stored in appropriate facilities: water pools in Cadarache and Grenoble or dry pits in Saclay and Cadarache (i.e., the Cascad facility).

Usually, the CEA chooses the reprocessing solution, because it is considered to be safer as far as waste is concerned. The fissile materials contained in these fuels follow a closed cycle: fabrication, irradiation, disassembly, reprocessing, re-fabrication and re-use). Management of this cycle implies the use of storage facilities for a period of just a few years while the short-lived radioactive isotopes decay.

However, this cycle has been stopped for most of the experimental reactors, which has resulted in an expanding inventory of spent fuels. The above-mentioned facilities could be filled to capacity as an obvious consequence of this situation.

Moreover ,these facilities are also used for the storage of damaged spent fuels. The weight (90 metric tons) and variety of these spent fuels make it necessary for the CEA to invest heavily in their safe packaging and reprocessing. There are six main types of such fuels (Table I).

Management of these spent fuels must take account of two main objectives:

- Safe removal and safe packaging of every fuel type for suitable reprocessing or long term storage.
- Organization of appropriate management and facilities in order to avoid conditions similar to the above-mentioned conditions in the near future.

TABLE I : SPENT FUEL COMPOSITION

Fuel types	Weight (tonnes)
Natural Uranium Graphite Gas	20
Pressurized Water Reactors	4
Heavy Water Reactors	50
Fast Breeder Reactor	6,9
"CARAMELS" OSIRIS reactor	6,5
Material Testing Reactor	15
Laboratory samples	1,3

To achieve these objectives, the CEA has developed and is implementing a three point plan:

- Immediate reprocessing in existing facilities at Marcoule (APM, UP1).
- Postponed reprocessing, planned for the early twenty-first century, involving regrouping of fuel elements in appropriate storage facilities at Cadarache.
- Long-term storage (< 50 years) in safe conditions, at the Cascad facility. This alternative implies that the CEA will have to choose after the year 2000 between reprocessing or direct disposal.

Reprocessing was possible at APM and UP1 in Marcoule until October 1997. Therefore, the CEA has decided to send as much fuel there as possible. Only heavy water spent fuel coming from the EL4 reactor and irradiated and non irradiated laboratory samples are stored at the Cascad and Pegase facilities in Cadarache.

2. SOLUTIONS FOR THE DIFFERENT TYPES OF FUEL

The solution consists of the following steps:

- fuel removal from the storage facilities;
- checking the fuel condition and that of its packaging;
- eventual handling and re-packaging in a new canister;
- transfer to Marcoule;
- reprocessing at APM or UP1
- and finally, managing the waste and recovering the fissile materials resulting from reprocessing.

Table II : SPENT FUEL DIAGRAM

SPENT FUEL TYPE	PRIOR TREATMENT AND PACKAGING	APM/TOR	COGEMA/ MAR400	COGEMA/ UP1
PWR		Dissolution		reprocessing
MTR			diassembly	reprocessing
CARAMELS	ISAI	Dissolution		reprocessing
UNGG/SACLAY	PRECIS		Underwater cutting	reprocessing
UNGG/MARCOULE	BSC		Underwater cutting	reprocessing
UNGG/CADARACHE	INB56 and STAR			reprocessing

The diagram (Table II) shows all of the operations for each type of spent fuel. Study of this diagram leads to two families of spent fuels:

- Fuels that can be sent for reprocessing to Marcoule without prior handling and repackaging.
- Fuels requiring handling and new packaging prior to their transfer to Marcoule: "CARAMEL" and UNGG (magnox-type) spent fuels.

3. REPROCESSING OF PWR, FBR and MTR SPENT FUELS

3.1. PWR

Spent fuels used in the development of PWR were initially stored at the Saclay, Grenoble and Cadarache facilities. They were later moved and packaged for reprocessing at the Marcoule TOR-UP1 facility. This plant has been used for separating valuable nuclear materials (uranium and plutonium) from fission products since December 1994. All of these fuels were reprocessed in 1996.

3.2. Rapsodie-FBR

Between November 1994 and March 1995 CEA transferred these fuels (blanket and core) from the PEGASE facility to the APM pilot work-shop at Marcoule. All of the fuels from RAPSODIE were reprocessed at TOR-UP1 in 1995.

3.3. MTR

MTR spent fuels, with HEU of US origin are used in research reactors such as Orphée in Saclay and Siloé in Grenoble. They are stored for decay of short-lived isotopes in pools located near the reactor and then transferred for storage at Cadarache (PEGASE). Since August 1996 most of them have been transferred to a pool at the COGEMA MAR400 facility. These assemblies are then cut underwater in order to remove their heads and bases of pure

aluminum. They were reprocessed at the COGEMA-UP1 plant in 1998. Reprocessing of the remaining quantities is now possible at the La Hague reprocessing plant operated by COGEMA.

4. REPROCESSING OF THE FUELS FROM THE OSIRIS RESEARCH REACTOR AND UNGG REACTOR TYPES

4.1. Osiris "caramel type"

Fuels of the Caramel-type stored in the Saclay and Cadarache pools were transferred to the Marcoule facility for reprocessing at the TOR-UP1 plant before its final shut-down at the end of 1997. Before reprocessing at TOR-UP1 they underwent mechanical conditioning at the ISAI laboratory (Irradiated Assemblies Monitoring Installation) also located in Marcoule.

4.2. UNGG Marcoule type and EDF NPP type

There are two alternatives for the storage of these fuels. Either they are stored in pools at Cadarache or in dry shafts in concrete blocks in Saclay and Marcoule. Each fuel storage required the design and construction of a specific facility in order to recover, handle and package them before reprocessing. These facilities, described below are PRECIS at Saclay, BSC at Marcoule, and INB 56 and STAR at Cadarache.

5. SPECIFIC FACILITIES

5.1. ISAI

The ISAI laboratory in Marcoule has two hot-cells. A "Caramel-type" fuel element consists of welded plates, which must be separated without breaking the first confinement barrier formed by the fuel cladding. In order to avoid any fire risk, this operation is performed under an inert atmosphere. Then, the plates are transferred into the second cell where the operators carry out the punching and cutting of each plate. The pieces, recovered through a gravity feed in aluminum canisters are sent for reprocessing to TOR-UP1.

5.2. Packaging of stored fuels at PRECIS in Saclay

In 1995, the CEA decided to remove these spent fuels stored up until then in dry shafts in two concrete blocks. Among the 720 containers stored in these blocks, 592 contained UNGG and heavy water fuel types which could be reprocessed at Marcoule UP1.

These fuels used thirty years ago for some experiments at the Saclay laboratories were transferred in containers, in pieces, cartridges and Uranium rods, most of them with their cladding. The PRECIS facility was necessary for their recovery under safe conditions involving:

- extraction from the shafts;
- transfer in containers through a confinement bag, to an airtight shielded cell filled with inert gas;
- identification by X-ray radiography of the cartridges;
- conditioning in canisters for transport in casks.

After they had been cut under water at MAR 400, they were sent to the UP1 plant for reprocessing by COGEMA before the shut-down of UP1 at the end of 1997.

5.3. Packaging of stored fuels at BSC in Marcoule

In 1996, the CEA decided to recover the 814 spent fuel assemblies stored in concrete blocks in Marcoule. These fuels had been kept for eventual specific examinations by the CEA Their removal required building a facility identical to that of PRECIS. These 814 spent elements were reprocessed at UP1 after transfer to MAR400 for underwater cutting.

5.4. Recovery and packaging of the fuels stored at INB 56 in Cadarache

About 2,000 spent fuel elements, packaged in steel canisters, were stored in these two pools over 25 years ago. These containers were placed in baskets and then stacked in levels. Some of them were no longer watertight. Once the pool water is filtered and decontaminated, visibility is good enough to perform the recovery operations safely with accurate equipment. The containers are pulled out one by one with a hook and transferred into a rotation device necessary to ensure gas removal (H_2) with maximum safety. Then they are transferred to the immersed radioscopy station in order to assess the state of the fuel and eventually detect water within the canister. After this test, the canisters are placed in a transfer container under inert gas, then they are put into a neoprene bag to prevent external contamination, and finally they are packaged for transport.

All the operations are remote-controlled in order to minimize risks, and monitored on video-cameras (some of them immersed). Images are recorded and monitored by the operators in a control room located in the building.

5.5. The STAR facility in Cadarache

The STAR facility is a high activity laboratory built specifically for the treatment, cleaning and conditioning of UNGG spent fuel elements. The purpose of the STAR facility and its associated processes is therefore to separate the nuclear fuel from the cladding, to chemically stabilize the nuclear material and to condition it in sealed canisters meeting road transport regulations and reprocessing specifications at UP1 in Marcoule.

Know-how of laboratories already in operation was used for the design of the STAR facility in order to comply with the latest design safety rules and to allow further developments and R&D on various types of fuel. The main components are described below:

- 3 independent cells, with leak-tight transfer lock chambers;
- 2 partitioned rear cells for fuel element admission;
- a shielded upper cell, covering the 3 main ones, for equipment access and maintenance;
- numerous access hatches, for the reception of various vertical or horizontal casks;
- cell walls lined with stainless steel, in order to make later decontamination work easier;
- total removability of every device, through the upper cell.

The spent fuel reconditioning process was developed by the CEA and validated by LECA. It includes the following operations performed in hot cells by remote handling:

- Removal of the canisters from the storage pool, -pre-identification and control.
- Transportation in shielded casks to the STAR facility.
- Reception of the casks and canister transfer to the processing cell.
- Identification and opening of the canister and identification of content.
- Mechanical removal of magnesium cladding.
- Separation of nuclear materials from Mg cladding waste and conditioning of this waste for specific disposal.
- Nuclear materials are then placed into a specially designed oven to go through following steps in order to withstand the effects of a hydrogen explosion.
- Drying at 100°C in argon cover gas.
- Hydride dissolution in argon cover gas at atmospheric pressure.
- Partial oxidation of highly flammable metallic uranium particles. This operation is carried out by O_2 injection, and under varying pressure, until a stable product is obtained.
- Conditioning of the stabilized materials in Aluminum alloy (AG3) canisters. These are fitted with soluble Magnesium windows and welded leak-tight.
- AG3 canisters, as well as their magnesium windows, are designed to meet the UP1 reprocessing requirements.
- Transfer of the plugged canisters to cell 2 for weighing, optional decontamination and air-tightness control.
- Transfer to cell 3 for interim storage and transportation by shielded casks (8 to 12 canisters in baskets) to the UP1 reprocessing plant.

In order to confine contamination as far as possible, all operations from fuel reception to canister sealing are performed in a single large cell (cell 1).

5.5.1. Operation assessment

By the end of 1996, about 500 UNGG cartridges had been packaged in STAR, out of which 300 have been reprocessed at the COGEMA UP1 plant. After a year of operation, the following points can be emphasized:

- about 90% of the cartridges do not match the expected features. The fuels have been heavily damaged during the 25 years spent under water;
- very few of the cartridges contain free water (5% vs. an expected 40 %) and cementation of this water is satisfactory;
- fuel cladding removal is easy;
- damaged fuel is pyrophoric;
- thermal processes take longer than planned (30 h versus a programmed 20 h);
- disposal of the waste produced is a critical part of the process.

6. CONCLUSION

The C.E.A spent fuel management plan has been operational since mid-1995. The CEA in cooperation with the COGEMA Group has demonstrated and established extensive experience in this area by recovering, conditioning and reprocessing a wide range of spent fuels, whole or in pieces, some of them stored in pools for over 25 years.

Before the shut-down of Marcoule reprocessing plant at the end of 1997, half of the UNGG from Saclay and Cadarache, and "Caramel-type" fuels have been reprocessed and the others (MTR, PWR, FBR and BSC/UNGG) completely reprocessed.

Since October 1996, the CEA is studying the problem of the remaining spent fuels. One of the solutions might be to send them to the COGEMA reprocessing plant at La Hague after a few years storage in Cadarache.

7. COGEMA'S EXPERIENCE WITH RESEARCH REACTOR AND EXOTIC SPENT FUEL MANAGEMENT

COGEMA has gained a large amount of experience in the reprocessing of research reactor and exotic spent fuels. Besides gas-graphite natural uranium fuel, the COGEMA Marcoule UP1 plant has, for instance, reprocessed for many years MTR spent fuels from the CEA and has been one of the key facilities in the CEA spent fuel management plan.

The COGEMA La Hague UP2 plant has processed several types of fuel since it begun its operations, most of them were LWR fuels, but exotic fuels like those coming from FBR were also reprocessed.

COGEMA has gained an industrial experience in the reprocessing of a wide variety of fuels including MTR, gas-graphite natural Uranium and FBR fuels. Although the Marcoule UP1 reprocessing plant is now shut down, the COGEMA La Hague plant is taking over. The La Hague plant is already available for U-Al and UO_2 type MTR spent fuel. R&D and feasibility studies are in progress to cope with metallic natural uranium (e.g. gas graphite) fuel as well as U_3Si_2, U-Mo, U-Zr-Nb MTR fuel. Through reprocessing, COGEMA offers a final and already proven solution for the durable management of research and test reactor spent fuel. The main features are :

- reduction of the ultimate waste volume and toxicity;
- recovery of the fissile materials, i.e. low enriched uranium of which can be re-enriched;
- safe and reliable confinement of the final waste in the form of qualified packages accepted in France and in many other countries;
- integrated services starting by taking delivery of the spent fuel at the reactor.

The COGEMA solution has already been selected by :

- CEA (France);
- Institut Laue-Langevin - Grenoble for RHF fuel; and,
- SCK-CEN - Mol Belgium for BR2 Fuel.

8. TRANSNUCLEAIRE'S EXPERIENCE WITH RESEARCH REACTOR AND EXOTIC SPENT FUEL TRANSPORTATION AND SERVICES

The Transnucléaire group has more than 30 years of specialized experience in activities in support of transportation, storage and handling of spent nuclear fuel (from both power and research reactors) on a worldwide basis.

Up to 1976, the TN Group carried out transports of MTR fuel elements and other fuel elements to Eurochemic at Mol, Belgium, and to Marcoule in France. Shipments originated

from France, Europe and Japan. Between 1978 and 1988 large quantities of MTR spent fuel (approximately 365 cask shipments) were made to the USA for reprocessing by AEC/DOE at either the Savannah River site in South Carolina or Idaho National Engineering Laboratory in Idaho (INEL). After 1988, the TN group continued to perform regular shipments of MTR spent fuel to Dounreay as well as to national reprocessing and/or storage facilities. In 1994, transportation of MTR spent fuel to the Savannah River site was restarted and at the end of 1997 the first deliveries of MTR spent fuel were made to La Hague reprocessing plant.

During this time the Transnucléaire group has developed and demonstrated the full capabilities required for successful support of transportation needs, including certification, procurement and operation of spent fuel casks, canisters for failed fuel assemblies, and auxiliary equipment required for loading and transportation of spent fuel. This expertise is not limited to the packaging, but extends to all considerations needed for operation, such as canister loading, lifting and handling equipment, transport frames and trailers, operating procedures, training and quality insurance. This experience includes cask loading and unloading, site and facility surveys, design and fabrication of facility/package interface equipment, package tie-down and restraint designs for trucks, railroad cars, and ship carriage, package maintenance and transportation planning and implementation.

Since 1963, the majority of the spent fuel assemblies shipped to reprocessing plants in France and the U.K. has been and continues to be delivered in casks designed, certified, manufactured, and operated by the TN group.

Several casks of the TN family have been developed to cover specific needs. Up to 1999, the IU04 cask, recently re-licensed in France and the US will continue to be used for MTR shipments. A new cask the TN-MTR with a much larger payload to weight ratio than the IUO4 will be in operation in 1999.

The TN81 cask has been developed by TN to transport vitrified high level residues from the reprocessing plant (La Hague) to interim storage facilities.

Other casks, including dual purpose casks for transport and storage, using several technologies are under development by TN.

SGN has developed for the CEA, Cadarache, the CASCAD facility, where unusual fuel elements, previously placed in canisters, are stored for 50 years in sealed pits ventilated by natural convection. A multipurpose storage facility has been designed by SGN to cover the needs of COVRA (Dutch central organization for radioactive waste storage). COVRA intends to store in this facility :

- intermediate and high-level waste arising from reprocessing of Dodewaard and Borsselle fuel at BNFL and COGEMA;
- spent HEU and LEU fuel from the three research reactors in Petten and Delft; and
- miscellaneous intermediate and high-level waste from the ECN at Petten.

TREATMENT OF SPENT FUELS FROM RESEARCH REACTORS AND REACTOR DEVELOPMENT PROGRAMS IN GERMANY

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Abstract

Quite a great number of different types of spent fuel from research reactors and development programs exists in Germany. The general policy is to send back to the USA as long as possible fuel from MTRs and TRIGAs of USA origin. An option is reprocessing in Great Britain or France. This option is pursued as long as reprocessing and reuse of the recovered material is economically justifiable. For those fuels which cannot be returned to the USA or which will not be reprocessed, a domestic back-up solution of spent fuel management has been developed in Germany, compatible with the management of spent fuel from power reactors. It consists in dry storage in special casks and, later on, direct disposal. Preliminary results from experimental R&D investigations with research reactor fuel and experience from LWR fuel lead to the conclusion that the direct disposal option even for research reactor fuel or exotic fuel does not impose major technical difficulties for the German waste management and disposal concept.

1. RESEARCH AND EXPERIMENTAL REACTORS IN GERMANY

At the end of 1997, six research reactors with a power ≥ 100 kW were operating in Germany. These comprise two TRIGA-type reactors, three swimming-pool reactors and one DIDO-type reactor. Moreover, one new research reactor is under construction, the operational start-up of which is expected in 2001. More details of these reactors are given in Table I. In addition to these reactors, there are eight research reactors permanently shut down, some of which have already been dismantled. These reactors are also included in Table I. It has to be mentioned, that the now-defunct research reactor at Rossendorf near Dresden and its fuel have been delivered by the former Soviet Union.

Table I: Research reactors in Germany (12/97)

Reactor	Location	Type	Power	Remarks
BER II	Berlin	pool	19 MW	in operation
FRG-1	Geesthacht	pool	5 MW	in operation
FRJ-2	Jülich	tank (DIDO)	23 MW	in operation
FRM I	München	pool	4 MW	in operation
TRIGA-MZ	Mainz	TRIGA	100 kW	in operation
TRIGA-HD	Heidelberg	TRIGA	250 kW	in operation, shut-down 1999?
FRM II	München	tank	20 MW	under construction
TRIGA-H	Hannover	TRIGA	250 kW	shut-down 1996
FRMB	Braunschweig	pool	1 MW	shut-down 1995
FRG-2	Geesthacht	pool	15 MW	shut-down 1992
RFR	Rossendorf	tank	10 MW	shut-down 1991
FRJ-1	Jülich	pool	10 MW	shut-down 1985
FRF	Frankfurt	TRIGA	1 MW	shut-down 1983
FRN	Neuherberg	TRIGA	1 MW	shut-down 1982
FR 2	Karlsruhe	tank	44 MW	shut-down 1981

The German research reactors are or were used for basic research in solid state physics (neutron scattering), isotope production and neutron activation analysis in medicine and biology, the exploration of radiation impacts upon materials and, finally, research on the behaviour of nuclear fuel. It is vital to continue current investigations in the future which renders further operation of some German research reactors indispensable. Safe management of the irradiated fuel elements arising now and in the future is therefore of primary importance. Furthermore, there are some facilities permanently shut down with considerable quantities of spent fuel, the safe treatment of which is a matter of urgency. These include, above all, the research reactor RFR, the zero-power installation RRR and the critical assembly RAKE at VKTA Rossendorf in the former German Democratic Republic.

Moreover, quite a number of training reactors (the likes of Siemens Unterrichts Reaktoren (SUR)) mainly at universities as well as critical assemblies or zero-power installations exist in Germany, the fuel of which has also to be managed after shut-down. To give an order of magnitude: nine facilities of that kind are still in operation, nineteen facilities have been shut down permanently, eight of them have been dismantled already.

Besides research and training reactors, eight pilot plants and prototype reactors were in operation in Germany. Activities in these eight plants were devoted to the testing of new reactor types and new fuel cycles. They have been all permanently shut down in the meantime, and the spent fuel of most of these plants has been reprocessed in Germany or abroad. The fuel of some of these plants, however, needs to be treated. Examples are the high-temperature gas-cooled pebble bed reactors AVR and THTR 300, the compact sodium-cooled nuclear power plant KNK II, and the nuclear ship Otto Hahn. AVR, THTR 300 and KNK II are in the decommissioning phase now, whereas the nuclear components of the Otto Hahn were dismantled in the early 1980s already. The ship was later used as a cargo ship with a conventional engine. Some characteristic data of these plants are given in Table II.

Table II: Characteristic data of some pilot plants and prototype reactors in Germany

Reactor	Location	Type	Power	Operation
AVR	Jülich	} gas cooled	15 MW_e	1966 - 1988
THTR 300	Hamm	pebble bed	300 MW_e	1984 - 1988
KNK II	Karlsruhe	sodium cooled FBR	20 MW_e	1977 - 1991
Otto Hahn	Hamburg	nuclear ship, PWR	38 MW_{th}	1968 - 1978

2. MANAGEMENT OF SPENT FUEL FROM RESEARCH REACTORS

2.1. Spent fuel arisings

If all MTRs are operated according to plan, some 100 irradiated fuel assemblies are discharged annually in Germany, being equivalent to < 440 kg. In addition, at the beginning of 1997, there was an inventory of about 800 irradiated fuel assemblies of various enrichment at the sites of the research reactors in former West Germany. To that a sizeable quantity of various types of fuel assemblies from VKTA's now-defunct reactors has to be added, being the equivalent of some 690 MTR-type fuel assemblies.

As far as TRIGA- or SUR-type reactors are concerned, no annual discharge of irradiated fuel has to be taken into account. These fuels have to be managed only after permanent shut-down of the installation.

Some typical characteristic data of German research reactor fuel assemblies are given in Table III.

Table III: Characteristic data of German research reactor fuel assemblies

Reactor	Dimension [mm]	Weight [kg] U-35	U	Total	Enrich. [%]	Type of Fuel
BER II		0,18	0,20	5,5	90	U Al_x - Al
FRG-1/2	873 x 76 x 81	0,32	1,63	7,0	20	U_3Si_2 - Al
		(0,27	1,37	6,6	20	U_3O_8 - Al)
FRM I	873 x 76 x 79	0,23	0,25	5,8	90	U Al_x - Al
		0,15	0,19	2,7	80	U Al_x - Al
FRJ-2	630 x 94 Ø	0,20	1,00	3,5	20	U_3O_8 - Al
		0,20	1,00	4,0	20	$U_x Si_y$ - Al
FRM II	1000 x 224 Ø	7,45	8,01	50	93	U_3Si_2 - Al
FRM B	873 x 76 x 80	0,14	0,16	5,5	90	U Al_x - Al
FRJ-1	638 x 76 x 76	0,26	0,33	4,4	80	U Al_x - Al
RFR EK-10	750 x 68 x 68	0,13	1,28	3,2	10	UO_2 - Mg
RFR WWR-M2	865 x 70 x 70	0,12	0,33	3,0	36	U Al_x - Al
TRIGA	721 x 38 Ø	0,038	0,18	3,5	20	U-ZrH
SUR	247/281 x 240 Ø	0,06/,07	3,3/3,8	13,5/15,3	20	PE + U_3O_8

2.2. Spent fuel management policy

Except for the Rossendorf facilities whose fuel assemblies are of Soviet origin and for which spent fuel management has to be provided in the beginning through interim storage, all the German research reactors have had their highly enriched fuel (HEU) supplied from the USA. Until the early 1970s, the spent fuel assemblies had been reprocessed in different European facilities, especially in Dounreay. Later, they were shipped to the USA (Savannah River) and reprocessed in the DOE facilities. At that time, the reprocessing waste arising in the USA and in Europe was kept in the country where the fuel had been reprocessed.

Due to protests of US environmentalists, reprocessing of foreign research reactor fuel was halted in the USA by the end of 1988. This led Germany to consider its own way of spent research reactor fuel management without giving up attempts, however, to open up again the Savannah River option.

Having all that in mind, a Germany-based option has been developed since the early 1990s with support of the Federal Ministry of Education, Science, Research and Technology (BMBF) which is well in line with the concept as applied to nuclear power plants in Germany. It includes dry storage of spent fuel assemblies in special casks. The fuel assemblies are subsequently conditioned for direct disposal without reprocessing. More details are given in chapter 2.3.

Due to urgent requests of influential political institutions to comply with the American non-proliferation policy and after completion of an environmental impact statement for the Savannah River site, DOE resumed taking back research reactor fuel assemblies of US origin in the summer of 1996. This take-back action was guaranteed for 10 years. Yet, the US authorities expect the individual countries to provide their own spent fuel management solutions afterwards.

Since its reopening, the US option is of top priority for research reactor fuel assemblies of US origin. Physikalisch Technische Bundesanstalt (PTB), owner of the permanently shut down MTR at Braunschweig, was among the first operators to ship the fuel of its research reactor to Savannah River in September 1996. In the meantime, another German station followed and more will follow soon.

Nevertheless, from the BMBF's point of view there remained some political uncertainty. In addition, the US solution is only effective until 2006. Similar international contracts which provide long-term solutions for spent fuel do not exist. That is why a domestic solution is rigorously sought in Germany, especially for fuel assemblies of non-US origin.

In case the US option proves problematic again while the domestic solution is not yet at hand, the only way out would be interim storage followed by reprocessing at AEA Technology, Dounreay, or at Cogéma's La Hague facility. In the past, the four German MTR stations currently operating and PTB at Braunschweig have signed reprocessing contracts with AEA Technology for some 500 fuel assemblies. The ensuing reprocessing waste will be shipped back to Germany.

As far as spent TRIGA fuel is concerned, Medizinische Hochschule Hannover is about to sign a contract with US-DOE for sending back the fuel from its permanently shut down reactor to INEEL at Idaho, USA. Transportation is planned for 1999. The operator of the Heidelberg TRIGA reactor which will be permanently shut down most probably in 1999 is also negotiating with US-DOE about taking back that fuel but keeps the option open to pursue a domestic solution. The TRIGA reactor at Mainz which has been refurbished recently will be operated well beyond 2006. Bearing the present US policy in mind, the domestic option for spent fuel might be pursued once that the reactor is shut down permanently.

No decision has been taken so far with regard to spent fuel management of the permanently shut down SURs. The "spent" fuel which has normally the quality of unirradiated material consists of 20% enriched U_3O_8 dispersed in disc-shaped polyethylene with a diameter of 240 mm and a thickness lying between 5 and 50 mm. Up to now the fuel of the dismantled SURs is stored. Options which are discussed in Germany are:

(1) to consider the fuel as waste and to dispose it of one day after dilution in order to meet the repository acceptance criteria with respect to the specific fissile material content of the waste;
(2) to remove the polyethylene and reuse the recovered uranium.

2.3 The domestic spent fuel management option

A domestic spent fuel management concept for spent research reactor fuel has been under development in Germany since the beginning of the 1990s, based on the direct disposal concept for spent fuel from power reactors [1, 2]. This concept envisages, as a first step, 30 to 40 years of dry interim storage of the fuel in special casks in a central German interim storage facility. After appropriate conditioning, the fuel elements are prepared for direct disposal. Within the framework of an R&D project supported by BMBF, German industrial companies developed the appropriate techniques pertaining to this concept in close co-operation with German research centres and research reactor operators. The application for the transport and the storage licence have been filed in 1993 and 1995, respectively. According to current planning, centralized storage of some research reactor fuel will start in the second half of 1998.

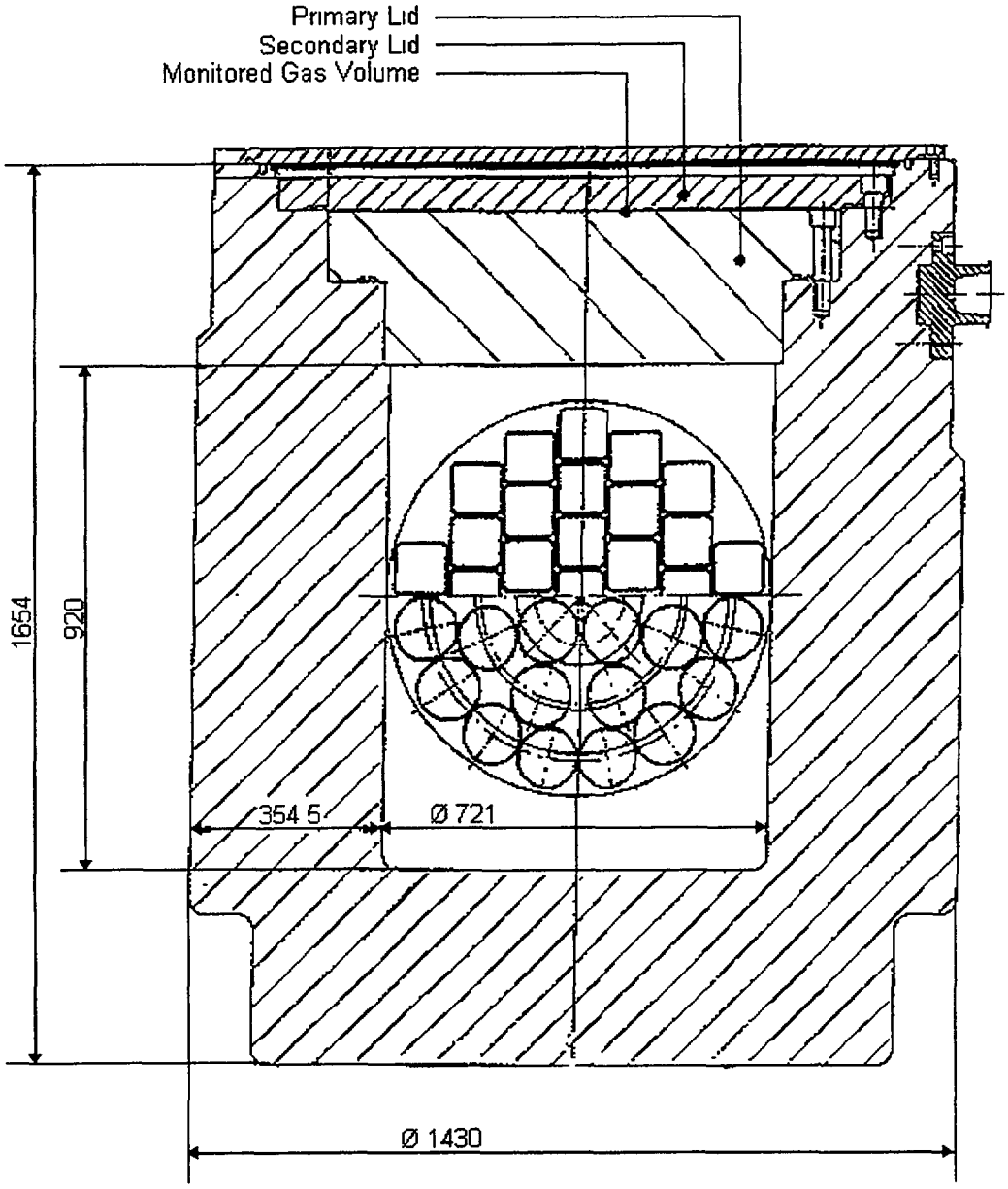

Figure 1: CASTOR MTR2 cask

Table IV: CASTOR MTR2 characteristics

External diameter (without shock absorbers)	1430 mm
Overall height (without shock absorbers)	1679 mm
Diameter of inner compartment	721 mm
Height of inner compartment	920 mm
Payload	
– box type MTR fuel assemblies	33
– tubular MTR fuel assemblies	28
– WWR-M2 fuel assemblies	49
– EK-10 fuel assemblies	42
– TRIGA fuel assemblies	78
Cask weight (loaded, without shock absorbers)	16 000 kg

According to the existing interim storage technology for power reactor fuel, Gesellschaft für Nuklear-Behälter (GNB) at Essen has developed the CASTOR MTR2 cask which is especially adjusted to the conditions at the research reactor sites The cask consists of a thick-walled cylindrical body made of ductile nodular cast iron and a double-lid sealing system which makes monitoring possible. The CASTOR MTR2 cask is schematically shown in Figure 1, some characteristic data can be derived from Table IV.

A number of storage racks whose geometry and content of neutron absorption material are adjusted to the requirements of the fuel assemblies to be stored, serve to accommodate and position the fuel elements inside the cask. The cask is designed such that it can hold all types of fuel assemblies of all existing MTR and TRIGA reactors as well as the fuel elements form the new FRM II under construction. For storage purposes, the interior of the cask is dried to a water content of less than 10 g H_2O/m^3 and filled with an inert gas (helium). This prevents corrosion of both the stored fuel assemblies and the metal seals at the lids.

The dimensions and the weight of the CASTOR MTR2 cask are such that it can be loaded and handled in the storage pools of most research reactor stations. For research reactors where underwater loading of the CASTOR MTR2 is technically not feasible, Noell (Freiberg) has developed a mobile loading unit. This unit will be used at Rossendorf, Berlin, and all TRIGA stations. VKTA Rossendorf, the future owner of this unit, has applied for the permit of its utilization at the end of 1993. The unit, which is now in its final stage of construction and testing, is schematically shown in Figure 2.

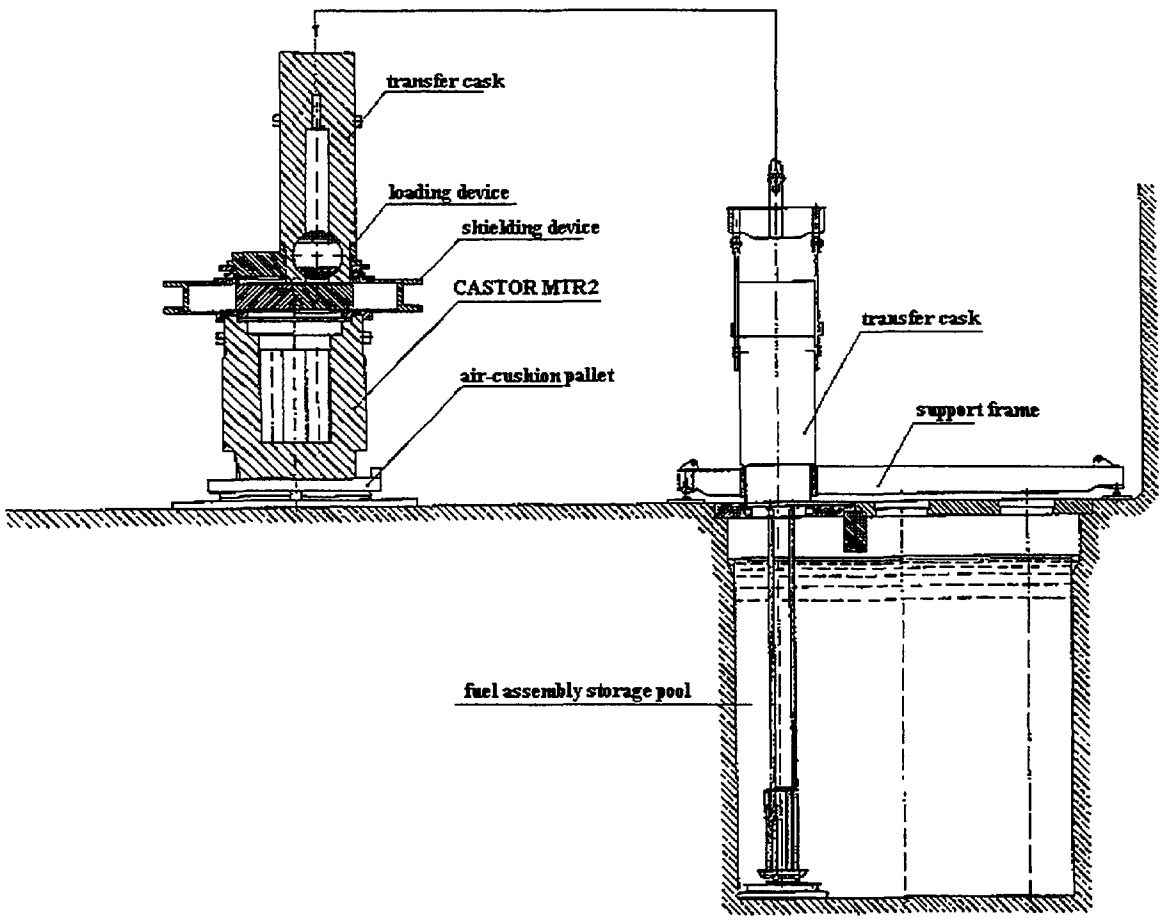

Figure 2: Mobile loading unit with transfer cask

Handling of research reactor fuel includes the following steps:

(1) Criticality-safety in the storage pool is assured by placing the fuel assemblies in so-called storage racks.
(2) Loading of the storage racks in a transfer cask which can be manipulated by the reactor crane.
(3) Transportation of the transfer cask to the mobile loading station mounted on top of the CASTOR MTR2 and reloading of the storage racks into the CASTOR MTR2 cask.
(4) Sealing of the CASTOR MTR2 cask.
(5) The CASTOR MTR2 is ready for interim storage and leaves the reactor hall by means of a pneumatic transport device.

The entire device can be transported in a 20-foot container and can therefore be used at different reactor sites.

The Jülich Research Center whose DIDO reactor will contribute some 30% to Germany's entire irradiated fuel discharged from research reactors by 2010 is performing a large-scale experimental research program on the behaviour of MTR fuel under dry storage [3] and final disposal conditions [4]. Research on intact fuel assemblies under dry storage conditions was completed in 1996, continuing research is focused on defect assemblies. Under representative dry storage conditions, only very small amounts of C-14 and H-3 were released from intact fuel assemblies, stemming from activated corrosion products in the outer layer of the fuel cladding.

Gesellschaft für Nuklear-Service (GNS) operates the spent fuel storage facility at Ahaus (BZA) and has offered storage contracts to the research reactor operators. At special positions of BZA three CASTOR MTR2 casks can be stacked one on top of the other. As it is not clear right now how many research reactor fuel assemblies can be shipped back to the USA, it is uncertain how many of the 30 positions originally planned at BZA will really be needed. Anyhow, the fuel from the Rossendorf facilities of Soviet origin cannot be shipped back to the USA. These fuel assemblies are the first candidates for interim storage at Ahaus. VKTA Rossendorf has ordered 17 CASTOR MTR2 casks from GNB. Loading and shipping of this fuel is expected for the second half of 1998.

After interim storage the spent research reactor fuel will be repackaged in a conditioning plant into a disposal cask. GNS is currently constructing a pilot conditioning plant at the site of its interim storage facility at Gorleben where the conditioning of spent LWR fuel is to be demonstrated. The plant will take up operation in 1999. It is designed in such a universal manner that it will also be able to handle spent research reactor fuel. The current reference concept for direct disposal of spent LWR fuel is as follows: The fuel elements are disassembled, the fuel pins are consolidated in POLLUX casks, and finally, the sealed and heavily shielded POLLUX casks are disposed of in the drifts of a deep geological repository in a salt dome [1]. In case the German direct disposal concept for spent LWR fuel will be changed one day, it can be adapted in such a way that it will accommodate the spent fuel from research reactors as well.

In 1993, a cost comparison for the management of spent research reactor fuel has been conducted in Germany, comparing domestic interim storage plus direct disposal with foreign reprocessing plus domestic waste disposal. The result of this comparison was that the domestic option is markedly cheaper than foreign reprocessing [5].

3 MANAGEMENT OF SPENT FUEL FROM PILOT PLANTS AND PROTOTYPE REACTORS

Several kinds of pilot plants and prototype reactors were in operation in Germany from the 1960s to the 1980s All of them have been permanently shut down in the meantime, some of them have already been dismantled The spent fuel of most of those installations has been reprocessed in Germany or abroad Some installations do exist, however, for which all or part of the fuel still needs to be treated

3.1. Gas-cooled pebble bed reactors AVR and THTR 300

The AVR at Julich and THTR 300 at Hamm have been operated from 1966 and 1984, respectively, to 1988 These gas-cooled reactors used spherical fuel elements which are schematically shown in Fig 3 Each 6-cm diameter fuel element consists of several ten thousands of coated particles with HEU-fuel embedded in a graphite matrix Most fuel elements also contain thorium

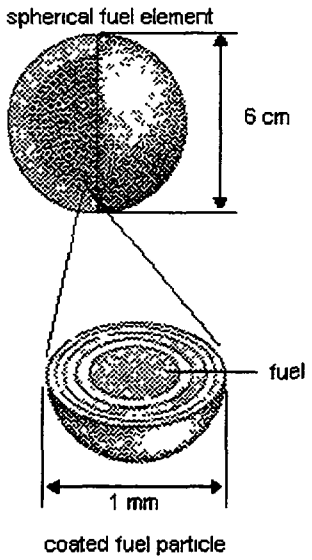

Figure 3: Fuel elements from AVR and THTR 300

About 288,000 fuel elements from AVR and 622,000 fuel elements from THTR 300 are stored in CASTOR casks at Julich and Ahaus, respectively Since no reprocessing technology is available for this type of fuel, these fuel elements are first-priority candidates for direct disposal Leaching experiments in salt brines have shown that these fuel elements exhibit excellent behaviour even under hypothetical accident conditions in a rock salt repository [6] Therefore, there will most likely be no need to switch from the interim storage casks to specially designed disposal casks one day The already existing CASTOR casks lend themselves to direct disposal of this type of fuel.

3.2. KNK II

KNK II was Germany's first fast nuclear power plant and served as a pilot plant for the prototype fast breeder reactor SNR 300 The initial design provided for a sodium-cooled but thermal reactor (KNK I) It was modified into a fast reactor later on and operated well until it was shut down for good in 1991 The fuel elements of KNK II-reached a high burnup, and closing of the U/Pu fuel cycle was demonstrated

KNK II used UO_2 -fuel with 6.75% enrichment, KNK II UO_2/PuO_2-fuel (MOX) with highly enriched Uranium. The KNK I fuel elements were sent to Marcoule in 1975 and were subsequently reprocessed there without any difficulty. Until 1993 the KNK II fuel was also shipped to Marcoule for reprocessing. Most of the MOX fuel could be reprocessed without problems but, due to the manufacturing process, some fuel was insoluble in nitric acid. The insoluble fuel still awaits appropriate treatment. In principle, these fuel pins could be stored in an interim storage facility and later on disposed of directly the same way as spent LWR fuel. Another possibility would be to reprocess this fuel in a special way at AEA Technology, Dounreay.

3.3. Nuclear ship Otto Hahn

The German nuclear ship Otto Hahn was operated from 1968 to 1978. During that time it covered a distance of 646,000 sm in 131 voyages, in 73 of which it was used as a cargo ship. The advanced pressurized water reactor operated well without major difficulties. In February 1979 the reactor was permanently shut down. The nuclear components were dismantled in the early 1980s. The ship was later operated with a conventional engine.

In its first core the Otto Hahn used SS-clad UO_2-fuel with an enrichment of 2.8 to 4.8%, whereas in the second core the fuel was Zircaloy-clad with 3.5 to 6.6% enrichment. The fuel was successfully reprocessed at the Karlsruhe reprocessing plant WAK. Only some fuel pins were left in the hot cells of the Geesthacht Research Centre for post-irradiation examinations. These fuel pins will, most probably, be reprocessed one day in France or Great Britain.

4. SPECIAL ASPECTS OF DIRECT DISPOSAL OF RESEARCH AND EXPERIMENTAL REACTOR FUEL

If direct disposal of research and experimental reactor fuel is envisaged ("experimental" reactor being a synonym for "pilot plant and prototype reactor") one has to keep in mind that spent fuel is not a waste form specially tailored for disposal and that the fuel still has a high concentration of fissile material. Therefore, aspects like the behaviour of the spent fuel under repository environment conditions and criticality in a repository have to be discussed in some detail.

4.1. Behaviour of spent research reactor fuel in a repository

Germany will dispose of its heat-generating waste in a deep geological repository located in a salt dome. The Gorleben salt some in Lower Saxony is being investigated for that purpose.

If properly back-filled and sealed after operation, the normal evolution of a repository in rock salt is such that the waste will not come into contact with any water or brine. Therefore, canister corrosion and fuel dissolution will not be part of the normal evolution scenario.

It cannot be totally ruled out, however, that small amounts of brine find their way to the waste one day via inhomogeneities in the salt dome. Therefore, Forschungszentrum Jülich in an R&D project is addressing the question how aluminium-clad research reactor fuel behaves in such a hypothetical accident scenario [4].

The results are very encouraging: Even though UAl_x-Al fuel dissolves rapidly in concentrated salt brines - in particular if iron as a corrosion product from the disposal cask is present - the released radionuclides will be immobilized again in the near-field by the corrosion products of the cask and the fuel cladding. Such type of barrier can be referred to as

"geochemical barrier" and was observed already in corrosion and dissolution tests with spent LWR fuel [7]. So, our preliminary conclusion is that spent MTR fuel is an adequate waste form for disposal.

4.2. Criticality in a repository

Disposal casks are designed such that criticality, even with highly enriched fuel, will not occur. Criticality has to be discussed only in case of cask corrosion and fuel dissolution. As mentioned already in chapter 4.1, this will not take place in the normal evolution scenario for a repository in a salt dome.

As far as the hypothetical accident scenario for a salt repository is concerned it could be shown that criticality does not occur in a repository with spent fuel from power reactors [8]. If, for example, spent research reactor fuel is not disposed of in a special disposal drift but is mixed with spent fuel from power reactors ("dilution" of the high fissile content of research reactor fuel), criticality would also be of minor concern. More detailed investigations are necessary, however, to prove this statement. On the other hand, one has to bear in mind that only small amounts of brine might come into contact with the waste. The brine contains chlorine which is a neutron absorber. That is why the critical mass in a brine-moderated repository in rock salt is about a factor of 10 larger than, e.g., the critical mass in a repository in hard rock with ground water moderation. Therefore, we think that criticality-safety in the repository can be proven more easily in rock salt than in hard rock.

5. CONCLUSION

A great number of different types of spent fuel from research reactors and development programs exists in Germany. Different strategies are at hand or are being developed to solve the spent fuel management problem. In the long run Germany might rely on its domestic solution consisting of dry storage followed by direct disposal.

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STORAGE AND MANAGEMENT OF FUEL FROM FAST BREEDER TEST REACTOR AND KALPAKKAM MINI

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Abstract

Two Research Reactors, FBTR (Fast Breeder Test Reactor) and KAMINI (KAlpakkam MINI) are in operation at Kalpakkan, India. FBTR is a 40 MWt reactor. It is the first reactor to use mixed carbide (70% PuC-30% UC) as driver fuel. Special precautions are needed to fabricate pellets in glove boxes under inert atmosphere to take into account the possibility of criticality, radiation, pyrophoricity and toxicity of PuC. FBTR has been operating with small core up to 12 MWt power. The initial limit was 250 W/cm linear heat rating and 25,000 MWd/t peak burnup. This limit was increased to 320 W/cm and 50,000 MWd/t respectively after rigorous analysis. At present the core has reached 40,000 MWd/t without any pin failure. After 25,000 MWd/t burnup one fuel subassembly (SA) was removed and PIE was carried out. The results were as expected by the analysis. In FBTR, fuel is stored in a container filled with argon and the container is cooled by forced circulation of air (during storage). Closing the fuel cycle is important for the breeder programme. Therefore, efforts have been made to set up a reprocessing plant. It uses the well proven purex process. The irradiated fuel is sheared in a single pin chopper and dissolved in an electrochemical dissolver. The resulting solution after adjusting the valency of Pu to IVth state is processed in the solvent extraction plant using 30% Tri-n-Butyl phosphate/n-dodecane as solvent. KAMINI is 30 kWt neutron source reactor which uses light water as moderator and coolant and has as a fuel U-233 aluminium alloy. Uranium-233 has been indigenously recovered from thorium irradiated in CIRUS reactor at Trombay. KAMINI was made critical on October 1996. It is housed in a vault below one of the hot cells in the Radiometallurgy laboratories of IGCAR. This reactor is planned to be used for neutron radiography of fuel elements and neutron activation analysis. It is available for use by research institutions and universities also. This paper describes the various stages of the fuel cycle of FBTR and KAMINI.

1. INTRODUCTION

Fast breeder test reactor (FBTR) is a 40 MWt reactor in operation at Kalpakkam, India. Normally in such small test reactors the fissile requirement of the core is met by highly enriched uranium along with plutonium and the reference fuel composition is 70% UO_2-30%/PuO_2. However, with the idea of replacing U^{235} completely by Pu^{239}, a plutonium rich mixed oxide fuel of the composition (U 0.24 - Pu 0.76)O_2 (natural uranium) was initially chosen in place of the already proven (U 0.7 -Pu 0.3)O_2 (enriched uranium) fuel. Unfortunately, preliminary metallurgical investigations revealed that (U 0.24 - Pu 0.76)O_2 is not compatible with sodium coolant and causes unacceptably high swelling rate. Therefore, this fuel was not pursued further for FBTR.

The next logical step was to explore the possibility of developing a plutonium rich mixed mono-carbide (MC) fuel. As compared to conventional MO_2 it has higher heavy atom density, higher thermal conductivity and more compatibility with sodium. Therefore, for the first time a fuel of composition 70% PuC-30% UC was selected as driver fuel in a fast reactor. Out of pile fuel-clad coolant compatibility experiments at 973K for 1000 hours confirmed the excellent compatibility with type 316 stainless steel and sodium. The materials used for clad tube and hexcan are 20 % cold worked (CW) 316 and 20% CW 316L respectively.

The fuel pellets and pins were fabricated at BARC, Trombay and fuel subassemblies were fabricated at Kalpakkam by Nuclear Fuel Corporation. The fuel was irradiated to 40,000 MWd/t without any failure of clad. Post irradiation examination was conducted on fuel SA after 25000 MWd/t burnup. It has shown encouraging results.

As closing the fuel cycle of breeder program is important, a reprocessing plant has been set up at Kalpakkam where FBTR fuel is planned to be reprocessed.

The various stages of fuel management are described below.

2. PELLET FABRICATION

Preparation of (U, Pu)C fuel pellets of controlled density and M_2C_6 phase contents is expensive and difficult compared to the conventional (U, Pu)O_2 pellets. More steps are required in the fabrication. (U, Pu)C is highly susceptible to oxidation and hydrolysis and is pyrophoric in the powder form, thereby needing a high purity inert cover gas (argon or N_2) with less than 25 ppm each of O_2 and moisture and continuous monitoring of the glove box atmosphere for oxygen and moisture content. Stringent control of oxygen, carbon and nitrogen content of the process intermediates is needed in order to avoid the metallic phase and to keep the sesquicarbide content within the acceptable limit of no more than 15 wt% in the final mixed carbide product. Since the vapour pressure of MC is higher than that of oxide, the temperature and vacuum at different steps in the process must be judiciously optimised so that plutonium volatilization losses are minimised. The two main steps in the fabrication are:

- Synthesis of (U,Pu)C in the form of buttons, fine powders, clinkers, granules, spheroids or microspheres, starting either from metal or the oxide.
- Consolidation of (U,Pu)C in the form of a fuel pellet by arc melting and casting, vibratory compaction or cold pressing and sintering.

Keeping in view the small annual requirement of fuel for FBTR a kilogram scale fabrication facility was set up at Trombay to produce carbide fuel pellets from UO_2 and Pu O_2 powder. The fabrication facility has two main wings, the pellet production wing and the process control wing. The pellet production set-up consists of 12 connected 1 to 2 m^3 glove boxes housing the powder metallurgy equipment. The process control wing consists of necessary instruments installed inside glove boxes for rapid physical and chemical analysis of the feed materials, process intermediates and the final product with respect to oxygen, nitrogen, carbon, uranium and plutonium, metallic impurities, phase contents and specific surface area of the powder.

High purity nitrogen is used as inert cover gas in all glove boxes on a "once through" basis at a flow rate of 60-100 l/min in order to have three or four box volume changes per hour. The glove boxes have high degree of leak-tightness, better than 0.02 vol% leakage per hour and are maintained at a slightly negative pressure of 50 to 200 Pa.

2.1. Stages for manufacture of pellets

The diameter of fuel pellet was based on the available stainless steel hardware procured initially for a mixed oxide core. The various stages for manufacture of pellets are given below:

2.2. Carbothermic reduction of oxide

UO_2, Pu O_2 and graphite are weighed with an accuracy of 0.01 g and the mixture is milled and ground for 12 hours in a planetary ball mill using agate bowls and tungsten carbide balls. The powder mixture is next pre-compacted at 7.5 MPa and then granulated and compacted at 75-150 MPa to approx. 10 mm diameter and 2 to 3 mm high tablets.

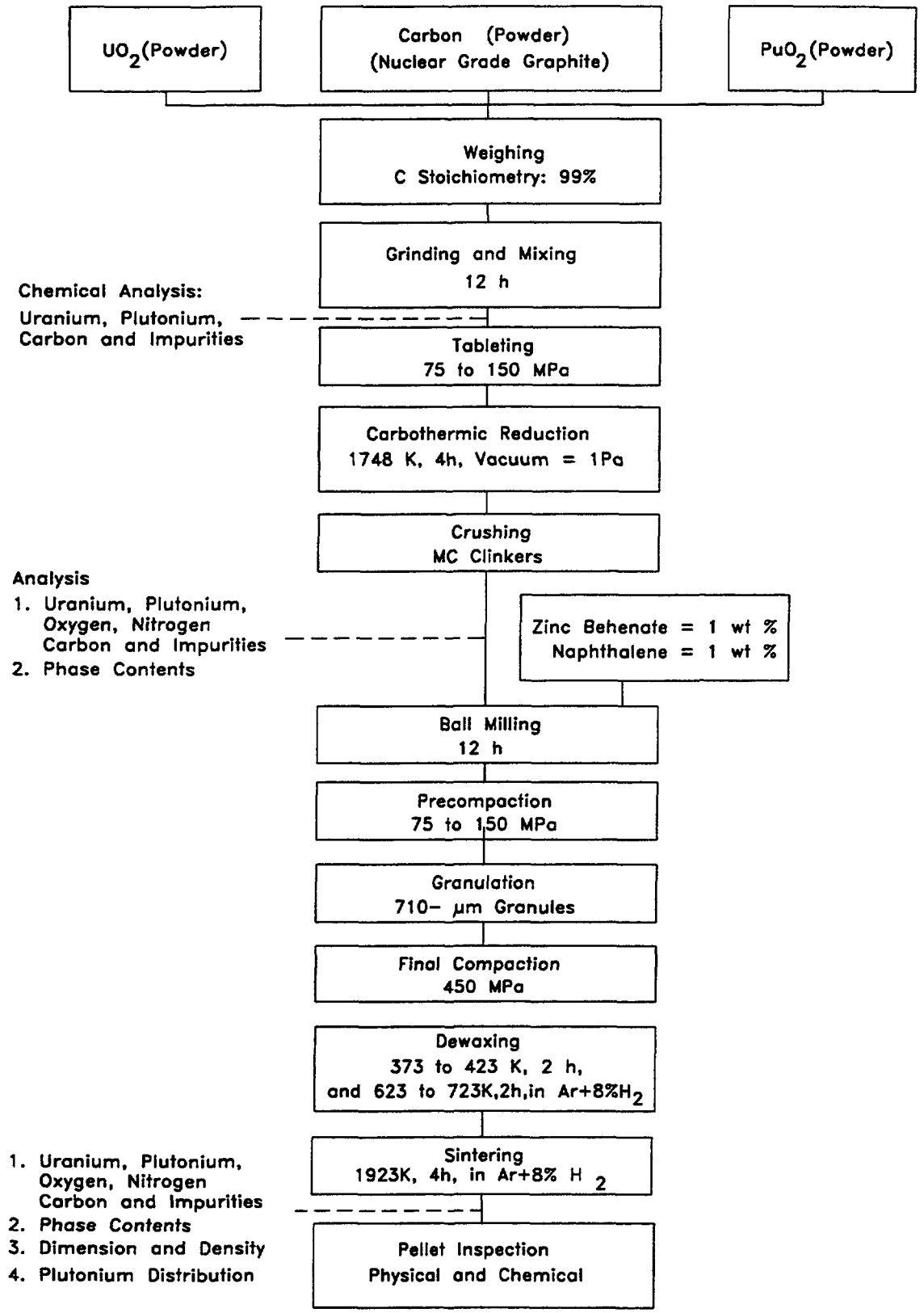


Fig.1 Flow sheet for fabrication of $(U_{0.3}Pu_{0.7})C$ pellets from UO_2, PuO_2 and graphite feed materials.

Approximately 600 g of these tablets are loaded into tantalum cups and subjected to vacuum carbothermic reduction at 1748 K for 4 h at approx. 1 Pa. Under these conditions, nearly 100% conversion to carbide has been achieved while keeping plutonium volatilization loss between 1 and 2 wt%.

2.3. Sintering of mixed carbide

The mixed carbide clinkers are crushed in a cross-beater mill and then milled for 12 hours in a planetary ball mill in batches of 0.8 to 1.2 kg in an agate bowl with tungsten carbide balls.

Next a binder-lubricant combination of napthalene and zinc behenate is mixed with the carbide power for 1 hour. This carbide binder powder mixture is pre-compacted between 75 and 150 MPa granulated and finally compacted at 450 MPa in a hydraulic press. The pellets are then loaded on tungsten trays, kept in a molybdenum charge carrier and de-waxed in a flowing Ar +8% H_2 atmosphere at 373 to 423 K for 2 h to remove the naphthalene and at 623 to 723 K for 2 h to remove the zinc behenate. The de-waxed pellets are subsequently loaded in a sintering furnace and sintered at 1923 K for 4 h in Ar +8% H_2. The resulting mixed carbide pellets have a shining steel lustre and meet all the required specifications.

These pellets having nominal diameter of 4.18 mm are put in clad tubes and sealed by welded end plugs. 61 pins are assembled to form a fuel subassembly.

3. PERFORMANCE OF MIXED CARBIDE FUEL

A small core with 23 fuel SA, called Mark 1, of the above composition was loaded in FBTR. At the time of loading, the limits set for the fuel were 250 W/cm for peak linear heat rating and 25000 MWd/t for peak fuel burnup. These limits were revised after detailed analysis to 320 W/cm and 50,000 MWd/t respectively. At present the fuel has crossed a peak burnup of 40,000 MWd/t without any failure.

3.1. Linear heat rating

The design criteria followed in fixing the allowable operating heat rating of fuel, is that there should be a margin of at least 15% between operating linear heat rating and the safety limit. The safety limit is defined as the rating at which incipient melting occurs after taking into account all uncertainties in properties and operation. The overpower trip threshold is set within the safety limit to take care of overshoot during transients.

3.2. Determination of safety limits

Maximum uncertainty in determining safety limits is caused by uncertainty in operating and linear power, the fuel-clad gap conductance, because of tolerances specified on clad outside diameter and thickness and pellet diameter. The cold diametrical gap varies over the range of 0.06 - 0.30 mm for fresh fuel. The gap conductance also varies with the linear heat rating because of temperature dependence of the hot gap and thermal conductivity of helium. There could be restructuring of fuel in reactor due to cracks developing owing to thermal stress and this could reduce the gap to some extent. In the out-of-pile tests, the fuel specimens were heated electrically up to a rating of 500 W/cm. The theoretical analysis without taking into account restructuring gave a design safety limit of 373 W/cm on linear power for fresh

fuel and 800 W/cm for fuel fully in contact with clad and reaching a saturated value of gap conductance of 1.5 $W/cm^2 \cdot K$. With this safety limit, the operating heat rating was limited to 320 W/cm for Mark-1 core fresh fuel untill some data on the swelling rate of fuel is obtained.

3.3. Burnup

Since carbide fuel of this composition is being used for the first time, there is no data available on the burnup capability of the fuel. The unavailability of data on swelling and creep causes uncertainty in predictions. Most of the data available for carbide fuel in literature are for standard mixed carbide fuel containing 15-20% PuC. This data have been suitably modified and adopted for FBTR fuel. The analysis to arrive at the achievable burnup is divided into two phases. In each phase different phenomena dominate and the total achievable burnup is the sum of the two. The first phase is known as free swelling phase in which the initial gap provided between clad and fuel pellet closes on account of fuel swelling due to irradiation. The achievable burnup in this is governed by linear power. For finding temperature distribution, the gap conductance is an important parameter. As the fuel bums, the fuel swells and the gap between fuel and clad reduces thereby increasing gap conductance. This reduces fuel centre temperature. Gap conductance keeps on increasing as the burnup proceeds and reaches a steady-state value as the fuel clad contact occurs. The time taken to reach this point determines the attainable burnup in the free swelling phase.

The second phase of the analysis is known as Fuel Clad Mechanical Interaction phase (FCMI). In this phase, once the pellet and clad make physical contact with each other, any further swelling of fuel exerts pressure on clad resulting in various stresses on fuel and clad thereby producing creep in them. FCMI pressure goes on increasing till a steady state equilibrium is attained between fuel swelling, clad creep and fuel creep. The steady-state value of gap conductance is assumed to be 1.5 W/cm^2K. The swelling rate of fuel becomes constant when the gap conductance saturates at the steady-state value. The life of fuel pin achievable in FCMI phase depends on creep rupture life of clad corresponding to FCMI steady state pressure.

The attainable burnup for Mark-1 core fuel of FBTR have been estimated for the entire core for a peak linear heat rating of 320 W/cm in order to consider the enhancement of burnup. The attainable burnup values for nominal temperature conditions and hot-spot temperature conditions are calculated. It indicated that burnup of 50,000 MWd/t is possible in the centre of core.

4. SPENT FUEL STORAGE

In FBTR fuel is discharged from the core and kept in an internal storage within the reactor vessel for cooling. It remains there until the decay heat reduces to less than 400 W. During subsequent fuel handling campaigns, the cooled fuel is removed from internal storage to the external storage with the help of a shielded and leak tight flask. External storage is a dry storage where shielding is provided by steel. Storage is divided in two zones, fissile zone and non-fissile zone. The storage capacity of fissile zone is 207 SA. Non-fissile zone can store 619 SA. The spent fuel is kept in an argon filled leak-tight container. The container is deposited in the spent fuel storage by transfer carriage and a shielded flask called secondary flask. In the external storage the container is cooled by forced circulation of air. The container is made leak-tight with multiple silicon O-rings. To maintain leak-tightness over prolonged storage, a low melting alloy seal has been provided over and above the O-ring seals.

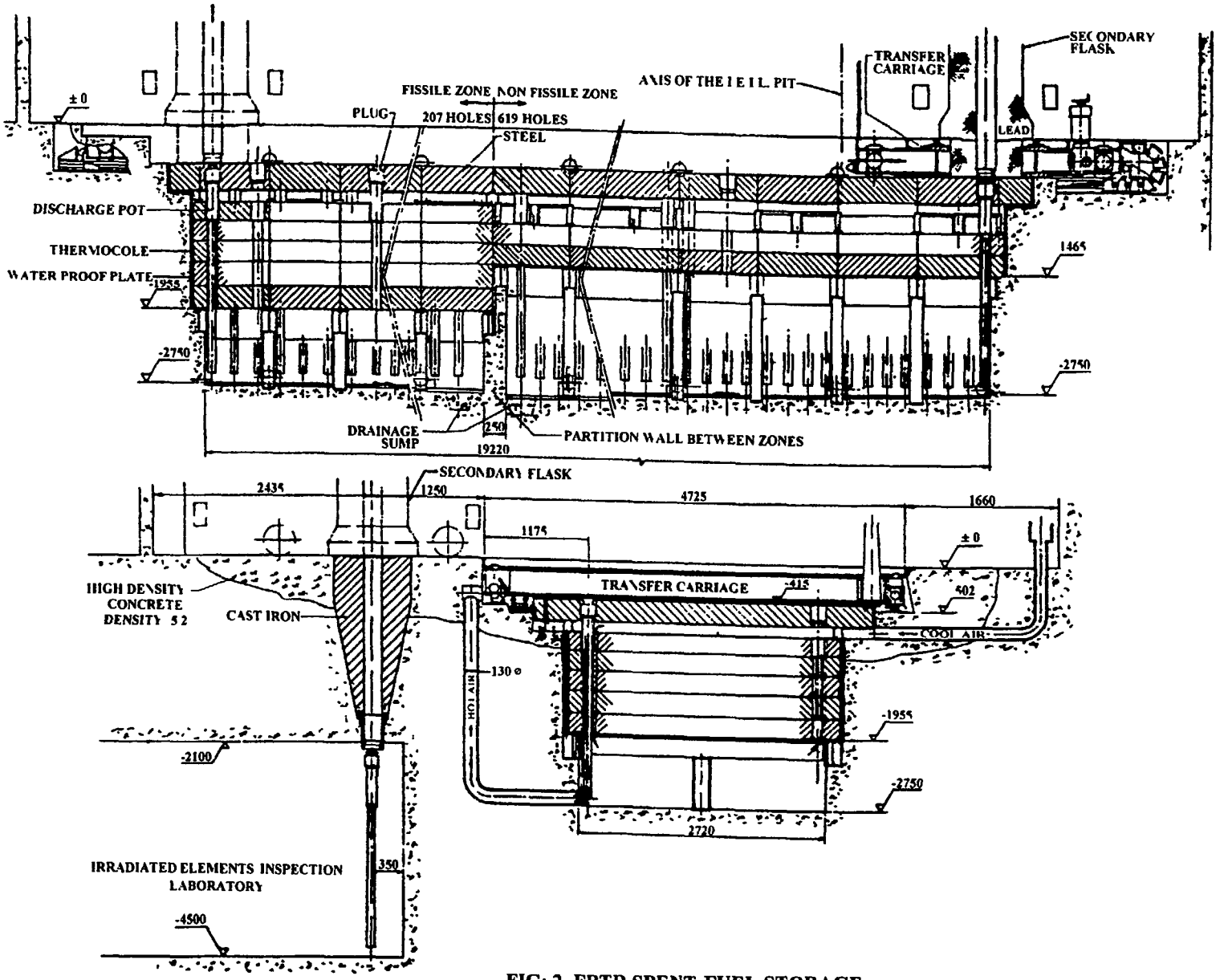


FIG: 2 FBTR SPENT FUEL STORAGE

The fuel along with the container is transferred by secondary flask, which deposits the container in an underground flask. The underground flask transfers the fuel SA to the cell of the radiometallurgical laboratory (Irradiated Element Inspection Laboratory) through an opening in the floor of the cell. Cutting of fuel SA and PIE is carried out in the cell. From there it is sent to the reprocessing facility.

5. POST IRRADIATION EXAMINATION

As mentioned above, the free swelling rate of the fuel has been derived from the data on (15-20% Pu-U)C fuel from literature and peak linear heat rating has been increased from 250 W/cm to 320 W/cm. To verify the data and to understand the behaviour of fuel and clad, Post-Irradiation Examination (PIE) was carried out on a fuel subassembly (SA) discharged from FBTR after reaching a burnup of 25,000 MWd/t. In addition, PIE excluding metallography has been carried out on four experimental fuel pins irradiated in FBTR. These fuel pins contain fuels with the present core composition as well as the proposed expanded Mark-II core of composition (55% Pu C, 45% UC). The irradiation experiments are undertaken to understand the beginning of life performance of these fuels, compare the performance of the fuel compositions of the present Mark-I core and of the proposed expanded Mark-II Core.

5.1. PIE facility

PIE facilities at the Radiometallurgical Laboratory can handle and examine highly irradiated advanced fuels in the inert gas (nitrogen) atmosphere hot cells where temperature, pressure and cell atmosphere are closely monitored and controlled. The PIE facilities consists of seven concrete shielded hot cells and a few lead shielded cells. A wide spectrum of non-destructive and destructive examination are covered. The following are performed in the cells:-

1. Sodium removal from fuel SA by ethanol;
2. Dismantling of fuel SA without damaging the pins by remotely operated CNS drilling and milling machines;
3. Neutron radiography in 30 kWt swimming pool type reactor KAMINI installed below the hot cells;
4. Leak testing of fuel pins by glycol leak test;
5. Measurement of diameter of fuel pins;
6. Eddy current testing of fuel pins to detect abnormalities on the clad tube as well as at clad-fuel interface;
7. X-radiography of fuel pins;
8. Micro gamma scanning to evaluate the fissile column length, distribution of fission products and burnup;
9. Preparation and examination of metallographic specimen from fuel pin.

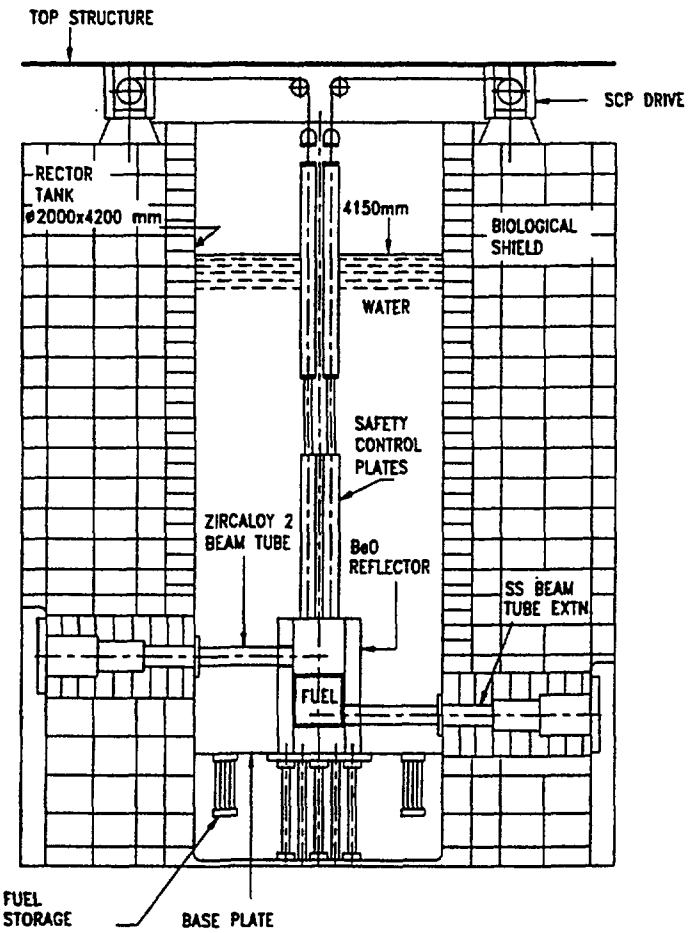
5.2. PIE of FBTR fuel subassembly

Central fuel SA from FBTR with total burnup of 25,036 MWd/t and peak linear heat rating of 320 W/cm was discharged in July 1996 and taken up for PIE. The following were the observation:

1. The shinning visual appearance of hexagonal sheath after sodium cleaning indicated no corrosion;

2. Dimensional measurement on hexagonal sheath did not show any swelling or bulging;
3. Fuel pins extracted from SA did not show any corrosion or deformation on visual examination. Only the central fuel stack regions of clad appeared coloured due to effect of temperature;
4. Glycol leak testing of fuel pins did not indicate any leak;
5. Measurement of diameter and length of pins indicated no deformation of clad;
6. Eddy current test of fuel pins did not indicate any surface defect or abnormality on the clad tube. The location of spring support indicated that pellet length has increased due to swelling of fuel;
7. X-radiography of fuel pins indicated increase in fuel stack length in the range of 2.17 mm to 5.35 mm;
8. Metallography was done on two pins. Several cracks at the middle section of fuel were observed. The fuel-clad gap was seen to be closing due to cracking as well as swelling of fuel.

The average swelling rate estimated from X-radiography results of nine pins as well as measured from the metallographic cross sections is approximately 1.2% per atom percent burnup. This indicated that the swelling rate for FBTR fuel has been lower than the predicted values. Hence linear heat rating can be further raised and is targeted as 400 W/cm based on analysis. The presence of fuel-clad gap seen in the metallographic cross section as due to cracks, indicate that, space is still available to accommodate further swelling of fuel. Hence fuel burnup can be further raised and is targeted as 50,000 MWd/t.



Reactor System	
power	30 kWt
Fuel	U-233 Al Alloy
Moderator & Coolant	Light Water
Reflector	200 mm Beryllium Oxide canned in zircaloy followed by water
Control	Two absorber type safety control plates (cadmium)
Flux	
– Core Avg	10^{12} n cm^{-2} s^{-1}
– Radiography Site	10^{6}–10^{7}n cm^{-2} s^{-1}
– sample irradiation Site	10^{10}–10^{11} n cm^{-2} s^{-1}
Biological shields	Ordinary and high density Concrete

Fig.3 KAMINI REACTOR

6. KAMINI REACTOR

The KAMINI reactor was made critical in October 1996 at IGCAR. It is located below the hot cells of the radiometallurgy laboratory. It is a neutron source, pool type reactor. It is used for activation analysis and neutron radiography. KAMINI is a U233 - aluminium alloy fuelled, light water moderated, beryllium oxide reflected reactor with thermal power of 30 kW. The fuel alloy contains 20 per cent uranium. A little less than 600 gm of U^{233} is used in the reactor core. 25 mm x 5.5 cm, 1 mm thick fuel plate is sheathed in 0.5 mm thick aluminium sheet. Eight such fuel plates are assembled in an aluminium frame to form one fuel subassembly. Nine such fuel subassemblies are used in the reactor core in a square matrix of 3 x 3. Beryllium oxide reflector of 20 cm thickness is used all around and at top and bottom of core. The reflector consists of BeO of different sizes and shapes encased in Zircaloy boxes. These reflector blocks make it possible to progressively decrease the, rate of reactivity addition during approach to criticality, and they can be used for long term compensation of reactivity loss due to fuel burnup. The reactor vessel is a cylindrical tank of 1 m height. The safety system of the reactor consists of a fast acting primary shutdown mechanism using two cadmium plates sandwiched in aluminium, falling into the reactor core under gravity. The reactor regulation is also accomplished with the help of these two cadmium elements. Highly active fuel SA discharged from FBTR can be transferred from the hot cell to a position in front of one of the beam tubes of KAMINI reactor through a sealed pipe.

The average neutron flux is about 10^{12} n/cm^2·s and at radiography site it is 10^6-10^7 n/cm^2·s. U^{233} used in KAMINI was recovered from thorium irradiated in CIRUS reactor. At present there is no plan to reprocess the spent fuel of KAMINI.

7. FUEL REPROCESSING

The success of fast breeder technology lies in the successful closure of the fuel cycle through reprocessing route. Fast reactor fuel reprocessing (FRFR) in India is a step to utilise the existing natural uranium resource effectively and also to tap the vast thorium resources of the country.

7.1. Fast reactor versus thermal reactor fuel reprocessing (TRFR)

The fuel of fast reactors which contains high plutonium contents needs a higher order of critically safe equipment. Therefore, it cannot be reprocessed in TRFR plants. As the short cooled fuel has high radioactivity, there is need to develop fast solvent contactors to reduce the solvent damage. The presence of large plutonium content demands leak-tightness of high order due to alpha particles.

7.2. Reprocessing FBTR fuel

The fuel is highly refractory in nature and its high radioactivity (approx. 1 TBq/g) makeing the reprocessing a highly challenging task.

Reprocessing involves a number of unit operations, fuel dissolution, fluid transport, evaporation, distillation and precipitation. The heart of the process is solvent extraction. The well proven operations become complicated owing to the necessity of carrying out these operations behind shielded cells. FBTR fuel reprocessing is based on the well proven purex process. The spent fuel is sheared in a single pin chopper and then dissolved in an

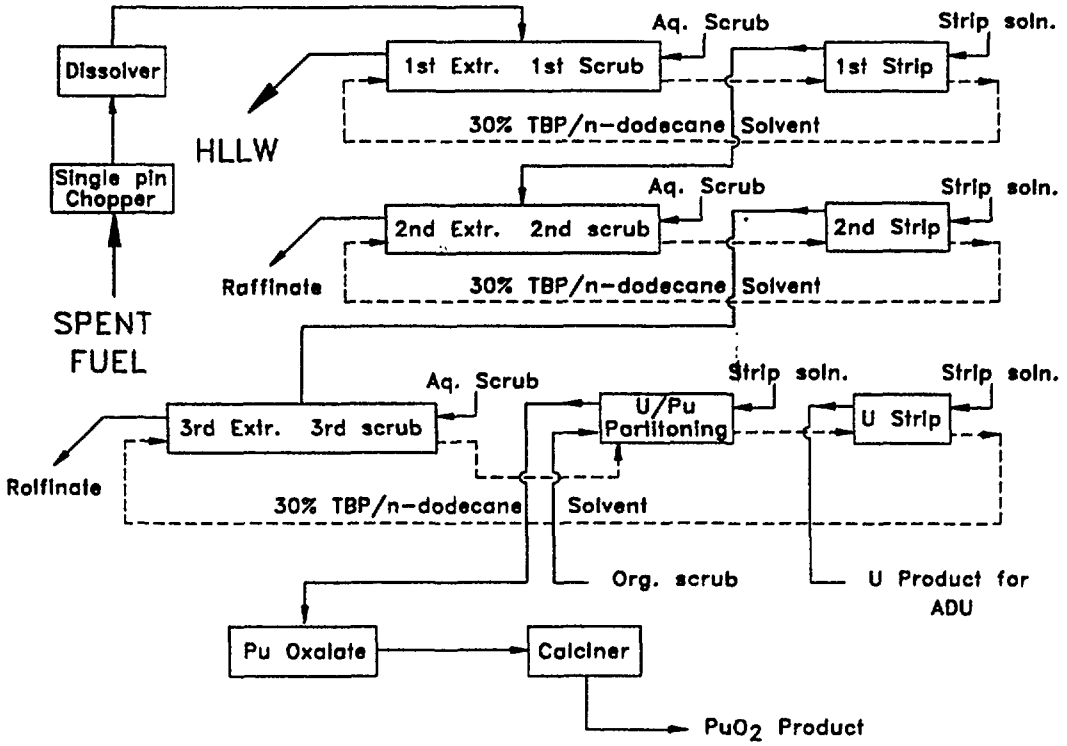


Fig.4 Reprocessing Flowsheet for FBTR Fuel

electrochemical dissolver. The resulting solution after adjusting the valency of Pu to IVth state is processed in the solvent extraction plant using 30% Tri-n-Butyl Phosphate (TBP)/n-dodecane as the solvent. TBP has a very high distribution coefficient for U, Pu compared to fission products. U, Pu can be selectively separated from fission products in a number of solvent extraction stages in counter current mode. As shown in flow sheets, there are two co-decontamination cycles, followed by a partitioning step wherein U and Pu can be separated from each other by reducing PU(IV) to PU(III) - an inextricable species, in an electro-reductive mixer-settler. Final U, Pu purification cycles result in meeting the required product specification.

7.3. Fuel dissolution

During dissolution of spent fuel, platinum group fission products remain as insoluble residues. Main components being Mo, Tc, Ru, Rh and Pd. The fine particles of these fission products distribute themselves between the aqueous and organic phases, leading to poor decontamination of the final product U and Pu.

It is difficult to dissolve carbide fuel and the by-product, soluble organic acids viz. oxalic acid and maletic acid etc., which interfere in the solvent extraction step are not destroyed. Both these problems are overcome by the development of Electro-Oxidative Dissolution Technique (EODT), which involves the dissolution of the fuel in 11. 5 M nitric acid containing redox intermediate Ag^{2+}. The Ag^{2+} generated in situ at the anode in the dissolver catalyses the dissolution and at the same time destroys the soluble organic acids to CO_2 and H_2O.

The dissolver is made of titanium where dissolution of fuel is carried out in nitric acid with silver nitrate as catalyst. The titanium dissolver body acts as cathode while platinum

coated stainless steel wire mesh acts as anode. The dissolver solution is clarified in a small diameter high speed centrifuge where almost all the insoluble residues are removed from the solution.

7.4. Solvent extraction

The presence of Pu in high concentration in organic solvent complicates the extraction behaviour due to the formation of a third organic phase which upsets the hydraulic performance of the extraction units. SIMPSEX (SIMulation Program for Solvent EXtraction (Kumar and Kogants, 1996)) code was developed for the analysis and simulation of flowsheets employing TBP as the solvent.The process employs low pressure evaporation and distillation where in both TBP and dodecane can be'recovered and reused.

8. SUMMARY

Normally highly enriched UO_2 fuel is used in a small fast reactor. In FBTR enriched uranium was planned to be replaced by PuO_2 (U 0.24 - Pu 0.76)O_2. Out-of-pile tests indicated non-compatibility of this fuel with sodium. Therefore, fuel of 70% PuC - 30% UC was selected as driver fuel. Fabrication of (U,Pu)C pellets is more expensive and difficult than that of (U,Pu)O_2. Oxides of plutonium and uranium are subjected to vacuum carbothermic reduction and then sintered to form pellets. Fuel SA are made from these pellets and loaded as Mark-I core into FBTR. This fuel has crossed a peak burnup of 40,000 MWd/t without any failure. PIE was done on one SA after 25000 MWd/t burnup. The result confirmed the conclusion drawn by analysis and fuel was cleared for burnup of 50000 WMd/t. The FBTR spent fuel is stored in a dry storage before sending for PIE and reprocessing. KAMINI, uses U^{233} as fuel and acts as a neutron source. Closing the fuel cycle is important for breeder program. High radioactivity and highly refractory nature of spent fuel make reprocessing a very challenging task. FBTR fuel is reprocessed by the well proven purex process. Spent pins are sheared and disolved in electrochemical dissolver. This is followed by solvent extraction.

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MANAGEMENT OF SPENT NUCLEAR FUEL AT THE STATE SCIENTIFIC CENTER OF THE RUSSIAN FEDERATION — INSTITUTE FOR PHYSICS AND POWER ENGINEERING

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Abstract

The report considers principle issues of management and disposition of the spent nuclear materials of research reactors at State Scientific Center of Russian Federation – Institute of Physics and Power Engineering (SSC RF-IPPE). The legal and regulatory basis formed to date in Russia for this kind of activities is presented. Results of research works dealing with spent fuel elements of research reactor AM that have been stored for 37 years are referred to. Based on the experience gained at IPPE with the operation of research reactors, measures are proposed that can ensure safe management and long-term storage of spent nuclear fuel, without planning its reprocessing in the foreseeable future.

1. LEGAL AND REGULATORY DOCUMENTATION FUNDAMENTALS

The development of legal and regulatory documents as a basis for management and disposition of radioactive wastes and spent fuel elements of research nuclear reactors have been rendered a great attention in Russia.

In the period from early 1991 up to now, the following Federal laws have been prepared and passed:

- On the use of nuclear energy (1995);
- On management of radioactive wastes (1996);
- On the protection of population and areas from accidental situations of natural and technology-related character (1994);
- On the protection of natural environment (1991);
- On ecological expert evaluation (1995).

Previously, documents were issued such as "Nuclear Safety Regulations for Research Reactors"; "General rules for safety assurance at research reactors"; "Health physics regulations for radioactive waste management", "Safety rules in nuclear fuel transportation at the national nuclear power system facilities", "Health physics regulations for designing and operation of nuclear reactors for research applications", regulatory documents for nuclear material accounting, control, and physical protection (MPC&A).

The Decree of the Russian Federation (RF) Government of October 11,1997, approved the Rules of organizing the State MPC&A system.

The work for upgrading the facilities' responsibility for the organization of correct and safe operation of research nuclear reactors and the use of radioactive substances (RS) and fissile materials (FM)

The Russian Federation Ministry for Atomic Energy has been charged as the customer for works dealing with the creation and functioning of the state system for accountancy and control of RS and FM.

Special Federal Program "Management of Radioactive Wastes and Spent Nuclear Materials, Their Utilization and Final Disposal for 1996-2005" has been developed and fulfilled in Russia. The Program provides the following:

- Liquidation and conservation of research nuclear reactors, critical facilities and test rigs that are decommissioned;
- Creation of storage for radioactive wastes and spent nuclear materials;
- Creation of facilities for conditioning of radioactive wastes and reconstruction thereof, etc.

SSC RF-IPPE takes part in this program.

A special Federal Program for nuclear and radiation safety in Russia has been under development, IPPE being an active participant of its creation. Taking into account the experience with accounting and control of spent nuclear fuel, radioactive materials and wastes gained by foreign nuclear research centers, Russia is developing its State concept for management of radioactive wastes, with IPPE actively participating as well.

2. BRIEF DESCRIPTION OF SSC RF-IPPE

SSC RF-IPPE named after Academician A. I. Leipunski was founded in 1946 to solve scientific and engineering problems of nuclear power creation and development. The Institute's research and engineering activities include the following trends:

- Sodium-cooled fast-neutron reactors;
- Graphite-and-water reactors;
- Reactors with lead-bismuth coolant for transportation devices and on-site operation;
- Nuclear power facilities operated at space facilities;
- Nuclear-pumped lasers.
-

R&D is carried out in the field of nuclear physics, nuclear reactor physics, low-temperature plasma physics, nuclear pumping for lasers, material irradiation studies, radiochemistry, technologies of reactor materials and fuel elements for reactor cores, thermal physics, hydrodynamics, nuclear and radiation safety, other areas of nuclear science and technology.

The following nuclear facilities and equipment have been constructed under the scientific surveillance of IPPE:

- World's first NPP;
- Fast reactors BR-10, BOR-60, BN-350 and BN-600;
- Two power units of Beloyarsk NPP;
- Bilibino NPP
- A series of transportation and space application nuclear facilities.

Active work has been underway at IPPE for the application of its developments to the national economy. The Institute has at its disposal a large experimental base, including research reactors, critical facilities, charged-particles accelerators, "hot cells", stores for "fresh" and spent nuclear fuel, etc. For the IPPE activities of over 50-year period, a considerable amount of FM and radioactive wastes have been accumulated at the facility. A certain part of spent FM has been dispatched for reprocessing; major part is stored at the IPPE Central storage for spent nuclear fuel and at "cooling" ponds at the research reactors. As to radioactive wastes, advanced facilities are under creation at the Institute for their cementing, pressurizing, and burning.

3. SOME NUCLEAR FACILITIES OF SSC RF-IPPE AND CHARACTERISTICS OF SPENT NUCLEAR FUEL

World's first NPP with AM reactor commissioned on June 27, 1954, is one of the Institute's experimental base objects. Its design-basis thermal power was 30 MW, and electrical one – 5 MW. After the initial 5-year period of operation in the NPP mode, it has been used as an experimental facility for testing various fuel elements and structural materials, testing and improvement of water-chemical modes, and as an irradiation device for radio-isotopic production build-up. The reactor core includes 128 fuel channels (FC), each containing 4 fuel elements (FE). Initially, uranium-molybdenum fuel was used (OM-9), subsequently replaced by uranium dioxide (UO_2) with low enrichment. Characteristics of nuclear fuel are given in Table 1.

In 1956, BR-2 mercury-cooled, 100 kW power reactor was commissioned at IPPE. Principle feasibility of using fast neutron reactors for power generation was shown with that reactor. In 1958, the BR-2 was replaced with BR-5, sodium-cooled, 5 MW. That reactor was used for the development of engineering and technological solutions for their subsequent use in commercial fast neutron reactors designing and construction. Fuel elements of various compositions were used for that purpose , as shown in Table 2 (plutonium dioxide PuO_2, uranium nitride UN, uranium carbide UC). Burn-up levels of 6-7% have been achieved, experience of operating non-sealed fuel elements gained, release of fission products into the coolant and gas plenums of primary circuit has been studied; the system for monitoring of non-sealed FE for delayed neutrons is created; safety system and dynamic monitoring of the reactor facility has been investigated, with a series of neutronic and material irradiation studies accomplished.

In 1973, reconstruction of BR-5 reactor was completed. Its power was upgraded to 10 MW, and it has been termed BR-10 since (see Table 2). To date, reactor BR-10 has been used for the studies of nuclear fuel and materials performance, isotope production, and neutron therapy. Engineering options for upgrading safety of fast-neutron power reactors have been tested and optimized at the facility.

Table 1 Irradiation Parameters of AM reactor SA

No.	Fuel Composition	Enrichment, U-235 (%)	Material of clad*)	Cross-section of the fuel part (mm)	Grams of U-235 in fresh SA	Average burn-up per SA (%)	Storage after irradiation (years)	Number of irradiated SAs in the store
1.	UO_2	3.6	E110	∅ 9.1	670	3.0	13	3
2.	UO_2	4.4	1x18n10t	∅ 14*9	310	3.5	7	36
3.	OM-9	5.0	1x18n10t	∅ 14*9	210	3.0	42	41
4.	OM-9	6.5	1x18n9t	∅ 14*9	270±10	3.0	36	68
5.	UO_2	10.0	1x18n9t	∅ 14*9	325±10	3.0	16	24
6.	UO_2	10.0	1x18n9t	∅ 14*9	310	3.5	9	12

Notes: 1. The number of SAs in the reactor – 128;
2. The number of fuel rods per SA – 4;
3. Fuel composition length – 1700 mm
4. Fuel rod can thickness – (0.15-0.20) mm

*) - According to the Russian classification of steels.

Table 2 Irradiation Parameters of BR (fast-neutron) reactor SA

No.	Fuel Composition	Enrichment U-235 (%)	Material of clad*)	Cross-section of the fuel part (mm)	Grams of U-235 in fresh SA	Average burn-up per SA (%)	Storage after irradiation (years)	Number of irradiated SAs in the store
1.	Pu		oh18n9t	∅ 10*0.3	200	Not calculated	41	77
2.	PuO_2		oh18n9t	∅ 5*0.4	530	4.5	37	180
3.	UO_2	90	oh18n9t	∅ 5*0.4	531	1-2	37	90
4.	UO_2	80	oh18n9t	∅ 5*0.4	475	1-2	37	90
5.	UC	90	oh18n9t	∅ 8.35*0.4	1200	5.2	28	60
6.	UC	90	oh16n15m3b	∅ 8.35*0.4	1200	5.2	28	60
7.	PuO_2		oh16n15m3b	∅ 5*0.4	600	6-12	18	120
8.	UN	90	oh16n15m3b	∅ 8.4*0.4	1330	5-7.7	7	120
9.	OM-9	natural	oh18n9t	∅ 9*0.4	13	Not calculated	41	30

Notes: 1. The number of SAs in the reactor – 128 (except for the Pu core);
2. Fuel composition length for the Pu core – 130 mm; others – 400 mm.

*) - According to the Russian classification of steels.

Experimental transportable power plant TES-3 of 1.5 MW power was developed and commissioned at the Institute in 1961. Fuel elements for it were fabricated from OM-3 alloy with enrichment of uranium-235 of 2.4 to 3.6%, and OM-9 alloy with 36 to 90% enrichment of U-235. Diameter of fuel elements amounted to 12 mm, and the length of fuel composition - ~1m. One fuel subassembly (SA) contained ~1 kg of uranium. Average burn-up reached 3%. The experience gained with the TES-3 development was used in the development of unit-transportable and naval NPPs with two-circuit integral self-regulating water-cooled-water-moderated reactor facilities with natural circulation of coolant in primary and secondary circuits. After the test operation of NPP, TES-3 was decommissioned in 1978.

In 1970-1984, at the IPPE's special test facility was used for testing seven specimens of power facility with thermionic reactor-converter "TOPAZ" and pre-launch tests of two standard facilities. Power generating channels of the "TOPAZ" were developed, fabricated, and undergone in-pile tests at the IPPE. Those FEs were fabricated from uranium dioxide with enrichment of U-235 ranging from 21%to 90%. The fuel elements had diameters of 10 mm, with the length of fuel composition of ~300 mm. The content of uranium per FE was 196g. Average burn-up amounted to 0.4%.

For reactor materials and fuel composition studies, the Institute has a complex of "hot" chambers and "heavy" boxes. Experimental and standard FE, SA, ampoules, blocks, items, and material samples received from research reactors AM, BR-10, BOR-60, and others, as well as from power reactor BN-350 are dismantled and investigated in these chambers and boxes.

4. TECHNOLOGY OF SPENT FUEL STORAGE AT SSC RF-IPPE

As a result of long-term operation of the experimental base nuclear facilities, a large quantity of spent nuclear materials has been accumulated at the Institute. They differ in their types (uranium, plutonium), enrichment of U-235 (ranging from 4% to 90%), burn-up depth, "cooling" period after irradiation, composition of the fuel contained, etc.

Spent fuel elements of research reactors are stored in a special storage designed for temporary storing of spent fuel (Central storage for spent fuel, CSSF). This storage is situated within the Institute's guarded perimeter with the status of protected area, remote from other production facilities. The CCSF belongs to the dry type storage; it has above and underground parts.

The upper part contains the storage life-support systems, such as: power- and water supply, special waste-water disposal, dosimetry control post, fire- and emergency alarm systems, ventilation, heating, and automatic water pumping-down devices. For spent nuclear fuel handling, the storage is equipped with overhead travelling crane of 20/5 t load-carrying capacity, reloading container, and various devices.

Spent fuel is kept in the underground part in special cells. The cells are made from heavy concrete; they form a square grid with the "pitch" of 800 mm. Stainless steel covers of different lengths are installed into the cells, with FE inserted therein.

Each cell is covered with protecting steel plug and rubber-sealed lead. The monolith cover from heavy concrete, protecting plugs, sealing leads, and the underground compartment foundation ensure the decrease of γ-radiation to admissible levels, both inside the storage, and outside, in complete loading of the storage with spent fuel of 6 x 10^7 g eq. radium activity.

Spent nuclear fuel received directly from the research reactors' cores is placed into the "cooling" ponds available at each research reactor, being stored there for 1.5 to 3 years, then dispatched to the CCSF. Therefore, a considerable diminishing of the activity and heat release from the spent FEs is achieved.

For the transportation of spent nuclear fuel from buildings it is handled in, from research reactors, "hot" chambers to the CCSF, the Institute' railway system is used, flat-car with removable protection container, hoisting devices, and diesel locomotive are used. Spent fuel is reloaded with remote control using a special reloading container.

The system of nuclear material accounting and control (MC&A) now in force at the Institute provides all necessary information on the content of each cell in the storage and cooling ponds of research reactors, as well as on each FE:

- Passport is made for each fuel element with reactor type indicated, as well as the manufacturer's serial number design specifications, mass of nuclear material before loading into reactor, and its calculated mass after unloading from the reactor, dates of loading and unloading, burn-up, number of the clad in which the FE is reloaded, date of dispatching for storage.
- Spent FE is transferred for storage after executing special documents signed by the Institute's managers;
- Special documents of CCSF record the following data for each cell of the storage: number, FE identities and quantity; passport numbers of the FE, and requirements for the FE transfer for storage, weight of uranium or plutonium; date of loading;
- After the clad with spent FEs is loaded into the cell, it is covered with sealing lead, with the date of loading and reactor type indicated on the lead;
- The FE presence is verified periodically by visual inspection of the storage cells;
- Concentrations of α-, β-active aerosols and γ-radiation dose rates are checked at the CCSF periodically.

The question of further fate of the spent FEs stored at CCSF has been regarded very attentively at IPPE. The reasons for this keen interest are:

- The CCSF has been filled by 80%. The remaining free cells are not sufficient for the placement of spent FE stored at the cooling ponds at the research reactors AM, BR-10, and at the "hot" laboratory, and the fuel elements available at the AM and BR-10 at present.
- When designing and creating the research reactors, spent fuel problems were not considered, and as a result, technologies for spent nuclear fuel

regeneration are not available for all cases. The processing of spent fuel elements with low level of nuclear material in small batches proves to be economically unprofitable. The standard and experimental fuel elements stored at CCSF have large variations in both fuel composition and design. For example, the AM fuel compositions consist of both uranium dioxide with U-235 enrichment ranging from 2% to 10%, and the OM-9 alloy with enrichment of U-235 from 4.4% to 7.0%. As for the BR-10 fuel compositions, they include both plutonium dioxide, and uranium dioxide with enrichment of U-235 of 90%, and uranium nitride with that of 90%. Spent fuel elements received after investigations from the "hot" laboratory also have widely ranging fuel compositions with enrichment of U-235 of 4% to 90%.

- The storage for spent fuel now available was planned to be used for a temporary storage of such fuel, and in principle, for the time being it meets the requirements. However, with the issue of new regulatory documents that make these requirements more strict, the reconstruction and modernization of this storage has to be scheduled for the nearest time.

It should be pointed out that the current situation with spent nuclear fuel of research reactors at the Institute proves to be typical also for other facilities that have research reactors.

In the Declaration of the Moscow Summit of 8 leading nuclear states convened on the 19-th of April of 1996, it is stated (article 18) that "state bodies are obliged to ensure safe management of radioactive wastes, as well as the development of rules for their appropriate processing, storaging, and final disposal".

5. ANALYSIS OF SPENT NUCLEAR FUEL DURING LONG-TERM STORAGE

For the 43-year operation time of the AM reactor of the World's first NPP (1954-1997), a large number of fuel subassemblies has been irradiated in its core. They are of the AM type (annular fuel elements with internal heat removal) with various kinds of fuel. Within the framework of Federal Program "Management of radioactive wastes and spent nuclear materials", investigations on fuel elements of one of the SAs used in the AM have been carried out at the Institute. The SA has been under irradiation since May, 1955 through July, 1958 (309 effective days) then was stored in the "cooling" pond at the reactor for 37 years, initially in water environment 25 years, then in dry storage 12 years. The following procedures have been carried out:

- Measurement of external diameter of FEs;
- External inspection;
- γ-spectrometry measurements;
- metallography studies;
- electron microscopy of FE clad surfaces;
- X-ray-spectral analysis;
- Studies of mechanical properties of the clad surfaces;
- Radiochemical investigations of FE clad material;
- Calculations on radionuclides and elements composition of the fuel.

As a result of the studies, it has been established that all 4 fuel elements of the SA are in a satisfactory state; no loss of tightness has taken place during the storage period. No defects have been found on the surface of FE, such as cracks, blisters, corrosion pits, dents, etc. No concentrating of the chemical elements capable of inducing the corrosion damage to FE surface takes place during storage (chlorine, iodine, fluoride).

The FE clad material has high mechanical characteristics. At ambient temperature, the limit of strength amounts to 820 to 1100 MPa, fluidity limit – 610 to 930 MPa; relative elongation – 4 to 38%. Similar mechanical properties were inherent with the material of AM reactor FE clad that was studied immediately after the irradiation, i.e., the storage did not affect the FE mechanical characteristics.

The activity of FE clad material is mostly due to radionuclides: Co-60, Cs-137, Sr-90, cobalt-60 being uniformly distributed over the clad thickness, whereas strontium-90 and Cs-137 are mostly accumulated within the surface layer (10-20 μm thick) of the external and internal clads that contacted the fuel composition. Cobalt-60 is the main radionuclide determining the steel activity; its activity after 38-year storage is 10 times as high as that of Cs-137. Activity of fuel element on the date of γ-spectrometry research (September 1996) is $7.5 \cdot 10^{11}$ Bq (20.3 Ci); it is determined mostly by cesium-137. The coefficient of non-uniformity of U-235 burn-up is equal to 1.23, which coincides with the calculated one, equal to 1.25. Thus, no migration of Cs-137 in long-term storage takes place.

The calculations of radionuclide and element composition of spent fuel for various periods of fuel element storage, up to 100 years, have shown that total activity of spent fuel is mainly contributed by long-lived nuclides-fission products. Total activity of the fuel composition on the date of unloading from reactor amounted to 6 · 105 Ci, and in 100 years it will be equal to 47 Ci, i.e., will be diminished by approximately 12 000 times.

6. CONCLUSIONS

The following should be pointed out in conclusion:

- Russia has the legal and regulatory basis for management of spent fuel from research reactors;
- The current situation at the SSC RF-IPPE with spent fuel proves to be typical for other facilities of Russia as well - considerable amounts of spent nuclear fuel have been accumulated at facilities with research reactors.
- Special Federal program "Management of radioactive wastes and spent materials ... for 1995-2000" includes the activities for the improvement of the situation with storage of spent nuclear fuel from research reactors;
- It is necessary to accomplish a very careful inventory taking of spent nuclear fuel in Russia, so that to decide what fuel is to be dispatched for reprocessing, and what part is to be dispatched for final disposal. For these purposes, state storage facility for spent fuel of research reactors must be created.
- For the storage of spent nuclear fuel for which no acceptable methods of reprocessing exist, and there is no feasible way to return this fuel to the state it had

originated from, a special regional storage should be created, with proportional participation of the states that have accumulated such fuel;

- Research works carried out at IPPE give grounds to conclude that long-term storage of fuel elements according to the technology accepted at the Institute has not resulted in any substantial damages of the FEs.

- Centralized dry storage is the preferred method for long-term spent fuel management, especially of aluminium clad fuel. The IPPE has 30 years of successful experience with this low-cost technology

List of legal and regulatory documents now in force in the Russian nuclear power system

1. Federal Law "On the Protection of Natural Environment". December 19, 1991.
2. Federal Law "On the Protection of Population and Areas against Emergencies of Natural and Technology-related Character". November 11, 1994.
3. Federal Law "On the Use of Nuclear Energy" November 21, 1995.
4. Federal Law "On Ecological Expert Evaluation" November 23, 1995.
5. Federal Law "On Management of Radioactive Wastes", November 24, 1995.
6. Federal Law "On Radiation Safety for Population" January 9, 1996
7. The concept of the State System for nuclear materials accounting and control (MC&A). October 14, 1996.
8. Regulations for the organization of state system for accounting and control of radioactive substances and radioactive wastes. October 11, 1997.
9. Safety regulations for nuclear fuel storage and transportation at nuclear power facilities. PNAEG-14-029-91. July 1, 1992.
10. General regulations for safety assurance at research reactors. December 30, 1994.
11. Rules of nuclear safety at research reactors. December 19, 1975.
12. Health physics rules for handling and management of radioactive wastes. October 1, 1985.
13. Health physics rules for designing and operation of nuclear reacors for research applications. 1974.

TREATMENT OF FUELS FROM DEVELOPMENT PROGRAMMES AND RESEARCH REACTORS IN THE UNITED KINGDOM

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Abstract

Within the United Kingdom there is significant experience in the chemical treatment of research reactor fuels. The UK policy of reprocessing has promoted the re-use of recovered fuel and has resulted in a small number of waste streams for future encapsulation in standard forms for disposal.

1. INTRODUCTION

During the late 1950's the UK Government set up a number of Nuclear Research Facilities at locations throughout the UK, under the Atomic Energy Act. The task of these facilities was the development of an understanding of the Nuclear Fuel Cycle for the civil application of nuclear energy in the production of Electricity. Many research reactors were built over a short period. On sites operated by the United Kingdom Atomic Energy Authority a total of 28 Research and Test Reactors were built with a further eight research reactors built on other sites in the UK.

As part of the UK's nuclear research policy there was significant effort from the start of the programme to investigate methods for the treatment of the diverse fuel types being proposed for commercial power stations, for research reactors and for the Fast Reactor.

This effort to develop treatment methods was concentrated at two UKAEA sites, Windscale, where the focus was on the treatment of power reactor fuel types and at Dounreay where the focus was on the MTR fuel cycle and the Fast Reactor Fuel Cycle. The successful development of the commercial gas cooled, reactor systems and their fuel production and treatment processes at the UKAEA Windscale site led to the formation of BNFL in 1971, when the main production units were separated from the UKAEA. BNFL operates the large scale power reactor fuel reprocessing plant at Sellafield. At Dounreay, the UKAEA concentrated their efforts on the development of the Fast Reactor Fuel Cycle and the maintenance of the UK's MTR fuel cycle.

2. DOUNREAY HISTORY

The construction of a series of facilities at Dounreay, in the North of Scotland, commenced with the identification of Dounreay as the site for the construction of the first UK Fast Reactor. The Dounreay Fast Reactor was constructed with the necessary support facilities which included a fuel fabrication plant, a uranium recovery facility and a reprocessing plant. To support the recovery and reprocessing facilities, waste treatment facilities and analytical facilities were also constructed.

Within the same timeframe, the UK embarked upon a programme of Material Testing and Research Reactor System Development facility installations, within the UK. At Dounreay a Dido type MTR was installed, with a MTR fuel fabrication facility and a plant for the reprocessing of spent MTR fuel to service the complete UK MTR capacity. With the installed facilities, during the 1960's and the early 1970's the UKAEA operated two separate fuel cycles on the Dounreay site one for the Dounreay Fast Reactor and one for the Dounreay MTR.

Following the construction and successful operation of the fabrication and reprocessing facilities for the MTR cycle, it was concluded that there was capacity in excess of the UK requirements and from 1962 the services of these facilities were offered to countries outside the UK.

In the early 1970's the decision was taken in the UK to install a Prototype Fast Reactor at Dounreay following the successful operation of the DFR. The new reactor fuel design was based on ceramic grade mixed oxide, with fabrication undertaken by BNFL due to their experience with AGR fuel. The reprocessing of the irradiated fuel however, would be undertaken at Dounreay, in the same facility as the DFR fuel, but with significant modifications due to the new fuel type. The strip out of the old facilities commenced in the mid 70's with the modifications completed to allow the first irradiated fuel from the Prototype Fast Reactor to be reprocessed in 1980.

Throughout the development of the different reactor systems in the UK, there has been an appreciation of the need to recover unburned fuel for re-use and to separate the fission products. Many laboratory scale tests have been undertaken at Harwell and Dounreay to allow the installation of versatile reprocessing facilities. At present, on the Dounreay site there are two installed reprocessing facilities of different scales, with a third unit currently planned.

1. The mixed Oxide Fuel reprocessing plant capable of processing up to 6 tonnes of Heavy Metal per year. This plant has a separate dissolution facility for unirradiated fuel.

2. The Materials Testing Reactor Fuel reprocessing plant capable of processing 1200 elements per year. This plant is now shut down and will be decommissioned.

3. A specialised small-scale reprocessing plant is currently planned for the reprocessing of fuels not suitable for reprocessing in the main plant This plant may be installed in an existing shielded facility but will only be operated after licensing. It is expected to have a capacity of up to 500 kg metal per year (depending on fuel type)

3. FUEL TREATMENT EXPERIENCE

Throughout the operation of the Dounreay Fuel plants there has been a lot of experience developed in terms of reprocessing flowsheets the main types of fuel already processed at Dounreay are listed below.

3.1. Molybdenum stabilised metallic enriched uranium fuel

This was the fuel from the Dounreay Fast Reactor. More than 8.5 tonnes of this fuel was reprocessed and the uranium reused, either in DFR fuel or MTR fuel. The fission product waste stream from this processing is currently in storage. The plant was reconfigured to permit

oxide fuel reprocessing, but the dissolver to permit reprocessing of metallic fuel is still available although significant work would be required to bring it up to modern standards.

3.2. MTR fuel reprocessing

Nearly 13,000 MTR fuel elements have been reprocessed in the Dounreay MTR Fuel Reprocessing Plant. The bulk of these elements have been uranium aluminium alloy, but there have been some uranium aluminium dispersed fuels reprocessed. The complete dissolution of the MTR fuel leads to a single waste stream for storage and conversion to a waste form suitable for disposal. A high proportion of the recovered uranium was re-used in MTR fuel. Some Dido/Pluto fuel was on its fourth cycle when the reactors were shut-down in 1990.

3.3. Mixed oxide fuel reprocessing

The mixed oxide fuel reprocessing plant is set up to allow the reprocessing of any combination of uranium and plutonium enrichment. The plant has been used to date for the reprocessing of irradiated PFR fuel and some fabrication residues. To date 35 tonnes of these fuels have been reprocessed. Plutonium recovered was recycled into new fuel for the reactor.

3.4. TRIGA fuel reprocessing

A new small scale reprocessing facility has been designed for installation at Dounreay to permit the reprocessing of TRIGA (uranium zirconium hydride) fuel. As this is a uranium feedstock with relatively difficult dissolution characteristics that is unsuitable for long term storage, the plant is being designed to treat the fuel and generate a waste stream for incorporation in a standard waste form for disposal. The recovered uranium could potentially be re-used in test reactors.

3.5. Carbide fuel reprocessing

As part of the fast reactor fuel development programme there were mixed carbide, carbonitride and nitride fuels core and uranium carbide breeder assemblies produced. UKAEA have treatment processes for the unirradiated carbide under design review prior to construction, with a view to extending their application to irradiated fuel. Small amounts of irradiated mixed carbide fuel have already been processed in small scale facilities.

3.6. Thorium fuel reprocessing

UKAEA have extensive experience in the recovery of uranium from thorium/graphite matrix fuel that is unirradiated or lightly irradiated. There has also been small scale development of dissolution and solvent extraction processes for small quantities of thorium metal and thorium oxide fuels containing uranium and plutonium.

3.7. Silicide fuel reprocessing

The full scale reprocessing of two uranium silicide fuel elements was reported in 1996 at the RERTR Conference in Korea and at the 1997 RRFM Conference in Bruges. UKAEA have utilised their experience from unirradiated silicide scrap recovery and their experience in MTR fuel reprocessing to demonstrate how silicide fuels can be processed to separate the fission products from the uranium. The uranium could then be either blended down for use in

power reactor fuel or mixed with higher enriched uranium for re-use at 20% enrichment in MTR fuel. Silicide reprocessing on a large scale requires a fuel reprocessing plant equipped with a solids removal system to eliminate the insoluble silicide component before the solvent extraction process.

3.8. SUR fuel recovery

SUR fuel is a uranium dispersion in polyethelene which was used in some German university reactors. The equipment for treatment of this fuel has been developed through laboratory scale trials and was designed to give the universities an alternative to storage. The equipment has not yet been installed.

4. UK CURRENT SITUATION

The current situation in the UK is that only one research reactor remains in operation, at Imperial College in London. All of the UK's uranium aluminium fuels have been reprocessed, with the exception of the Jason fuel. The fission products from the MTR fuel have been separated for encapsulation in a cement matrix. Other development fuels are either destined for recovery in the facilities at Dounreay or are of a low activity and fissile content which may allow them to be disposed of directly. The UK policy of reprocessing has allowed the diverse assortment of fuel types produced over the last 35 years to be treated to re-use the fissile material and to separate the waste products into standard streams for future conditioning for disposal.

This policy of reprocessing has allowed the UK to concentrate the requirements for future waste disposal facilities into a small number of waste streams, which are similar for both power reactor and research reactor systems.

5. UK POLICY ON REPROCESSING OF OVERSEAS MATERIAL

The UK policy on the import of radioactive material for recovery requires there to be a reusable (valuable) product and for any separated waste to be returned to the country of origin as soon as possible and within a period of 25 years. This return of waste requires an Intergovernmental exchange of letters to be in place before any fuel delivery. This Intergovernmental exchange of letters states that neither government will take regulatory or other action that will prevent the return of the wastes. Hence, any contract for overseas fuel reprocessing in the UK will require a firm commitment on the return of waste. An example of the contractual terms for waste return are included in Appendix 1. The facilities at Dounreay are no longer available for the reprocessing of overseas fuels and we plan to do no more than complete their currently identified commitments before being decommissioned.

6. TRANSPORTATION

Any spent fuel treatment will require the fuel to be transported from its current site to the treatment site. Followed within the stated period, by the movement of the treated residue to the disposal or storage facility.

Radioactive transport is, in spite of its impeccable safety record, an emotive subject. It is the weak link in any fuel treatment scenario as it can be readily targeted by pressure groups leading to a ratcheting of regulations. The recent returns of vitrified wastes between France

and Germany were a demonstration of this. The cost to the German Government for carrying out its intergovernmental obligations were politically and financially damaging.

More needs to be done to explain to the public the intrinsic safety of these transport operations and the storage conditions, otherwise there will be no nuclear transports, without high costs and high profile escorts.

7. SUMMARY

There is significant experience in the reprocessing of many types of Research Reactor fuels at Dounreay. The UK's policy of reprocessing has resulted in very few unanswered fuel treatment questions and has led to a minimum number of waste streams that will be conditioned to a form for future disposal.

FUEL ELEMENTS MANUFACTURED BY UKAEA 1957-1997

REACTOR	FUEL ELEMENTS PRODUCED
HIFAR (Australia)	1057
FRJ-2 (Germany)	1628
Dido/Pluto (UK)	5483
Herald (UK)	344
Herald (Chile)	40
Safari (S Africa)	60
DR-3 (Denmark)	500
Universities (UK)	126
Apsara (India)	40
RV-1 (Venezuela)	20
HFR (Netherlands)	176
BR2 (Belgium)	6
TOTAL	**9480**

ELEMENTS REPROCESSED 1958-1996

COUNTRY	NO ELEMENTS	URANIUM (kg)
United Kingdom	9306	1438.3
Belgium	240	59.5
Spain	6	10.8
Denmark	950	105.6
France	289	98.1
Australia	264	32.2
India	83	14.0
Germany	918	135.4
South Africa	216	29.5
Greece	39	29.6
Sweden	168	24.8
Japan	410	82.7
TOTAL	**12889**	**2060.5**

OVERVIEW OF THE US SPENT NUCLEAR FUEL PROGRAM

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Abstract

This report, Overview of the United States Spent Nuclear Fuel Program, December, 1997, summarizes the U.S. strategy for interim management and ultimate disposition of spent nuclear fuel from research and test reactors. The key elements of this strategy include consolidation of this spent nuclear fuel at three sites, preparation of the fuel for geologic disposal in road-ready packages, and low-cost dry interim storage until the planned geologic repository is opened. The U.S. has a number of research programs in place that are intended to provide data and technologies to support both characterization and disposition of the fuel.

1. INTRODUCTION

As a result of the end of the Cold War, the mission of the United States Department of Energy (DOE) has shifted from an emphasis on nuclear weapons development and production to an emphasis on the safe management and disposal of excess nuclear materials including spent nuclear fuel from both production and research reactors.

Within the U.S., there are two groups managing spent nuclear fuel. Commercial nuclear power plants are managing their spent nuclear fuel at the individual reactor sites until the planned repository is opened. All other spent nuclear fuel, including research reactors, university reactors, naval reactors, and legacy material from the cold war is managed by DOE. DOE's mission is to safely and efficiently manage its spent nuclear fuel and prepare it for disposal. This mission involves correcting existing vulnerabilities in spent fuel storage; moving spent fuel from wet basins to dry storage; processing at-risk spent fuel; and preparing spent fuel in "road-ready" condition for repository disposal. The term "road-ready" means that the fuel canisters are transportable under current regulations and will not have to be reopened prior to final disposition.

Most of DOE's spent nuclear fuel is stored in underwater basins (wet storage). Many of these basins are outdated, and spent fuel is to be removed and transferred to more modern basins or to new dry storage facilities.

DOE's current inventory comes from:

- DOE test and materials production reactors
- Non-DOE U.S. government reactors
- U.S. university research reactors
- Foreign research reactors
- U.S. Navy propulsion reactors

DOE currently manages about 2,500 metric tons of heavy metal (MTHM) of spent nuclear fuel. This is a small percentage of the total amount of spent nuclear fuel being stored in the United States. The commercial nuclear industry is currently storing about 35,000 MTHM of spent nuclear fuel at its reactor sites.

DOE expects that up to an additional 100 MTHM of spent nuclear fuel will be received into inventory (primarily from domestic and foreign research reactors and the Naval Reactors Program) over the next 40 years. This is in comparison to about 70,000 additional MTHM expected to be generated by the commercial nuclear industry.

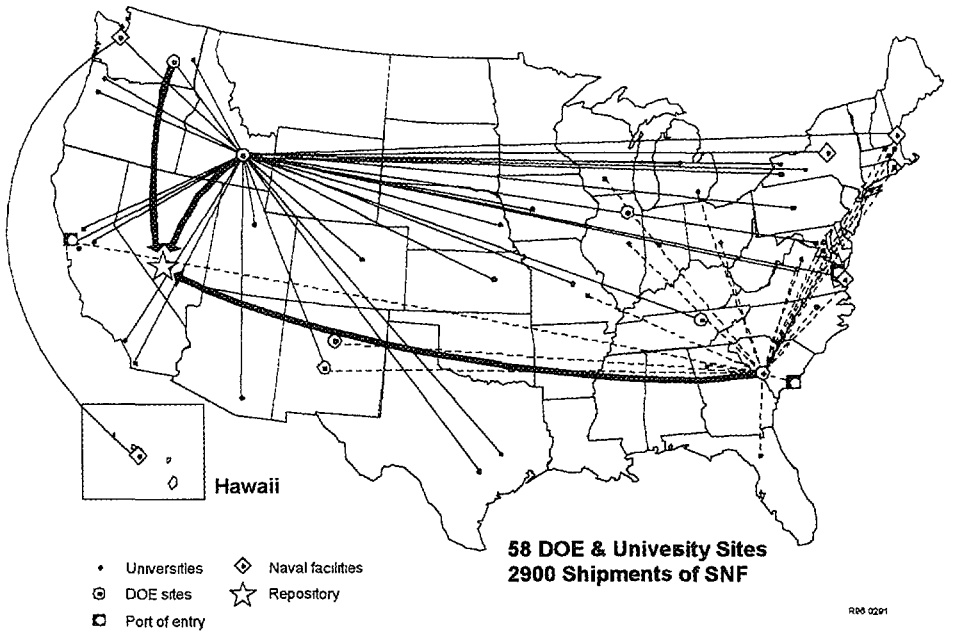


Figure 1: DOE's Challenge for Spent Nuclear Fuel

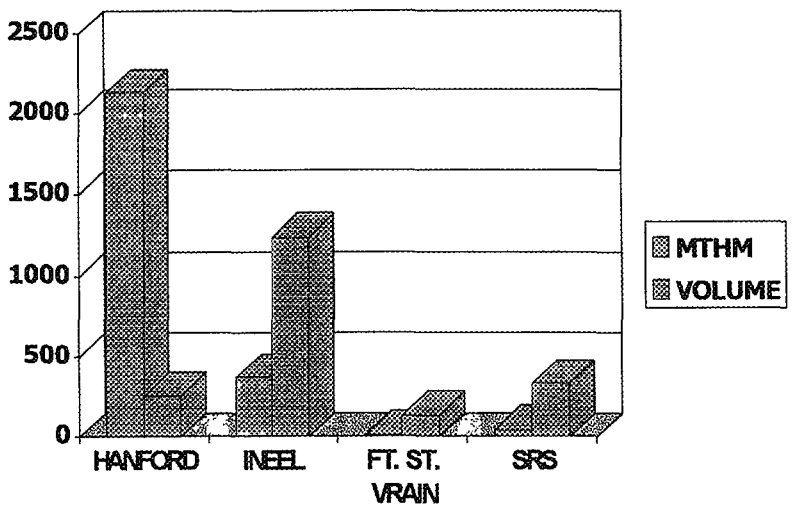
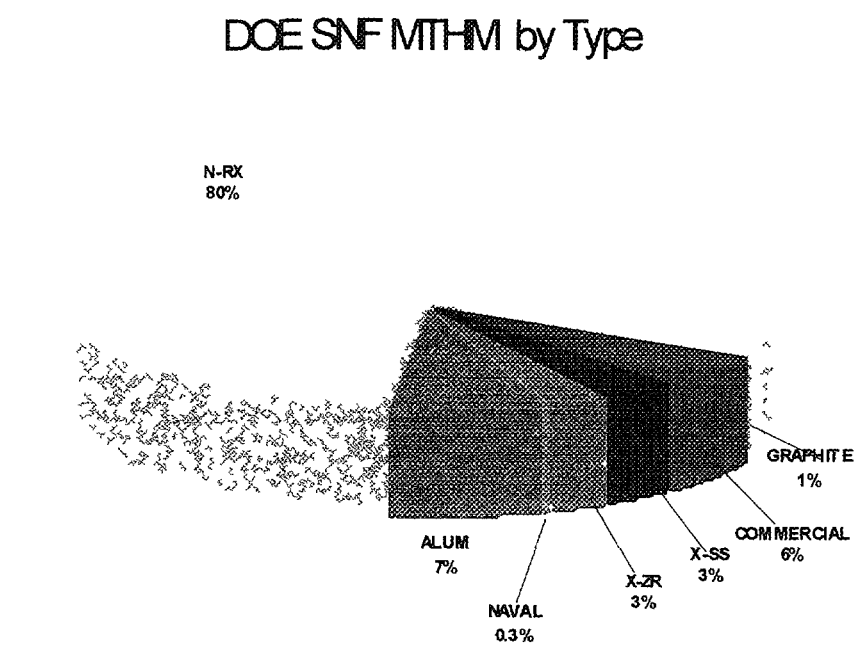


Figure 2: Quantities of Spent Nuclear Fuel After Consolidation



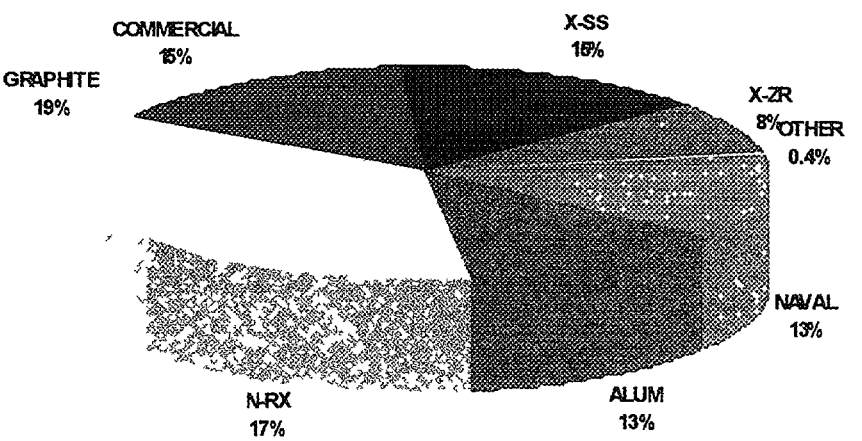


Figure 3 Heavy Metal and Volume of DOE Spent Nuclear Fuel

In 1995, DOE completed a complex-wide environmental impact analysis that resulted in spent fuel being sent to one of four principal DOE sites for interim storage (up to 40 years) prior to shipment to a repository This "regionalization" by fuel type will allow for economies of scale yet minimize unnecessary transportation

The "regionalized" approach to managing DOE's spent nuclear fuel will be as follows

- Aluminum-based spent fuel will be consolidated and stored at the Savannah River Site At-risk fuel (fuel posing health and safety risks under current storage conditions) will be processed at the Savannah River Site

- Non-aluminum based spent nuclear fuel from domestic and foreign research reactors will be consolidated and stored and the Idaho National Engineering and Environmental Laboratory.
- Hanford production reactor spent fuel will remain at Hanford.
- Naval fuel will be shipped to the Idaho National Engineering and Environmental Laboratory for examination and storage.
- Fort St. Vrain fuel will remain at the former Fort St. Vrain reactor site.

Figure 1 shows locations of all spent nuclear fuel currently or planned for management by DOE along with the four interim storage sites, Fort St. Vrain, Hanford, Idaho National Engineering and Environmental Laboratory, and Savannah River. Figure 2 shows the quantities of spent nuclear fuel that will be managed at these interim storage sites. Figure 3 shows the volume of the various types of spent nuclear fuel.

2. THE NATIONAL SNF PROGRAM

DOE formed an office of spent fuel management (EM-67) in 1993 to develop a strategy for interim management of all non-commercial spent nuclear fuel and provide for the final disposition. Initially the program was focused on development of policy and procedures for interim storage. Today, the program, with a strong component deployed in the field, has transitioned to an implementation program. The U.S. has an aggressive plan to place all DOE managed spent nuclear fuel into "road-ready" dry interim storage.

The National Program has a strong headquarters component that provides policy direction and coordination with other DOE organizations. The field program primarily directs the technical effort to ensure that DOE SNF is in the repository license application and is lead by the Idaho National Engineering and Environmental Laboratory The program is a multi-laboratory program that draws on the expertise of many national labs

Early on, it became apparent that an accurate portrayal of the SNF inventory was necessary to develop the path towards ultimate disposition. As such, DOE developed a national spent fuel database. The purpose of the database is to provide DOE with the information necessary for national-level planning in the management of all SNF that is currently or may come under the control or possession of DOE-EM through 2035. SNF is defined as nuclear fuel that has been permanently withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing. Management of SNF includes those activities associated with storage, transportation, conditioning, and disposal. The database supports those organizations with responsibility for managing the SNF by collecting information into one source, providing data in a uniform standard format, and providing data that have been validated.

A comprehensive assessment of the radiological impacts due to the disposal of DOE-owned SNF is an essential component to successful submittal of the repository license application, scheduled for 2002. The National Program has developed a multi-year plan to understand expected dose rates at the accessible environment due to DOE SNF and to understand the importance of various parameters (e.g. radionuclide release rate) on repository performance.

Another essential component for successful DOE SNF disposition will be the analysis of canister and waste package criticality. It must be demonstrated that criticality events are no more probable for DOE SNF packages, both in package and in the surrounding geology, than for commercial SNF. Criticality will likely prove to be one of the most challenging aspects of DOE SNF dispositioning because of the many geometries, concentrations, enrichments, etc. Waste package fissile material limits may be established to provide general site guidance and allow the development of a repository licensing application. Each site will need to perform criticality analysis for actual package loading to demonstrate compliance with repository acceptance criteria and transportation requirements.

3. ULTIMATE DISPOSITION

A planned geologic high-level waste repository will not be available to begin accepting DOE spent fuel for approximately 20 years. DOE and the U.S. Nuclear Regulatory Commission have not yet developed waste acceptance criteria, which will provide critical information for establishing a performance goal for the packaging. DOE is taking advantage of this time to develop a number of treatments for those spent nuclear fuels that cannot be disposed of directly. At this point it is too early to say how practical a particular technology will eventually prove to be.

The U.S. stopped reprocessing of spent nuclear fuel for fissile material production purposes in 1992. Since that time, only limited reprocessing has been performed on spent nuclear fuel that presented environmental, safety or health risks. In order to safely manage DOE's spent fuel prior to geologic disposal, DOE will continue to process certain "at-risk" spent nuclear fuels, develop new dry storage facilities, and package its spent nuclear fuel so that it is suitable for direct disposal in a repository. DOE is currently evaluating other non-processing treatments for HEU aluminum based fuels. DOE's current strategy is to place all spent nuclear fuel in dry storage and in "road-ready" condition for eventual transport to, and ultimate disposal in a geologic repository. While the repository waste package designs are still being evaluated, DOE is currently planning on co-disposing non-commercial SNF with high level waste glass in a single waste package (see Figure 4).

4. TECHNICAL CHALLENGES

DOE is actively pursuing resolution to many issues that face the custodians of the SNF at the interim storage sites. Whereas DOE's inventory contains many exotic fuels from research and production reactors, development of repository acceptance criteria is challenging. Much of DOE's SNF inventory is disrupted (See Figure 5) and may require special packaging before shipment to the planned repository. Standardized containers are being developed that will allow for transport of disrupted or failed fuel and address long-term waste isolation issues. The use of standardized containers across the DOE complex will reduce the number of waste packages for analyses and thus lower costs for both interim storage and ultimate disposition.

Ongoing and planned research is generally directed at issues related to long-term interim storage with subsequent transportation to a repository. The following sections briefly describe active and planned research areas.

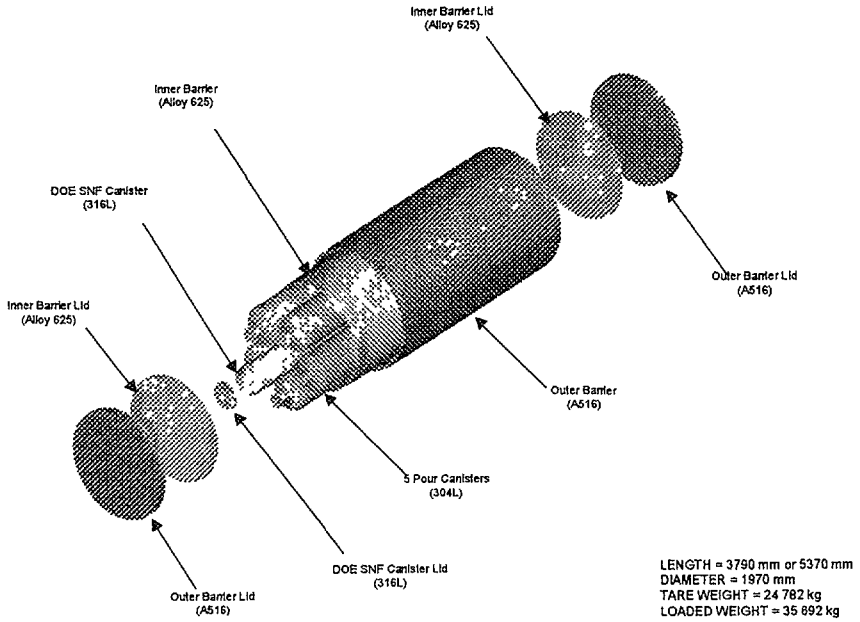
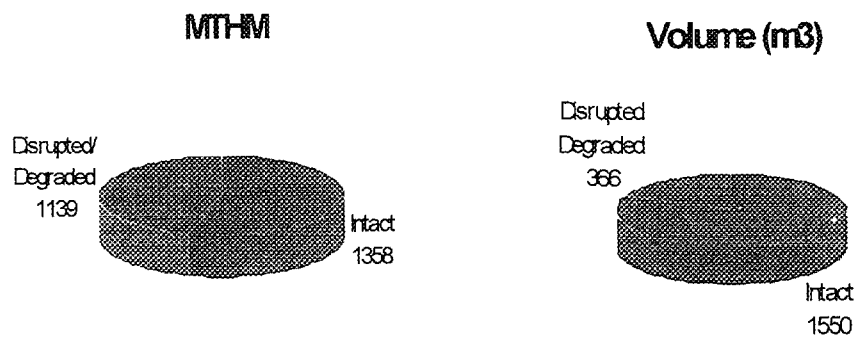


Figure 4: Co-Disposal Concept for Waste Emplacement



Examples

Reactor (2100 MTHM) is about 50% degraded

MI is disrupted (82 MTHM)

here are small quantities of virtually every type that have undergone PIE
here are small quantities of FERMI, TRIGA, and others that were damaged during reactor operations

All data are estimates.

Figure 5 Condition of DOE's Spent Nuclear Fuel

4.1. Release Rate Testing

DOE is examining historical data to determine if data exists to predict how SNF will degrade in the planned repository A key element of the degradation model is the expected release and subsequent transport of radionuclides within the repository Where data are needed, DOE is obtaining samples of the SNF types and performing flow-though and drip tests to develop release rate data These tests are expensive and time consuming Thus, it is important to review available literature and group similar fuels together based on fuel compound or other parameters to reduce the number of tests

4.2. Non-Destructive Assay

DOE has generated and accumulated a broad variety of SNF as a result of numerous production, testing, development, and research activities over the years. The plan for disposition of most of the DOE SNF is to direct dispose of it in the first geologic repository. In order to place this material in a repository, it is essential that key data be qualified per the acceptance criteria at the repository. The current data for most of the DOE SNF are not readily qualifiable to these criteria and in many cases it may not be possible to qualify the data. Therefore, DOE is examining promising non-destructive assay techniques that could be used to obtain new data that meets the acceptance requirements. If measurements can not readily be made with sufficient accuracy, then it may be possible to use NDA to corroborate the existing data and thereby qualify it.

One NDA technology under development is capable of making blind measurements independent of changes in geometry. This system is expected to provide fissile as well as burnup information allowing certification to the repository acceptance criteria without destructive analysis.

4.3. Alternative Treatment

DOE stores over 150 different types of spent fuels, many of them unique to DOE, at its weapons production sites and national laboratories. The Department's ability to direct dispose many of these fuels in a geologic repository without further processing is now being examined. Some fuels may not be suitable for direct disposal because they could pose a criticality risk, are chemically reactive or prone to corrosion. A few DOE spent fuels may exhibit hazardous characteristics as defined under the Resource Conservation and Recovery Act (RCRA) and the regulations that implement it (40 CFR 261 et seq.). The preliminary repository waste acceptance criteria prohibits RCRA-hazardous or mixed wastes. Programmatic cost and scheduling issues must also be considered. Qualifying the variety of DOE spent fuels for repository disposal could be a very time consuming and expensive process. Reducing the volume of materials to be disposed would greatly reduce the Department's spent fuel cask purchasing costs and transportation.

At the present stage of development, the electrometallurgical treatment process appears to be a promising option for the metallic sodium bonded spent fuels, but research will need to be completed to make this determination on a factual, less speculative basis. Several other treatment processes are also being examined for this purpose. For the aluminum-based fuels, the Savannah River site is considering a melt-and-dilute technology. This new treatment technology will produce a low-enriched homogeneous waste form that will be acceptable for repository emplacement. Savannah River is also considering direct disposal and reprocessing as a contingency to be used if no acceptable new treatment and/or packaging technology is ready to be implemented by the year 2000.

4.4. Drying

The transition to interim dry storage and ultimate disposition in the planned repository requires that DOE understand the quantities of both bound and free water that may be packaged with the SNF. Residual water in storage or disposal containers may accelerate corrosion of the SNF resulting in early release of radionuclides. Research in drying has been

concerned with quantifying the water that remains with spent fuel after removal from wet storage. Both heated and cold-vacuum technologies are being pursued.

4.5. Chemical Reactivity

It is known that some of the DOE-SNF has degraded cladding that has interacted with basin storage water to create compounds that could potentially be chemically reactive under certain conditions. Current regulations require that packages destined for the national repository "shall not contain explosive or pyrophoric materials or chemically reactive materials in an amount that could compromise the ability of the underground facility to contribute to waste isolation or the ability of the geologic repository to satisfy the performance objectives". This requirement does not preclude the existence of these materials, but states that the amount of material present will not result in conditions degrading performance of the repository.

The overall strategy for addressing this issue is:

- Assess existing literature data and established computer analysis capability;
- Compare available data and tools to program requirements;
- Obtain and implement tools to support analysis capability, including implementing appropriate quality controls to allow evaluation and acceptance by the NRC;
- Define required laboratory studies to augment information, and incorporate the data resulting from those studies into the analysis capability;
- Establish working interfaces within DOE and the interim storage site programs to assure transfer of data and integrated analysis and application of results.
- Identify and analyze scenarios for the placement of SNF in the repository.

5. SUMMARY

Management of a diverse inventory of exotic research reactor SNF has been successfully performed in the U.S. Paramount to the success of the program has been an accurate SNF inventory, and a well-defined strategy for ultimate disposition. During the next decade, the U.S. will have made a concerted effort to prepare for the ultimate geologic disposal of DOE-owned spent nuclear fuel. As a result, DOE, in close consultation with its stakeholders, will have:

- Resolved or eliminated spent nuclear fuel storage vulnerabilities identified in existing plans of action;
- Constructed any needed dry handling facilities and dry storage facilities and moved most of its spent nuclear fuel to dry storage;
- Begun preparing the spent nuclear fuel for disposal in a geologic repository;
- Processed all the degraded spent nuclear fuel and targets approved for such processing;
- Successfully safeguarded high-enriched uranium from foreign research reactors that could be used to manufacture nuclear weapons.

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