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Management and Storage of Research Reactor Spent Nuclear Fuel

Proceedings of a Technical Meeting held in Thurso, United Kingdom, 19–22 October 2009

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MANAGEMENT AND STORAGE OF RESEARCH REACTOR SPENT NUCLEAR FUEL

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MANAGEMENT AND STORAGE OF RESEARCH REACTOR SPENT NUCLEAR FUEL

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FOREWORD

Responsible use of nuclear technology requires that in addition to safety, security and environmental protection, credible solutions be developed for addressing the management of research reactor fuel throughout the full fuel cycle, including its disposition after removal from the reactor core.

It is clear that the end point of the research reactor fuel cycle is associated with the development of a geological repository to be used for disposition of the spent fuel assembly after it is properly conditioned or for disposition of the waste resulting from its reprocessing. Developing a geological repository for spent nuclear fuel and high level waste is not an easy task. Only a few countries have made great progress in the implementation of geological repositories, including Finland, France, Sweden and the United States of America. The technology and costs involved in developing a geological repository and maintaining it for hundreds of years make it prohibitive for most countries, especially those countries with only one or two research reactors and no nuclear power programme. Furthermore, in most countries the inventories of spent nuclear fuel and high level waste grow slowly.

For these reasons, many Member States are currently deferring the decision on selecting an end point solution for their research reactor spent nuclear fuel, and it is expected that hundreds of research reactors worldwide, both operational and shutdown but not yet decommissioned, will continue with at-reactor or away-from-reactor storage of research reactor spent nuclear fuel (RRSNF) for long periods. This situation has raised concerns about the condition of the stored fuel and storage facilities, because spent fuel at both research and test reactors is being stored for longer periods than originally planned and in larger quantities. Furthermore, considering that 124 of the 240 operating research reactors and 181 of the 217 shutdown but not yet decommissioned research reactors are over 40 years old, it is understandable that concerns focus on the condition of the stored fuel, the ageing fuel storage facilities, the extension of their lifetimes and the ultimate disposition of spent fuel assemblies.

To help managers of research reactors and RRSNF storage facilities deal with interim storage of spent fuel, the IAEA has developed a programme of activities to study and discuss how to safely maintain the integrity of the fuel and to improve storage conditions until a final decision on the end point is made. Within this programme, a technical meeting was organized to discuss good practices for the management and storage of RRSNF. The meeting was held during October 2009 in Thurso, United Kingdom, and was attended by experts representing organizations having proven experience in handling and storing RRSNF.

The information assimilated during the meeting was collected and assembled in this publication, with the purpose of making available to managers of research reactors and managers of RRSNF storage facilities, information on current good practices on interim storage of RRSNF.

The IAEA wishes to thank all those who participated in the technical meeting, as well as N. Iyer, D. Vinson, J. Lian, P. Standring and A. Soares for their contributions to the final version of the report. The IAEA officers responsible for this publication were P. Adelfang and S. Tozser of the Division of Nuclear Fuel Cycle and Waste Technology.

EDITORIAL NOTE

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SUMMARY

This publication is divided in two parts. The first part includes the summary that introduces problems related to research reactor spent nuclear fuel management; a brief discussion about the options available for research reactor spent nuclear fuel (RRSNF) management; considerations about advantages and disadvantages of each option; a brief discussion about IAEA activities related to RRSNF management, and a very short summary of the IAEA Technical Meeting on Good Practices for the Management and Storage of Research Reactor Spent Fuel, held in Thurso, United Kingdom, during the week 19–22 October 2009.

The second part of this publication presents the proceedings of the technical meeting, which captures almost all the worldwide experience accumulated on interim storage of RRSNF. The presented papers identify and discuss the basic methods and activities related to dry and wet interim storage of RRSNF. The papers provide evidence that the integrity of RRSNF may be compromised when key parameters are not controlled. This may lead to additional efforts, e.g. fuel assembly repackaging, isolation of breached fuel, additional water conditioning with deionizers and filters, etc. This second part is divided into three sections: (a) review of standards and general criteria for spent fuel management, (b) wet storage practices and experiences, and (c) dry storage practices and experiences.

1. INTRODUCTION

For over 60 years, research and test reactors have made valuable contributions to the development of nuclear power, basic science, education, training, materials development, and radioisotope production for medicine and industry.

To allow the operation of the reactors, thousands of fuel elements were used, from different designs, types, shapes, material composition and enrichment. Considering that one of the purposes of the test reactors was to test and qualify fuel for specific nuclear applications, during these 60 years a large number of fuel types and fuel designs have been irradiated. In many cases, similar programmes have been undertaken in different countries, resulting in small amounts of spent fuel that cannot be considered as standard fuel. 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 disposal.

According to the IAEA research reactor database [1], since December 1942, when the Chicago graphite pile CP-1 went critical for the first time, 678 research reactors[1] (RR) were built in the world. From the 678 RRs built, 241 are operational in 56 Member States and 202 have been declared as shutdown in 26 Member States, but not yet decommissioned; with 13 explicitly declared as "temporarily shut down". This raises the concern that many of the shut down, but not yet decommissioned reactors still have fuel, both fresh and spent, at the sites. An extended delay between final shutdown and decommissioning will certainly affect both cost and safety at the time of decommissioning, mainly due to the loss of experienced staff (already ageing at the time of shutdown) necessary to participate in decommissioning activities.

With very few exceptions, in which there is no provision for fuel replacement, in all other research reactors the spent fuel elements are removed from the core when they reach the limit burnup, and replaced by new elements, allowing the reactor to operate for many years.

The research reactor spent nuclear fuel (RRSNF) has to be kept in a safe environment, generally in aqueous storage, for an extended time following reactor discharge to allow for the decay of fission products accumulated during reactor service and the removal of heat generated by this decay process. Usually it is stored either in the at-reactor (AR) pool or in away-from-reactor (AFR) wet facility, where the fuel elements are maintained for long periods.

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[1] In this publication the term "research reactor" also includes critical and sub-critical assemblies.

This wet storage can be extended, in AR or AFR facilities, in some cases over long periods, or the RRSNF may be transferred to dry storage (AR or AFR) sites and stored dry for even longer periods. Neither one of these strategies for RRSNF management can be considered as the end point of the back-end stage of the RR fuel cycle.

Considering current technologies, the end point of the RR nuclear fuel cycle is attained when the RRSNF is reprocessed, with the high level and long lived wastes disposed of in a geological repository and the useful isotopes reused; or when the spent fuel assembly is directly disposed, after proper conditioning, in a permanent geological repository.

Regardless of how long the extended interim storage is drawn out; the resolution of the back-end problem will remain, and proliferation, safety and physical security concerns will continue and so will continue the commitment of the RR operating organization/Member State to ensure safe, secure and economic management (especially storage) of its own RRSNF.

As many Members States are presently deferring a decision on selection of the end point solution for their RRSNF, it is expected that hundreds of RRs worldwide, both operational and shut down but not yet decommissioned, will continue storing RRSNF AR or AFR for long periods. It is also understood that it may be a long time before there is any national, regional or international deep geological repository available for most RRSNF, and in the interim, the development of new technologies will be necessary, in addition to political and legal changes, as well as transformation of public attitudes that may favour the implementation of a final repository.

The decision will be more difficult for countries with one of two research reactors and no nuclear power programme, because in additional to technical issues, it involves political issues, and sometimes prohibitive costs, if the country has to develop a national infrastructure to dispose only the research reactor spent fuel. Therefore, for many of the 443 operational and shutdown research reactors the continued safe, reliable and economic handling, management and storage of the RRSNF will be a serious issue, because in some cases the storage period and amount of spent fuel stored is, or very soon will be, much greater than originally planned, requiring studies concerning life extension and modifications to increase the capacity for the storage facility, until a decision is made on the final disposition of the fuel.

2. OPTIONS FOR RRSNF

After discharge from the core, the RRSNF is usually stored under water for cooling during a period of time between 3 and 5 years, to remove the so called residual decay heat, typically in at-reactor (AR) facilities. During this period a decision has to be made about the future of the RRSNF. A management strategy has to be defined based upon the available options, which are, (1) send the RRSNF for reprocessing, (2) return the fuel to the country where it was originally enriched (transferring the problem to others), (3) make a decision to directly dispose the spent fuel in a national repository, 4) transfer it to a dry storage facility or 5) simply keep it in wet storage for a longer period, postponing the time to make a final decision.

2.1. RRSNF reprocessing

Reprocessing is a proven technology for dealing with standard types of MTR fuels, and some countries have taken advantage of this option. However, it has not been used for small quantities of some experimental spent fuels, and even for some standard types of fuel, e.g. TRIGA type fuel, where it has only been carried out on a demonstration scale. In addition, political and economic issues may be very difficult to overcome, in order to justify this option, especially for cases where small quantities of RRSNF are involved. Moreover, in many cases, especially for countries with one or two research reactors but no nuclear power programme, the infrastructure to take-back high level waste from reprocessing is not available, and transportation costs of the fuel to the reprocessing plant, and the return of the waste is prohibitively expensive. The situation is aggravated by the continual ratcheting of transportation standards for public relations concerns rather than for safety reasons.

Before making the decision for reprocessing the spent fuel, the reactor owner and/or the reprocessing company have to answer some important questions (assuming that the problems associated with the transportation of the RRSNF have been solved):

- What is to be done with the reprocessed fissile material (uranium and in some cases plutonium)? Can it be used for the fabrication of new research reactor fuel elements or sold for other peaceful uses, and will the reactor owner get a credit for it?
- What is to be done with the waste from the reprocessing? Will the waste be kept by the reprocessor or will it be returned to the reactor owner? What are the specifications for the transportation cask for the waste from reprocessing? Is the reactor owner in fact able to take back the waste, or are there national restrictions, limitations and laws which make it impossible? If the reactor owner receives the waste, is the country ready for its final disposal?

2.2. Return of RRSNF to the country where it was enriched

Return of the fuel to the country where it was originally enriched, has been selected as the preferred option by countries that are eligible for taking advantage of the two international RRSNF take-back programmes; the USA Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) acceptance programme and the Russian Research Reactor Fuel Return (RRRFR) programme. Many countries are taking advantage of these programmes [2, 3].

However, these programmes are only available for USA and Russian origin fuels, and some additional fuels classified as “gap material”. The take-back option is currently not available for countries with fuel elements containing uranium enriched in countries others than the USA and the Russian Federation. Reports indicate that some facilities have been storing this category of RRSNF for more than 20 years in wet storage, awaiting an expected political decision that could allow the inclusion of the material in one of the two take-back programmes.

There is no established deadline for the duration of the RRRFR programme, but it is only available to Former Soviet Union (FSU) or the Russian Federation research/test reactors in 17 countries that possess nuclear fuel supplied by the FSU or the Russian Federation.

In May 2019, the US take-back programme will cease accepting RRSNF from foreign research reactors that have US-origin fuel irradiated before May 2016. The programme may be extended, however, it is clear that the take-back programmes will not continue indefinitely, and at some stage in the not too distant future many countries will face the problem of finding their own solutions regarding the final disposal for relatively small amounts of spent fuel, unless they decide to permanently shut down their research reactors before May 2016 (for countries with US-origin fuel).

2.3. Final disposal of RRSNF

Final disposal is a very expensive solution that can be afforded by only a few countries. For countries with no nuclear power programme, the construction of geological repositories for the relatively small amounts of spent fuel from one or two research reactors is obviously not practicable. For such countries, access to a regional or international repository for research reactor fuel would be the ideal solution. However, this option has been mainly investigated and developed for standard fuels from power reactors, and in principle the licensing process for the disposal facility may be complicated 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 reported in the open literature [4]. 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.

For this option it is also necessary to consider the possibility of developing a regional or multinational repository. Clearly, access to a regional long-term interim storage facility and eventually a multinational repository is an ideal and acceptable solution, and international interest considering this solution has been expressed in two major events:

(1) The International Conference on Storage of Spent Fuel from Power Reactors, held in Vienna, 2–6 June 2003, organized by the IAEA in cooperation with the OECD Nuclear Energy Agency, in which one of the conclusions was: “Representatives of Member States with smaller nuclear programmes informally expressed continued interest in regional storage initiatives, as well as topic specific workshops and training courses”.

(2) During the Forty-seventh Regular Session of the IAEA General Conference in 2003, when the Director General of IAEA stated: “Our consideration should also include the merits of multinational approaches to the management and disposal of spent fuel and radioactive waste. Not all countries have the appropriate conditions for geologic disposal and, for many countries with small nuclear programmes for electricity generation or for research, the financial and human resource investments required for research, construction and operation of a geologic disposal facility are daunting. Considerable economic, safety, security and non-proliferation advantages may therefore accrue from international cooperation on the construction and operation of international waste repositories. In my view, the merits and feasibility of these and other approaches to the design and management of the nuclear fuel cycle should be given in-depth consideration”.

The possibility of a regional and multinational repository is also addressed by the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, which states on its article (xi): “Convinced that radioactive waste should, as far as is compatible with the safety of the management of such material, be disposed of in the State in which it was generated, whilst recognizing that, in certain circumstances, safe and efficient management of spent fuel and radioactive waste might be fostered through agreements among Contracting Parties to use facilities in one of them for the benefit of the other Parties, particularly where waste originates from joint projects”.

In reality, many issues have to be discussed before a regional or multinational repository is established. The implementation of this option requires a significant political effort to resolve the complex array of agreements needed. However, since an international repository would resolve most of the problems being tackled by the Global Threat Reduction Initiative (GTRI) in the areas of non-proliferation and securing materials, it is understood that it could be pursued jointly by GTRI and IAEA.

2.4. Interim dry storage of RRSNF

Dry storage includes the storage of spent fuel in vaults or in casks. Vaults consist of reinforced concrete buildings (or modulus) containing arrays of storage cavities suitable for containment of one or more fuel units. A cask is a sealed metal cylinder containing the spent nuclear fuel, and which may or may not be easily transported. The concept of dry storage utilizing casks has been selected by some research reactor managers for interim storage of RRSNF, with the perspective that, depending on the design of the cask, it can eventually be used to transfer the spent fuel to the final disposal site, or to a reprocessing facility.

Interim dry storage using casks presents many advantages compared to other options, however it is not a very simple process, and requires some infrastructure that may be too expensive, especially when the spent fuel assembly discharge rate from the reactor is not very high, for example, below 10 fuel elements per year. Corrosion damage, gas pressurization, metal embrittlement, and radiation damage are mechanisms that may have the potential to eventually degrade cask performance, and it is necessary to demonstrate that the damage sustained over the storage period (up to 50 years or more, according to the RRSNF management programme) does not adversely affect the canister’s ability to remain leak tight during storage and eventual transportation of the spent nuclear fuel.

In order to avoid the development of any deleterious mechanism, it is necessary that the RRSNF be properly dried, backfilled, sealed, and maintained under controlled storage conditions. Sometimes a hot cell is used for the loading and drying operation, and attention must be paid to the security of the RRSNF after the reactor reaches its lifetime. As mentioned before, considering the technological and economic issues involved, this option, interim dry storage, may not be feasible for many countries with very small nuclear programmes, comprising only one or two low power research reactors.

2.5. Extended interim wet storage of RRSNF

RRSNF can be stored, in AR or AFR wet facilities, for long periods. However, to keep the RRSNF in wet storage, it is essential to have a suitable storage structure and a very strong programme for maintenance of the water quality. Supporting structures must be appropriated, with materials compatible with the materials of the fuel assemblies. Physical arrangements should facilitate water circulation to avoid the development of stagnant regions within the pool, and, as feasible as possible, minimize the possibility of having settled solids, like airborne dust, corrosion products and precipitated salts in the water. The suspended solids may settle on the surface of RRSNF cladding and may start a localized corrosion process, that once initialized can progress, potentially compromising the integrity of the RRSNF if left unmitigated.

Excellent water quality in spent fuel wet storage facilities is essential to achieve optimum storage performance. Experience shows the remarkable success of many research reactors where the water chemistry has been well controlled. In these cases, aluminium clad fuel elements and aluminium pool liners show few, if any, signs of either localized or general corrosion, even after more than 30 years of exposure to the water in the pool. However, maintaining high quality water of a storage pool requires stringent control of physical-chemical parameters, like conductivity, pH and ion concentrations, and usually this is not an easy task. In general, it requires continuous operation of the associated systems, which is not the case in most reactors, especially the ones that operate a few hours per day, and few days per week. In these facilities, usually the storage pool systems follow the reactor operational cycle. Most research reactor storage pools were designed as service pools where the fuel would be kept only for about 10 years, considering that another option would be available after that time. They were typically not designed and constructed with sufficient support systems to facilitate long term interim storage, and consequently, many cases of poor water conditions in storage pools have been reported; in some cases causing severe degradation of the fuel cladding [5].

2.6. Advantages and disadvantages of each option for RRSNF

The available options discussed previously present advantages and disadvantages. For most countries, especially the ones with no nuclear power programme, the best solution is to return the fuel to the country where it was originally enriched. However, fuel repatriation options are available for only a limited time and for a limited scope of materials. Continued operation of research reactors will require an alternate disposition path. In addition, according to the information available; a large number of fuel elements may not be eligible for the existing acceptance programmes.

Developing a national reprocessing capability requires the development of a very sophisticated technology that is not cost-effective to handle the fuel of a few research reactors. The cost of reprocessing the fuel abroad may be a challenge for most of these countries. The option of reprocessing fuel abroad does not preclude the necessity of addressing the issue of final disposition of the high level waste generated by reprocessing. Likewise, the development of a national repository may not be feasible or practical for economic reasons. The development of a regional or international repository would be an ideal solution. However, it may require decades of negotiations and agreements before it can be implemented. Issues such as compensation, property, possibility of future utilization, responsibility for maintenance, etc. would need to be clearly defined, and laws need to change; since many national legislations forbid receiving radioactive material that is considered "waste".

It is not surprising that interim storage, wet or dry, has been in the past and still is the spent fuel management strategy of choice at the overwhelming majority of research reactor facilities. It is a short term and relatively low cost option that keeps treatment strategies open, but delays positive action. Its main purpose is to maintain the integrity and the retrievability of the spent fuel, looking forward for the possibility to have a more feasible option in the future and postponing the decision about a final solution. The main challenge of the RRSNF interim storage is the necessity to avoid fuel degradation during storage. Safe, extended, interim storage requires detailed knowledge and mitigation of potential corrosion mechanisms. It is necessary to ensure material compatibility with the structures and components of the storage system in service. This may be a difficult task, considering that about 139 of the 241 operating facilities and 141 of the 202 shutdown but not yet decommissioned facilities are over 40 years old.

For the five options available, three of them represent a definitive solution for the reactor operator: return of the fuel to the country of origin, send for reprocessing, or send for final disposal. If none of these options is available, then the only alternative left is the interim storage of the spent fuel, and the reactor manager needs to take a decision between wet and dry interim storage, which is not a very easy decision.

Wet storage looks simpler, because there is the feeling that the only action needed is to keep the fuel under water, in the service pool or in a storage pool. However if an up-to-date water quality management programme, with excellent purification system and a corrosion surveillance programme is not implemented, in a few years the spent fuel will be affected by corrosion, compromising its integrity. Therefore, when selecting wet interim storage, the reactor manager needs to be aware that operational costs are considerable, and for all the period of the interim storage. Non-sealed dry storage is similar to wet storage, the only difference being that dry non-sealed storage does not require a water purification system, which can result radiation doses to maintenance staff. On the other hand, dry non-sealed storage requires a continuous ventilation system, with air free of moisture (humidity), which also implies in operational costs for a long time.

The sealed dry storage option has lower operational costs than wet and non-sealed dry storage, but the initial investment is higher. A high technological infra-structure is needed to handle, dry and seal the spent fuel. It may include the necessity of building a hot-cell, or the development of some equipment with enough shielding, where the fuel can be properly dried and encapsulated. Once the fuel is properly dried and put in the canister, the probability of corrosion decreases considerably, and no additional action is needed, unless there is some evidence requiring a mitigating action, like for example, canister pressurization.

3. IAEA ACTIVITIES IN SUPPORTING OF SOLUTIONS FOR RRSNF

Problems associated with research reactor spent fuel storage have loomed larger in the international nuclear community during the last decades. A number of concerns were evident at the beginning of 1993, when it was clear that many research reactors were in a crisis situation or rapidly approaching a crisis situation. In every case, this was due to spent fuel storage and management problems and the constraints of national laws. It was clear that the capacity for spent fuel storage had reached or was close to the design limit at many research reactors and there were concerns, from a materials science point of view, about ageing materials in ageing storage facilities, and its consequences for the integrity of the fuel elements.

A critical case was documented in which corrosion of the cladding of the stored spent fuel became a serious concern, when 137Cs nuclide activity was detected in the water of the storage pool; an indicator of loss of integrity of the fuel cladding [5].

Considering all these concerns and real problems, the Agency decided to promote activities with research reactor managers and operators, in order to share information and experience on this issue, and eventually to establish a programme of activities that would culminate with a publication, intended to make available to the whole research reactor community, a consistent set of internationally accepted good practices for management of RRSNF.

To obtain an overall picture of the size and extent of the RRSNF management problem, the Agency organized an Advisory Group Meeting (AGM) on Storage Experience with Spent Fuel from Research Reactors, which was attended by twelve participants and three observers representing thirteen different countries [6]. At the end of the meeting the participants reached 3 conclusions, reproduced below:

- In response to the general trend to store research reactor fuels for longer and longer times in wet *interim* storage, there is a need to identify from the recorded experience to date, the critical parameters controlling corrosion and other forms of material ageing leading to the degradation of mechanical and physical properties. There is a large amount of information at the various research reactor sites that has never been analysed, correlated and summarized. This needs to be gathered together, and conclusions drawn should culminate in a "*code of good practices*" for the safe and reliable storage of research reactor fuels in water.
- It is clear that those involved in the storage and management of research reactor fuels have a useful source of information available from the parallel but more advanced development of the storage of fuels from nuclear power plants. Nevertheless, it is concluded that the specific problems of research reactor fuels should be identified and evaluated, especially those related to materials, operational procedures in storage facilities and criticality evaluations associated with rack design and expansion, in an attempt to define and prioritize areas where information exchange is needed or research efforts need to be initiated or strengthened.
- Special attention should be paid to the conditioning and packaging of failed fuel. There is widespread concern that, at some time in the future, there will be a spate of failures of aluminium clad fuels all occurring at roughly the same time and corresponding to about the same long time in wet storage. In such a case, there would be a need for international collaboration that would somehow ensure the availability of an underwater canning capability for failed MTR fuel. In addition, the participants saw that there was an urgent need to develop *an exhaustive set of standard safety criteria* applicable to the development, construction and operation of interim storage facilities for research reactor fuels.

Based on these conclusions, the Agency started a programme to help research reactor managers to deal with the spent fuel management problem. The activities included Technical Meetings (TM) and Coordinated Research Projects (CRP), which resulted several publications available for all managers of research reactors and RRSNF storage facilities.

The activities involved:

- Exchange of experience in storage options, procedures and practices [4, 6];
- Management of ageing materials in spent fuel storage facilities [7, 8];
- Corrosion and other forms of material ageing leading to the degradation of mechanical and physical properties of RRSNF[9, 10];
- Study of regional solutions for research reactors in Latin America [11];
- Exchange of experience in the two international RRSNF take-back programmes; the USA FRRSNF acceptance programme and the RRRFR programme [2, 3].

Also, in an effort to better quantify the status of RRSNF management and storage worldwide, the IAEA developed a database on the subject, the Research Reactor Spent Fuel Database (RRSFDB). For this RRSFDB, the IAEA surveyed research reactor operators in Member States with research reactors and received answers from a limited but representative number of research reactors.

Since most research reactor fuels are shipped in assembly form, in the RRSFDB spent fuel numbers are recorded in assemblies, where a fuel assembly is defined as "the smallest fuel unit that can be moved during normal reactor operation or storage". Even so, questions regarding numbers of fuel assemblies obviously caused confusion to respondents to the questionnaires. Consequently, the data received was reviewed and corrected by a panel of experts who knew details of the various fuel assembly designs.

Analysis of the received data presented the following picture:

- Several facilities reported to store more than three different types of spent fuel types, and eventually spent fuels of different enrichments, for example facilities originally fuelled with Highly enriched uranium (HEU) that decided to convert the reactor core to low enriched uranium (LEU).
- Considering that a large variety of fuel types and fuel assembly geometries are in use in research and test reactors, special storage conditions are often necessary. In addition, different types of transport casks and different techniques for dealing with failed fuel may also be necessary.
- Overall, by the time of the survey (2003), there were 62 870 spent fuel assemblies stored in the facilities that responded to the RRSFDB questionnaires and another 32 932 assemblies in the standard cores. Of these 62 870 assemblies, 46 394 were in industrialized countries and 16 476 in developing countries, while 22 686 were HEU and 40 184 LEU [12].
- The distribution of fuel types among the reactors in the RRSFDB showed that the majority was of MTR, TRIGA or standard Russian types, with a significant percentage (28%) classified as "other" types. This underlines the fact that many experimental and exotic fuels exist at research reactors around the world, and that this variation poses problems for their continued storage, transportation, and ultimate disposal.
- The numbers of US origin and Russian origin fuel assemblies at foreign RRs, by the time of the survey (2003), were 15 531 and 29 673, respectively. In 2005 these numbers were 12 850 and 24 803, respectively [13].

Considering that the processes leading to the degradation of materials involved in the wet storage of RRSNF were well understood, and that significant experience on the long term interim storage of RRSNF within the Member States was available, in 2008 the Agency organized a meeting with international experts to discuss the feasibility and necessary steps to produce a publication intended to make available to the whole research reactor community a consistent set of internationally accepted good practices for management of RRSNF.

The meeting had a focus on long term wet and dry storage technologies and strategies for research reactor spent fuel, and the experts agreed that:

- The publication of a guidance document covering interim storage of research reactor spent fuel would make a great contribution in advancing RRSNF management practices;
- Given the uncertainty on the final disposition path and the anticipated long storage periods, the guidance document should consider the life management/life extension of the entire storage system/infrastructure vis-à-vis solely the SNF performance during storage;
- There is limited information available and guidance provided on dry storage of RRSNF. On the other hand, the technical basis and lessons learned in wet storage of RRSNF are well documented. However, in the context of the current day challenges, for safe and increasingly cost effective interim storage of RRSNF, the best practices already outlined for wet storage might require revising to draw upon the latest developments in wet storage chemistry envelope for cost effective spent fuel management.

Considering these conclusions, the experts recommended that a technical meeting should be organized, with the purpose of assembling operators and managers of research reactors and spent fuel storage facilities, to review their practices and plans for the management of RRSNF. The meeting participants should represent organizations that have experience in wet and dry storage of RRSNF.

The technical meeting would give an overview of storage system needs and the management of SNF in both wet and dry storage systems. The information assimilated during the technical meeting should be integrated into the development of an interim storage guidance document.

A consultancy team would synthesize and assimilate the information presented in the Technical Meeting, and develop assignments and the schedule for issuance of the interim storage guidance document.

The guidance document should develop examples of best practices and lessons learned for wet and dry storage. This could include best practices of the entire storage cycle and/or discrete portions of the storage life cycle, e.g. drying treatment, surveillance programme and characterization.

In summary, the team concluded that a guidance document on interim storage of RRSNF was indeed warranted, emphasizing the experience on dry storage and building on existing IAEA documents with respect to wet storage. In order to update the available information toward the development of the guidance document, the group proposed that the next step should be the organization of the proposed technical meeting. The technical meeting was held in October 2009 in Thurso, United Kingdom, and is summarized in Section 5 of the current publication.

4. SUMMARY OF THE TECHNICAL MEETING

Under the IAEA regular budget activity on updating the information on interim storage of RRSNF, a Technical Meeting on Good Practices for the Management and Storage of Research Reactor Spent Fuel, held in Thurso, Scotland, during the week 19–22 October 2009.

The main purpose of the meeting was to assemble operators and managers of research reactors and spent fuel storage facilities, to review their practices and plans for the management of RRSNF. The meeting participants represented organizations that have experience in handling and storing RRSNF, to exchange information and transfer lessons learned to managers of research reactors about the main issues related to RRSNF storage.

The meeting was primarily composed of invited participants from Member States and the participants were required to present a review of the RRSNF interim storage status and/or issues and challenges and also provide a written paper. The meeting also included a tour and review of the spent fuel retrieval and storage activities at the UKAEA Dounreay facility in Thurso.

The presented papers included discussion of the following specific topics with the primary focus on the technical and safety issues related to interim wet and dry storage of RRSNF:

- National legislation;
- Legal issues;
- Institutional and political aspects and public acceptance;
- Sources and quantities of RRSNF;
- National practices;
- National facilities and capabilities;
- Logistic and supporting infrastructure (local on national basis);
- Conformance with nuclear safety requirements;
- Quality assurance;
- Safeguards;
- National plans/options related to storage of RRSNF for the next 50 years;
- Comparison between wet and dry storage of RRSNF.

The meeting was hosted by the Dounreay Site Restoration Limited (DSRL) and held at the Weigh Inn, Thurso, UK.

The meeting was attended by 32 experts from 19 Member States. The participants provided 27 technical presentations and 22 papers that can be classified into three categories, (a) review of standards and general criteria for spent fuel management and national programmes, (b) wet storage practices and experiences, and (c) dry storage practices and experiences.

5. CONCLUSIONS

The reports presented at the technical meeting (TM) demonstrated the diversity of practices with respect to interim storage of spent fuel in research reactor facilities. Most Member States practice either wet storage or dry storage and many of the Member States currently depend on the USA or the Russian Federation for the eventual disposition of the fuel under the existing take-back programmes. A limited number of Member States have the flexibility or the infrastructure to support permanent disposition of spent fuel. The costs for development and maintenance of a geological repository for hundreds of years makes it prohibitive for most countries, especially for those with one or two research reactors, and no nuclear power programme. The same argument applies to the possibility of reprocessing.

Other conclusions drawn by the meeting participants were:

- Given the uncertainty on the final disposition path, it is foreseen that long term, wet or dry interim storage will be the selected option for most research reactor managers.
- A number of detailed reports have been issued on the technical bases and lessons learned for wet storage, however, there is very limited information and guidance provided on dry storage; hence, in addition to the proceedings of the meeting, an IAEA guidance document covering interim storage of spent fuel, with emphasis in dry storage technology, will add value to advance the RRSNF management practices.
- The guidance document should contain examples of good practices of the entire storage cycle, and discrete portions describing details of the storage life cycle, e.g. drying treatment, surveillance programme and characterization, and consider the life management/life extension of the entire storage system/infrastructure vis-à-vis solely the RRSNF performance during storage, using the best knowledge and technology available to maintain the safety and integrity of the RRSNF.
- The information assimilated during the technical meeting should be integrated into the development of the interim storage guidance document, and a plan developed to solicit and gather SNF storage practices from Member States who were not represented at the TM.

The technical meeting also helped the participants to familiarize with the IAEA safety standards, and provided the framework for the development of the interim storage guidance document.

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REVIEW OF STANDARDS AND GENERAL CRITERIA FOR RESEARCH REACTOR SPENT FUEL MANAGEMENT AND NATIONAL PROGRAMMES

(Part 1)

IAEA SAFETY STANDARD SSG-15 — STORAGE OF SPENT FUEL

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Abstract

This paper highlights some of the safety recommendations presented in IAEA Safety Guide SSG-15, on the "Storage of Spent Fuel", a document produced to provide up-to-date guidance and recommendations on the design, safe operation and assessment of safety for different types of spent fuel storage facilities, considering wet and dry storage of spent fuel from nuclear power and research reactors.

1. INTRODUCTION

The IAEA has produced an IAEA Safety Standard Series (SSG-15) on the "Storage of Spent Fuel" [1]. The scope of the Safety Guide includes the storage of spent fuel from water moderated reactors and can, with due consideration, also be applied to other fuel types, such as those from gas cooled reactors; research reactors and also to spent fuel assembly components and degraded or failed fuel that may be placed in canisters.

The objective of the safety guide is to provide up-to-date guidance and recommendations on the design, safe operation and assessment of safety for different types of spent fuel storage facilities (wet and dry), considering different types of spent fuel from nuclear reactors, including research reactors, and different storage periods, including storage going beyond the original design lifetime of the storage facility.

The safety guide also presents guidance and recommendations on how to fulfil the safety requirements established in the following IAEA Safety Requirements publications: Safety of Nuclear Fuel Cycle Facilities [2], Radioactive Waste Management [3], Safety Assessment for Facilities and Activities [4], and The Management System for Facilities and Activities [5].

2. CONTENT OF SAFETY GUIDE

The safety guide contains approximately 600 recommendations covering the roles and responsibilities of government, regulatory bodies and operating organizations, as well as recommendations covering management systems, safety case and safety assessments as well as the safety measures to be considered. These recommendations are summarized below:

2.1 Role and responsibilities

The storage of spent fuel should be undertaken within an appropriate national, legal and regulatory framework that provides for a clear allocation of responsibilities. Specifically:

Government: The government is responsible for establishing a national policy for management of spent fuel as well as establishing a legal and regulatory framework to implement that policy. The legal framework should establish an independent regulatory body.

Regulatory body: The regulatory body is responsible for establishing the requirements and providing appropriate regulatory guidance for the safe storage of spent fuel. The regulatory body should verify that the operation of the storage facility is compliant with the legal and regulatory requirements and ensure that the operating organization provides the necessary personnel and technical and financial resources, for the life time of the facility.

Operating organization: The operating organization is responsible for safety. It should prepare a safety assessment and safety case that demonstrates the safety of the spent fuel and the safety of the spent fuel storage facility, and operate the facility in compliance with the safety case and legal and regulatory requirements.

2.2 Management system

The objective of the operating organizations management system should be to ensure the safety of the spent fuel and the spent fuel storage facility. The management system should be established and regularly assessed to ensure continuous improvement in operation and safety. In view of the increasing storage periods, the safety guide highlights the importance of ensuring adequate records are maintained.

In view of the increasing storage periods, the safety guide also recommends that the management system should address the provision of adequate resources, the preserving of knowledge and its transfer to people joining the organization, as well as succession planning for technical and managerial staff.

2.3 Safety assessment and safety case

A safety assessment and safety case are required to be produced to demonstrate the safety of the spent fuel and the spent fuel storage facility. The safety case should be the primary input to the licensing process and a means of demonstrating compliance with legal and regulatory requirements.

The safety case should demonstrate how the safety functions: containment of radionuclides (including fuel integrity); criticality safety; heat removal; radiation shielding and retrievability, are to be provided. The last safety function relating to retrievability is very important. Storage is by definition an interim measure; however, storage periods are being extended. Ultimately the spent fuel will have to be retrieved for either disposal or reprocessing. The safety case should therefore address safe handling of spent fuel following the period of storage taking into account any degradation affecting the ability to retrieve and handle the spent fuel.

The safety case should identify the structures, systems and components required to ensure safety and define their safety function and determine their performance requirements, including their redundancy, diversity, segregation and maintenance, inspection and testing requirements.

2.4 Safety considerations

The safety guide provides general overarching recommendations for a spent fuel storage facility and specific recommendations covering the design and operation of the facility. These recommendations are summarized below:

2.4.1 General

The design of a spent fuel storage facility should consist of relatively simple and preferably inherently safe systems which ensure that the safety functions are fulfilled for all operational states and accident conditions, including external hazards.

Spent fuel and spent fuel storage facilities should be sufficiently resistant to degradation for the life time of the facility and the storage environment should not adversely affect the properties of the spent fuel, particularly considering retrieval after any extended storage period.

Design should adopt a multiple barrier approach to containment and include consideration of the fuel matrix, fuel cladding, storage cask/vault and building structures.

Design of safety systems required to achieve the safety functions should consider: minimum human intervention; minimum reliance on monitoring and allow retrieval of spent fuel for inspection and reworking.

2.4.2 Design

The design should adopt the defence in depth approach aimed at preventing accidents and mitigating them should they occur.

The spent fuel storage facility should have a reserve capacity to allow re-shuffling of spent fuel for inspection, retrieval and maintenance work.

2.4.2.1.1. Containment

The design of the facility should ensure that the spent fuel cladding is protected against degradation. It is therefore important to determine cladding corrosion rates and when appropriate to maintain pool water quality. The design should include ventilation systems to ensure collection of airborne radioactive particles which should be provided with filtration systems and monitoring and maintain sub-atmospheric pressures inside the storage building.

2.4.2.2. Criticality

The safety guide recommends the use of geometrically safe configurations to ensure subcriticality under all credible circumstances. If this is not possible, additional means should be used such as fixed neutron absorbers or burnup credit. If using fixed neutron absorbers, it should be ensured by design and fabrication that the absorber will not be separated or displaced during all operational states and accident conditions. If using burnup credit, the design should minimize the possibility of spent fuel being misplaced during storage.

If the design allows for boiling of pool water, the change in water moderator density should be included in the criticality safety assessment.

Use of soluble poison to ensure criticality safety of pool storage should be avoided.

Dry spent fuel storage facilities should maintain a criticality margin even under water flooding.

2.4.2.3. Heat removal

The safety guide recommends that heat removal capability should be provided to ensure that the temperature of the spent fuel cladding does not exceed its maximum allowable. If reliance is placed on active heat removal systems, they should satisfy the deterministic single failure criterion.

2.4.2.4. Radiation shielding/radiation protection

The source term should consider bounding conditions, e.g. enrichment, burnup and maximum storage inventory. Where pool water provides shielding the water level should be maintained with an adequate and suitable water supply. Penetrations below the required minimum water level should be excluded.

2.4.2.5. Retrievability/handling

Handling equipment should be designed to minimize damage to spent fuel. The forces being applied to the handling of spent fuel should be limited and sharp corners which could damage the surface of spent fuel should be eliminated. Latching mechanisms should be provided to prevent accidental release and there should be speed restrictions covering the movement of spent fuel and equipment.

2.4.3 Operation

The facility should be operated in accordance with written procedures to ensure compliance with operational limits and conditions. The operating procedures should cover: receipt, handling, storage, retrieval of spent fuel and also cover handling of degraded or failed fuel.

Modification of equipment or procedures should require prior approval and should be subjected to review and assessment based on the safety significance of the modification.

Spent fuel with decreased integrity should be canned to maintain quality of the storage environment.

Operating experience and events at the facility and other similar facilities should be collected and analysed in a systematic way to identify any design or operational improvements.

2.4.3.1 Criticality

The safety guide recommends that operating procedures for maintaining criticality safety should be subjected to a rigorous review to ensure that they remain consistent with the safety requirements of the design and existing operating practices.

Operational controls to maintain water condition should be provide to prevent boron dilution where soluble boron is used for criticality control.

2.4.3.2 Heat removal

Operational procedure should ensure that the rate of change of temperature is controlled, does not exceed limits and minimize the unavailability of pool cooling system.

2.4.3.3 Radiation shielding/radiation protection

Operational procedures should be provided to avoid loss of shielding, e.g. due to handling errors. An operational radiation protection programme should be in place to ensure that radiation doses are maintained as low as reasonably achievable and should include consideration of: classification of areas; controlled access; training; monitoring of radiation and contamination levels and the provision of dosimetry.

2.4.3.4 Retrievability/handling

Procedures should be provided for detecting and handling degraded and failed fuel, e.g. detecting ^{85}Kr, ^{134}Cs, ^{137}Cs, and other fission products.

3. SOME DEFINITIONS RELATED TO SPENT FUEL STORAGE

When dealing with spent fuel the definitions to be used are contained in the IAEA Safety Glossary [6], and the “Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management” [7]. Various terms are also defined in the IAEA Safety Standard Series (SSG-15) on the “Storage of Spent Fuel”[1].

Damaged Fuel*:* Safety guide SSG-15 makes reference to damaged fuel in paragraph 6.130 as: “Loss of integrity of spent fuel cladding may lead to the release of radioactive material into the storage environment. Such fuel can be referred to as degraded or failed fuel.”

Reference [6] also includes a reference to ***damaged fuel*** in the definition of gap release: “Release, especially in a reactor core, of fission products from the fuel pin gap, which occurs immediately after failure of the fuel cladding and is the first radiological indication of fuel damage or fuel failure.

***Dry Storage**:* [6] defines dry storage as: “Storage in a gaseous environment, such as air or an inert gas. Dry storage facilities include facilities for the storage of spent fuel in casks, silos or vaults.

***Good Practices**:* The preamble in the IAEA Safety Standards refers to good practices, under the description of Safety Guides, where it states: “Safety Guides provide recommendations and guidance on how to comply with the safety requirements, indicating an international consensus that it is necessary to take the measures recommended (or equivalent alternative measures). The Safety Guides present international good practices, and increasingly they reflect best practices, to help users striving to achieve high levels of safety. The recommendations provided in Safety Guides are expressed as ‘should’ statements.”

A definition of *good practices* is also given in the IAEA’s OSART (Operational Safety Review Teams) Guidelines [8] as: “A good practice is an outstanding and proven performance, programme, activity or equipment in use that contributes directly or indirectly to operational safety and sustained good performance. A good practice is markedly superior to that observed elsewhere, not just the fulfilment of current requirements or expectations. It should be superior enough and have broad application to be brought to the attention of other *[facilities]* and be worthy of their consideration in the general drive for excellence. A good practice has the following characteristics:

- Novel;
- Has a proven benefit;
- Replicable (it can be used at other plants);
- Does not contradict an issue.

***Long Term Storage**:* Reference [1] contains a discussion and definition of the phrase “Long Term Storage” Paragraph 1.6 states that: “Whilst design lifetimes of up to one hundred years have been considered and adopted for spent fuel storage, in view of the rate of industrial and institutional change, periods beyond around fifty years are deemed to be “longer term” in the context of this document.

Annex 1 of Reference [1] also states: “Long term storage is considered in this report as storage beyond fifty years, and with a defined end point. The storage endpoint is important since it becomes the basis for the design life of the facility, packaging requirements, financial guarantees and the planning basis for subsequent disposal facilities.

Operating lifetime: Reference [7] contains a definition of operating lifetime as: “Operating lifetime means the period during which a spent fuel or a radioactive waste management facility is used for its intended purpose. In the case of a disposal facility, the period begins when spent fuel or radioactive waste is first emplaced in the facility and ends upon closure of the facility;

***Wet Storage**:* Reference [6] defines wet storage as: “Storage in water or in another liquid. The universal mode of wet storage consists in storing spent fuel assemblies or spent fuel elements in pools of water or other liquids, usually supported on racks or in baskets and/or in canisters that also contain liquid. The liquid in the pool surrounding the fuel provides for heat dissipation and radiation shielding, and the racks or other devices ensure a geometrical configuration that maintains subcriticality.”

4. SUMMARY

The management of spent fuel from nuclear power plants and research reactors is an important concern relating to the use of nuclear energy. Repository projects are being delayed and many countries have yet to decide on the final destination of spent fuel. Consequently, long term storage is becoming a reality. The main safety recommendations from the safety guide have been highlighted; reference should therefore be made to the Safety Guide SSG-15, “Storage of Spent Fuel” [1], for the complete set of safety recommendations that would ensure the safe storage of spent fuel. However, it is important to note that in view of the extended storage periods, the requirements for containment, detection and handling of damaged fuel and its ultimate storage are becoming priority issues.

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OVERVIEW OF CRITERIA FOR INTERIM WET AND DRY STORAGE OF RESEARCH REACTOR SPENT NUCLEAR FUEL

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Abstract

Following discharge from research reactors, spent nuclear fuel may be stored “wet” in water pools or basins, or it may be stored “dry” in various configurations including non-sealed or sealed containers until retrieved for ultimate disposition. Interim safe storage practices are based on avoiding degradation to the fuel that would impact functions related to safety. Recommended practices, including environmental controls with technical bases, are outlined for wet storage and dry storage of aluminium-clad, aluminium-based research reactor fuel. For wet storage, water quality must be maintained to minimize corrosion degradation of aluminium fuel. For dry storage, vented canister storage of aluminium fuel readily provides a safe storage configuration. For sealed dry storage, drying must be performed so as to minimize water that would cause additional corrosion and hydrogen generation. Dry storage system design must also consider the potential for radiolytically-generated hydrogen from the bound water in the attendant oxyhydroxides on aluminium fuel from reactor operation.

1. INTRODUCTION

The design for most research reactor fuel is an aluminium-based fuel matrix that is clad with an aluminium alloy [1]. There are important considerations however in the use of aluminium and aluminium alloy claddings for reactor fuel during reactor operation, and especially in post-reactor fuel storage. Aluminium alloys show very different corrosion behaviour from fuels produced with stainless steel and Zircaloy claddings. For example, during reactor operation, aluminium claddings can develop relatively thick oxide layers [2] that are subject to spallation [3]. Also, aluminium fuel is particularly susceptible to corrosion degradation in water, particularly if the quality of the water is poor [4].

After discharge from the reactor and initial cooling in a water pool, nuclear fuel may be continued to be stored “wet” in an at-reactor (AR) pool, or in away-from-reactor (AFR) pool facility, or it may be placed in dry storage. Interim storage of spent nuclear fuel in these two types of storage systems, even for short time (e.g. less than 1 year), must consider potential degradation mechanisms in preparing the fuel for storage and the storage environment. The overall objectives for a fuel storage system are to (1) maintain degradation of the fuel within acceptable limits; and (2) minimize degradation of the storage system itself to meet the mission needs for interim spent nuclear fuel storage.

This paper provides a summary of recommended storage practices including technically based controls to environments of storage so as to avoid significant degradation of the fuel. The practices are focussed on aluminium-clad, aluminium-based fuel in wet and in dry storage systems. The practices to avoid significant degradation of the storage system materials are not addressed in this paper.

2. CRITERIA FOR STORAGE

Storage criteria are defined as the set of practices that are needed, including environmental controls, to limit the fuel materials’ degradation during storage. Fuel degradation can impact both safety functions and the fuel condition assumed for its ultimate disposition.

2.1 Fuel degradation impacts to safety functions in a fuel storage system

The change in fuel condition during preparation for, and during the storage period, should be limited to avoid adverse impacts to:

– Criticality safety;

- General containment by the fuel/clad system[2];
- The ability to retrieve the fuel;
- Options for ultimate disposition.

An understanding of requirements for each of these functions is important to identify limits to fuel degradation in storage. For example, corrosion of the fuel that causes extensive damage to its structure may challenge the ability to grasp and handle the fuel with tools.

2.2 Limits to fuel degradation in storage

Degradation of spent nuclear fuel in storage systems may include one or more of the following phenomena:

- A loss in net section of cladding through cracking, pitting, or general thinning;
- Embrittlement of the fuel/clad structure;
- Distortion of physical shape (sometimes termed "geometry") of the fuel;
- Diffusion of fuel or fission products through the cladding;
- Diffusion of volatile fission products through the fuel matrix (in fuel that may have breached cladding);
- Alteration of the fuel due to chemical reactions (e.g. oxidation at breaches in cladding).

Except for embrittlement of the fuel/clad structure that is a result of operation of the fuel in the reactor, each of the other degradation phenomena can be limited by storage practices including controls to the environments of storage. General practices, with the underlying technical bases, for wet storage and dry storage systems are described in the following sections. The specific practices for a specific design of a storage system could be developed from these general practices.

2.3 Wet storage criteria for aluminium spent nuclear fuel

Corrosion attack of the aluminium claddings is the primary degradation phenomena for aluminium fuel in wet storage. The types of corrosion that can cause significant attack to aluminium alloys in water are listed in Table I. Of these types, pitting, crevice, galvanic, sediment-induced, and filiform corrosion are the most prevalent for aluminium fuels in wet storage [4, 5].

As an example of corrosion attack to aluminium fuel, Figure 1 shows filiform corrosion that occurred in an assembly stored in poor water quality conditions [6]. Photographs of corrosion damage of several of the other types of corrosion are contained in figures in Reference [5].

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[2] In many interim storage systems, the cladding is not typically credited to provide confinement in the safety basis. Nevertheless, large breaches may cause undesired gross release of radioactivity, and for this reason, general containment provided by the fuel/clad system in interim should be maintained.

TABLE I. CORROSION TYPES FOR ALUMINIUM FUEL IN WATER STORAGE

General corrosion
Pitting corrosion
Crevice corrosion
Galvanic corrosion
Intergranular corrosion
Stress corrosion cracking
End-grain attack
Erosion-corrosion
Blister formation (filiform corrosion)
Microbial corrosion
Sediment-induced corrosion

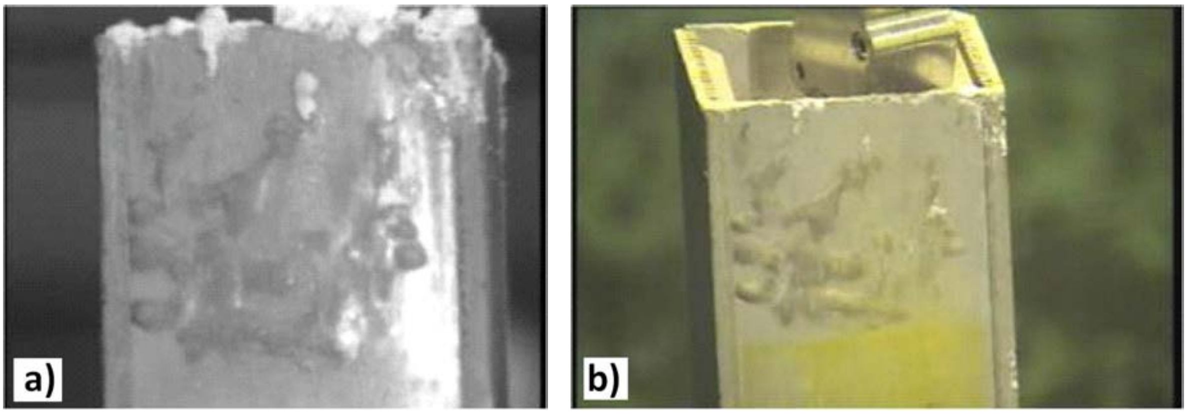

FIG. 1. Filiform corrosion causing blister formation in the outer aluminium fuel plate of an MTR design fuel assembly stored in poor water quality. Figure 1a) is an above-water photograph taken at the original fuel storage site. Figure b) is an under-water photograph of the same fuel assembly taken at the Savannah River Site, the present storage location [Figures reproduced from Reference 6].

The only plausible degradation of the fuel in wet storage significant to impact the safety functions of containment by the fuel/clad system and ability to easily retrieve the fuel, is that caused by corrosion. Corrosion of aluminium alloys can be minimized with controls of water quality and avoidance of crevices and galvanic couples with the fuel and storage system materials. Water quality is defined by a set of parameters that are used to characterize the water physical and chemical conditions. It includes pH; conductivity; dissolved impurity species; undissolved solids; colloids; organic substances; biological organisms; and temperature.

The recommended practices for wet storage of aluminium spent nuclear fuel, including the set of ranges and/or limits for the physical and chemical parameters of water quality, is contained in Reference [5]. The document also includes a detailed review of the corrosion of aluminium in water and the technical bases for the practices of wet storage of the research reactor spent nuclear fuel.

An important consideration is the initial condition of the fuel that would be placed in a wet storage system. Even breached fuel can be safely stored in a basin with active water clean-up systems. However, if it is significantly damaged, with excessive leaking of radioactivity and loss of ability to easily handle and move the fuel, it is recommended that the fuel be stored underwater in isolation canisters. These management practices for breached and damaged fuel are discussed in a companion paper [7].

2.4 Dry storage criteria for aluminium spent nuclear fuel

An evaluation of mechanisms that could degrade aluminium fuel for a range of dry storage environments and impact the safety functions for storage is contained in references [8] and [9], and is summarized here. Limits to the amount of fuel degradation (e.g. loss of cladding net section due to corrosion; fuel distortion due to creep) are coupled to limits for the environments of storage (e.g. air; temperature, water vapour). The limits to the environmental conditions were also developed as candidate acceptance criteria for a 50-year dry storage period for the aluminium fuel in references [8] and [9].

The fuel temperature in a dry system would be higher than in wet storage due to removal of water as a cooling media and consolidation of fuel to reduce the footprint of a storage system. Potential degradation modes of the aluminium fuel due to thermal processes only are distortion due to creep and diffusion of radionuclides through the cladding. In addition, the cladding is embrittled due to its irradiation in the reactor.

An evaluation of fuel distortion due to creep was previously reported [10]. Figure 2 shows the fuel plate model, and the results of finite element analysis under Coble creep deformation for temperatures from 150 – 250 °C for a 50-year period for several assumed grain sizes for aluminium. Creep rupture of the cladding is not expected due to the high ductility of annealed aluminium. However, excessive distortion would be expected to impact the ability to remove the fuel from the storage system, and a limit of 200 °C was specified [8] to avoid a deflection of 2.5 mm due to fuel plate slump due to creep for a 50-year period. Fuel and storage system-specific designs should be evaluated to avoid high fuel temperatures that would cause excessive creep for the planned storage period.

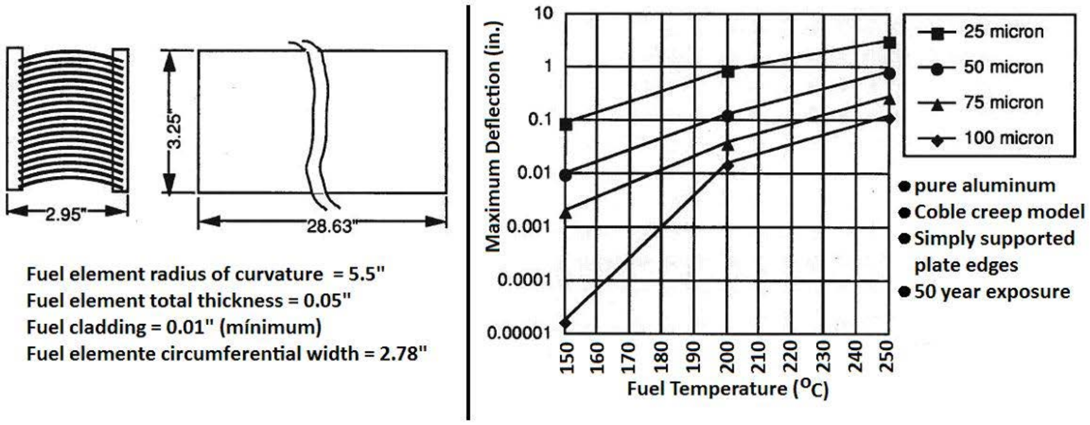

FIG. 2. Maximum vertical deflection of a single fuel plate after 50 years as a function of temperature. Average grain sizes for the aluminium are indicated in the legend [figure reproduced from Reference 10].

The diffusion of fission products, actinides, and uranium through the aluminium cladding can impact leachability of the fuel, and would result in a release of radioactivity if corrosion of the cladding would occur following the through-clad diffusion of the radioisotopes. Diffusion of radionuclides from the fuel through the aluminium cladding was also previously reported [9]. Considering the metrics of [11]:

Time to breakthrough = $0.06\ l^2/D$ and

Time to steady-state = $0.45\ 1^2/D$,

where D is the diffusivity (cm^2/sec) and l is the cladding thickness (cm), the results show breakthrough of Cs and U through an assumed 0.76 mm thick aluminium cladding in 50 years at a temperature of approximately 250 °C.

Aluminium cladding alloys are embrittled following exposure to a fast ($E_n > 0.1$ MeV) neutron fluence greater than approximately 1 x 10^{19} n/cm^2. This condition is typical for the spent fuel. The tensile ductility as measured by total and uniform strain of irradiated 1100 and 6061 aluminium reduces to less than 5% and 1%, respectively, between 150 and 200 °C [12]. No gross leakage of radioactivity would be expected even if cracking due to excessive strains did occur since the fuel core remains metallurgically-bonded with the cladding even at high burnup (See Figure 3)

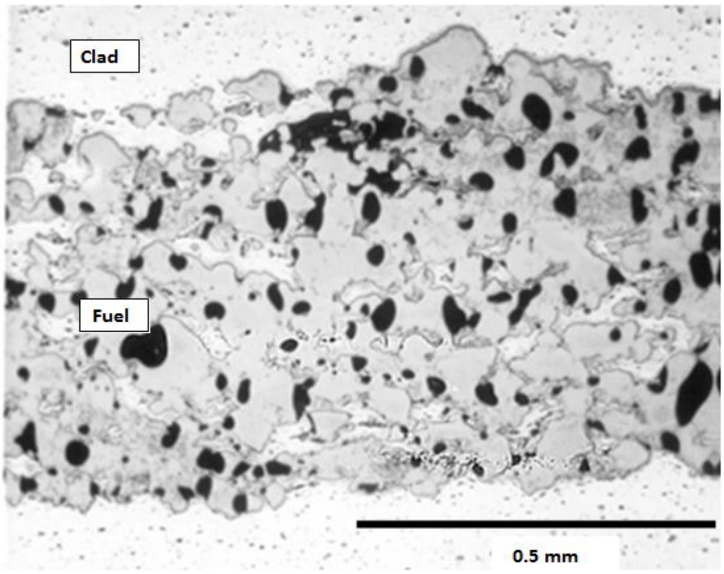

FIG. 3. Cross-section of aluminium-clad, UAlx-Al fuel irradiation to >50% burnup.

3. TYPES OF DRY STORAGE

Two distinct types of dry storage systems are envisioned: a “sealed system” which would store fuel assemblies in fully-sealed containers and a “non-sealed” system which would store assemblies in non-sealed containers or holders open to the environment of the facility.

3.1 Sealed dry storage

The approach to avoid excessive degradation in a sealed system is to dry the contents of water remaining in the fuel/container to-be-sealed, for both free and bound waters, to a level such that if the water is consumed by corrosion of the fuel, acceptable degradation of the fuel is maintained; and also that the production of hydrogen does not pose a threat. Gases produced may overpressure containers, embrittle materials, reach flammable concentrations, accelerate corrosion, or form pyrophoric species that would impact safe post-storage retrieval. As an example, rapid corrosion attack can occur with a combination of air, water, and the attendant radiation of a fuel.

A large set of vapour corrosion tests were performed for several cladding alloys and uranium-aluminium alloys to evaluate the corrosion response for temperatures up to 200 °C and relative humidity up to 100% [13]. Boehmite is the predominant oxide that forms at temperatures above approximately 80 °C [14] in accordance with:

$$4H_20 + 2Al = A1_20_3{\bullet}H_20 + 3H_2$$

The growth of a coherent passive hydrated oxide film under water vapour conditions, with concurrent metal consumption can be modelled with a mechanistically-based, phenomenological model given by [13]:

$$\Delta w,\ oxide\ thickness,\ metal\ loss = Ae^{(-Q/RT)} time^{n}$$

where Δw is the weight gain of a specimen which is proportional to the average oxide thickness on the specimen, and also to the metal consumed in the corrosion reaction.[3] In general, the parameter A is a function of alloy composition, cover gas, water vapour composition, and relative humidity (RH). The activation energy Q is a function of alloy composition and n is a function of alloy composition, and water vapour composition. A "critical" RH level below which corrosion stops has been observed for several metals and is between 40 to 70% for aluminium at room temperature [15]. At a temperature of 150 °C, the critical relative humidity is observed to be approximately 20% for the aluminium cladding alloys 1100, 5052, and 6061 aluminium [13].

Further tests were performed in which aluminium cladding alloys were placed in a sealed capsule with air and water and placed in a gamma cell [13]. The results showed that rapid corrosion occurred relative to the cases with no radiation as shown in Figure 4. In addition, corrosion occurred at all values of relative humidity, that is, there was no threshold in RH for corrosion to occur as compared to the case (not shown in Figure 4) with no radiation. The mechanism of the attack was attributed to a lowering of the pH of the water by radiolysis of the air. Additional information on impact of radiolysis on corrosion of aluminium in vapour/air environments is reported in Reference [16].

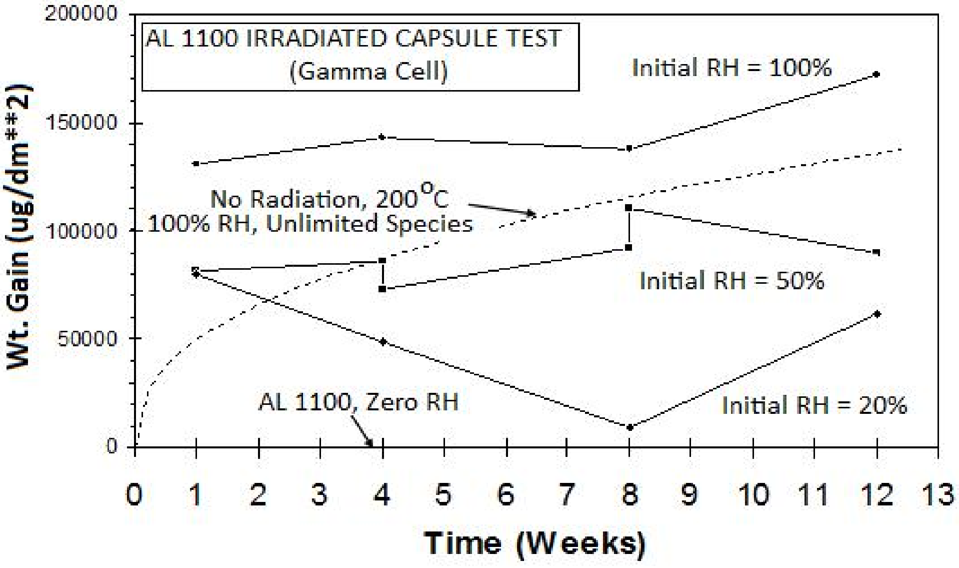

FIG. 4. Kinetics of vapour corrosion of aluminium 1100 under air and water vapour exposure under gamma (60Co) radiation at 1x104 Gy/hr [figure reproduced from Reference 13]. The water vapour in the capsule tests was limited and was effectively consumed by 1 week of exposure.

Aluminium fuel may be exposed to the environment through cladding breaches. Tests of the corrosion of uranium-aluminium alloys in dry air [17] and under water vapour conditions [13] have been reported. Figure 5 shows the results of testing non-irradiated aluminium-10% uranium and aluminium-18% uranium in saturated water vapour at 200 °C in comparison to 1100 aluminium. The formation of a passivating oxyhydroxide film on aluminium-uranium alloys limits its corrosion with time, similar to the pure aluminium cladding alloy 1100. Furthermore, no significant galvanic couple between the aluminium 1100 and the aluminium-uranium (Al-UAl_x) alloy materials in pure water was observed [13]. Similar behaviour is expected for the other aluminium-based dispersoid fuel (e.g. U_3Si_2-aluminium and U_3O_8-aluminium).

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[3] For Boehmite growth on aluminium: Metal Loss (mm) = 3.030×10^{-8} x Wt. Gain (□g/dm^2); Oxide Film Thickness (Boehmite film in nm) = 0.0533319 x Wt. Gain (□g/dm^2).

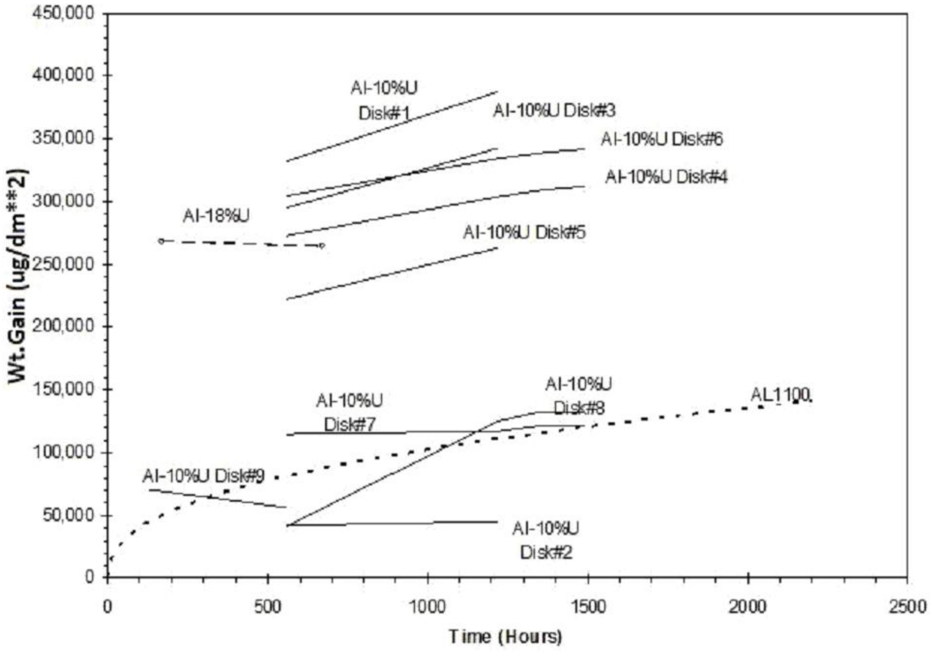

FIG. 5. Weight gain of aluminium and aluminium-uranium alloys in water vapour at 200 °C (Figure reproduced from Ref. [13]).

Another phenomenon that is expected to occur is radiolysis of the oxyhydroxides on the fuel to generate hydrogen gas. Exposure of oxyhrodroxides of aluminium, present on spent fuel, to gamma irradiation can produce radiolytic hydrogen gas [18]. Tests of pure Bayerite and Boehmite powders in both fully dried and with water absorbed from humid air conditions were investigated in initial work reported in Reference [18]. The oxides were in typical fine crystalline form (as powders) and were prepared under several conditions including particle size, and "dry" (without physisorbed and chemisorbed water) vs. "wet" conditions. The specific purpose of exposing these oxides to gamma radiation was to quantify hydrogen production under Argon cover gas in a closed stainless steel container. Other species were also measured (e.g. H_2O, O_2).

Sealed containers contained gibbsite ($Al_2O_3{\bullet}3H_2O$) and boehmite ($Al_2O_3{\bullet}H_2O$) powders prior to placement in the gamma irradiator. The powders were procured in powder form, with particles ranging up to 100 microns in size. Ten grams of both compounds were stripped of adsorbed water by oven drying at 60°C (no chemically bound water was removed) before loading the test vessels. In two samples, the gas was dry and in two, the gas was saturated. Two blanks were also tested, one with gas over 10 grams of demineralized water, and one with dry Argon only. The samples were placed in a Shepherd 109-08 gamma cell with a cobalt source, and irradiated at $5\text{x}10^5$R/hour ($5\text{x}10^3$ Gy/hour) for 168 hours. The gas was analysed in a Pfeiffer GAM400 mass spectrometer. Measurable quantities of hydrogen were generated from the dry and the moist boehmite while the Gibbsite produced little or no measurable hydrogen. Figure 6 shows the results of hydrogen that evolved from the materials.

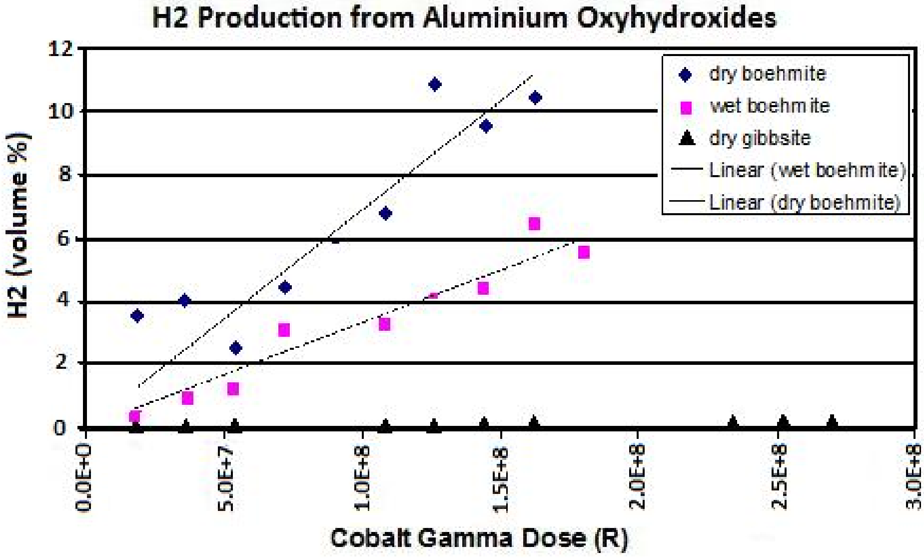

FIG. 6. Evolved hydrogen from gamma irradiation of aluminium oxyhydroxides (1 R = 10-2 Gy) [reproduced from Reference 18].

3.2 Non-Sealed dry storage

A non-sealed system is one in which fuel would be in contact with the ambient air in a dry storage facility (e.g. outside air temperature and relative humidity conditions). Since the corrosion of aluminium exposed to ambient air progresses at a slow rate, even at high humidity levels (up to 100%) at near room temperatures, safe storage in a non-sealed system can be achieved. One caution is non-sealed storage in "dirty" atmospheres (those containing low chloride and sulphate compounds). A non-sealed system would also be subject to demonstration of confinement in storage of breached fuel.

4. DRYING CRITERIA

If water remains with the fuel/canister configuration, material interactions may result in fuel corrosion damage, fuel container corrosion damage, criticality safety issues, generation of flammable gases (hydrogen), or formation of pyrophoric species (uranium hydride). As noted, water is of particular concern in sealed fuel storage containers.

To limit these issues in both vented dry storage and, imperative for sealed dry storage, it is important to remove the attendant water from the fuel to the extent practicable. The total amount of water associated with a spent fuel and canister system are from three basic sources; these are (1) free water (which includes water in pits in the cladding, crevices, etc.), (2) waters of hydration on existing oxyhydroxides, and (3) absorbed waters on oxyhydroxides and on aluminium metal and other materials in the system. General guidelines for drying of spent fuel to remove these waters have been developed [19].

A specific time-temperature recipe for use in a drying system for a fuel is highly dependent on the following factors:

- Fuel configuration (e.g. plate fuel with crevices);
- Fuel condition (e.g. fuel with various oxyhydroxides of various thickness containing chemically-bound and adsorbed waters);
- Features of the fuel-in-canister system to be dried (e.g. debris on fuel, configuration of the fuel/canister).

4.1 Drying for non-sealed storage systems

Free water removal to avoid extensive corrosion (e.g. that would occur if aluminium fuel were to be in a canister with bulk free water) is the primary concern for non-sealed storage systems. Any hydrogen generation from corrosion or radiolytic reactions with the bound water in the oxyhydroxides would be vented.

A Fuel Drying Station has been developed [20] and is in use at INTEC for aluminium spent fuel to remove free water [21]. The aluminium fuel being stored at INTEC, in the Irradiated Fuel Storage Facility in CPP-603, is in a vented (non-sealed) dry storage system to mitigate potential issue with potential gas generation. Aluminium spent fuel has also been dry stored in other countries including Canada and Australia [22].

4.2 Drying for sealed storage systems

For sealed storage systems, a drying requirement for interim dry storage is based on avoiding hydrogen gas (H_2) buildup during storage in sealed canister. H_2 is generated through both corrosion reactions with water and through radiolytic reactions of radiation with oxyhydroxides.

Assuming the reaction goes to completion, 1 ml of H_2O yields 0.042 moles of H_2. The generation of hydrogen in the above mentioned reaction to produce $Al_2O_3{\bullet}H_2O$ bounds that for the reaction to produce $Al_2O_3{\bullet}3H_2O$ that occurs at temperatures below approximately 80°C. Relationships have been developed to relate the volume fraction of free water to the hydrogen pressure in a sealed canister volume for the reaction at completion [8].

Another impact of hydrogen buildup is the potential for production of UH_3, a compound that is pyrophoric under certain conditions. Dispersoids such as UAl_x would not be reduced by expected hydrogen pressures to produce UH_3; however, oxides of uranium could be if the partial pressure of H_2O is low enough and the partial pressure of H_2 is high enough in an H_2O/H_2 system. However, because aluminium surrounds the oxides in the dispersoid fuel and does not allow direct contact with the fuel particles, this is not expected to result in significant production of UH_3.

Considering the potential for existing UH_3 on uranium metal fuels retrieved from basin storage, only uranium metal fuels may need to be stabilized. A stabilization treatment with the technical bases has been developed [23] to convert UH_3 to uranium oxide. No significant amount of UH_3 is expected to be present on aluminium-based fuels [23]. In addition, the stringent drying requirements ensure that the expected H_2 buildup from corrosion is extremely low so that UH_3 production due to H_2 gas contacting exposed fuel meat is negligible.

It is also important in drying aluminium spent fuel to avoid very high drying temperatures. The drying temperature should not exceed ~250 °C to avoid the potential for hydrogen blistering and gross cladding failure.

5. CONCLUSIONS

Planning, development, and implementation of a system for final disposition of the spent nuclear fuel from many research reactor sites are not likely to be completed in near future. Wet and dry storage systems can be used for storage in the interim, with the imperative to limit fuel degradation during storage to enable its safe storage and retrieval for its ultimate disposition path. Aluminium spent nuclear fuel poses challenges in interim storage primarily due to it susceptibility to corrosion and related degradation phenomena.

For wet storage, the corrosion of aluminium can be minimized with controls for water quality and elimination of crevice conditions, stagnant water conditions, and galvanic couple conditions in the fuel storage system.

For dry storage, vented storage would allow for incomplete drying including any impact of the presence of oxyhydroxide films on aluminium that contain chemically-bound water. For sealed dry storage, consideration must be given to sufficiently dry the fuel to avoid additional corrosion with concomitant hydrogen generation and the impact on canister pressurization and ability for safe retrieval. Initial work has also indicated that hydrogen can be generated from the oxyhydroxides.

In the absence of corrosion-related phenomena, fuel deformation due to creep may become significant to impact retrievability at high fuel temperatures (e.g. > 200 °C for a 50-year storage period). Diffusion of radionuclides from the fuel through the aluminium cladding may also become significant at high cladding temperatures.

Continued storage that provides full retrievability for long term storage (e.g. > 50 years) will require vigilance in maintaining environmental limits of storage. Continued surveillance and evaluation of the fuel and storage system materials to verify the predictions of the impact of the environment on materials behaviour would also be needed to demonstrate reliable safe storage throughout the desired storage period.

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SPENT FUEL MANAGEMENT AND STORAGE AT THE FINNISH FIR TRIGA REACTOR

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Abstract

The FiR 1 reactor, a 250 kW TRIGA reactor, has been in operation since 1962. Presently spent fuel elements are stored in the racks on the walls of the reactor tank and in dry storage pits. After a sufficient cooling, the spent fuel elements are transferred from the reactor tank to the storage pits. The handling of fuel elements is made easier if the elements to be transferred have been stored for several years in the reactor tank. The capacity of the dry storage pits is sufficient for all the spent fuel elements of the reactor. After an eventual shut down of the reactor, the spent fuel temporary storage at site will only be required for quite a limited time if a decision is made to return the spent fuel to the USA. If we decide to continue operation of the reactor after the expiry date of the USDOE spent fuel acceptance policy (May 2016), the spent fuel can be incorporated into the domestic final repository for spent fuel of the Finnish Nuclear Power Plants. The final disposal facility is due to start operations in 2020.

1. INTRODUCTION

The FiR 1 reactor, a 250 kW TRIGA reactor, has been in operation since 1962. The main purpose of running the reactor is for Boron Neutron Capture Therapy (BNCT). The BNCT work dominates the current utilization of the reactor. The weekly schedule still allows one or two days for other purposes such as isotope production, neutron activation analysis and training. Figure 1 shows the BNCT Facility at the FiR 1 Reactor.

The current inventory of spent fuel comprises ten elements stored in the racks on the walls of the reactor tank and 13 elements stored in dry storage pits.

The storage pits were originally designed to be water filled, but are now used for dry storages where lead shielding is used instead of water, to provide the necessary radiation shielding. The handling of fuel elements is easier if the elements to be transferred have been decay stored for several years in the reactor tank.

The capacity of the dry storage pits is sufficient for all the spent fuel at the reactor**.** Upon an eventual shut down of the reactor the spent fuel temporary storage at site will only be required for quite a limited time if a decision is made to return the spent fuel to the USA. If we decide to continue operation of the reactor after the expiry date of the USDOE spent fuel acceptance policy (May 2016), the spent fuel can be incorporated into the domestic final repository for spent fuel of the Finnish Nuclear Power Plants. The final repository is due to start operations in 2020 and they will continue for tens of years. Even in this case the full capacity of the temporary storage pits will only be needed for a few years.

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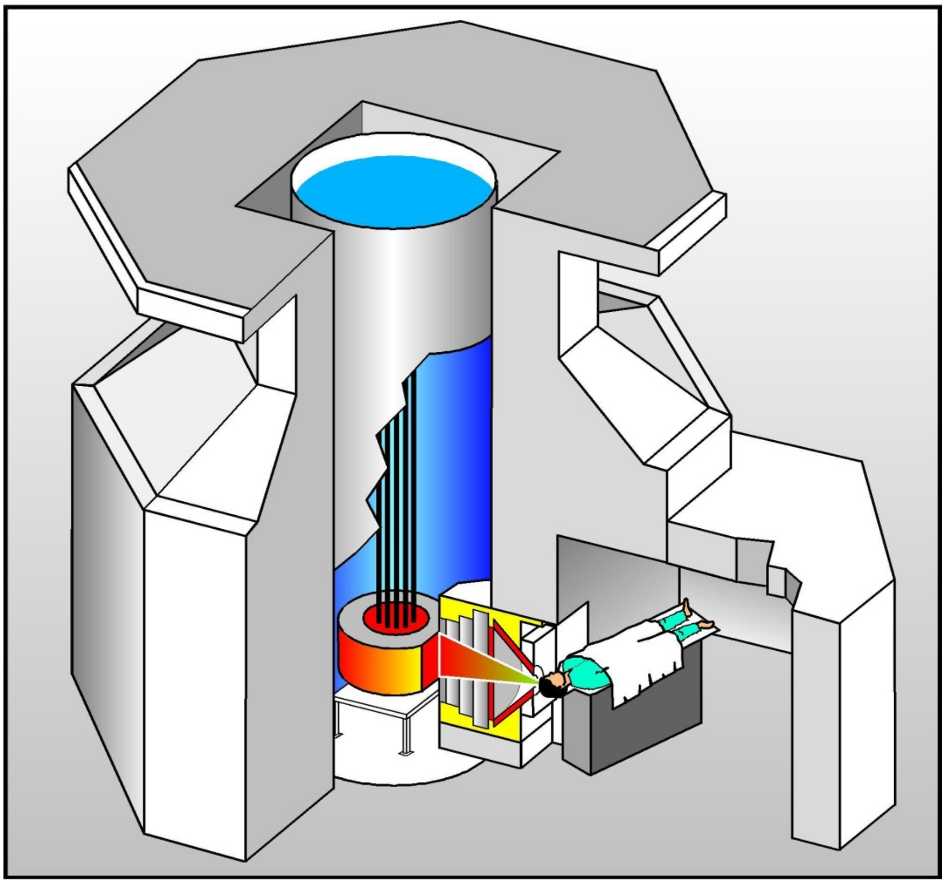

FIG. 1. BNCT Facility at the FiR 1 reactor.

2. FINAL DISPOSAL SOLUTION OF SPENT FUEL IN FINLAND

In 1995 the Finnish Nuclear Power Companies founded a separate company, named Posiva, to develop the technology and carry out safety analysis and site investigations for implementing the spent fuel final disposal. In 1999 Posiva submitted an application for a decision in principle for a final repository to be built at Olkiluoto, on the western coast of Finland. Olkiluoto is one of the two nuclear power plant sites in Finland. At the end of year 2000 the Finnish government approved the application and sent it to the parliament for ratification. The ratification took place in May 2001. Separate licenses still will be needed for the construction of the facility, scheduled to start in 2010, and for the operation, ten years later. The government alone will grant these licenses and no political aspects are supposed to be involved in the license process.

For final disposal, the spent nuclear fuel will be encapsulated in airtight copper canisters and incorporated into the bedrock at a depth of 500 m. The safety of this deep underground repository is based on multiple natural and engineered barriers. Each canister will contain 12 normal fuel assemblies from nuclear power plants. The present concept for TRIGA fuel elements is that the elements will be loaded in canisters with the same outer dimensions as the canisters that will be used for nuclear power plant fuel assemblies. This ensures that the TRIGA fuel can be easily handled in the final disposal facility and loaded in the heavy copper canisters. Figure 2 shows the final disposal canisters.

FIG. 2. Final disposal canisters.

3. SPENT FUEL SITUATION AT THE FIR 1 REACTOR

In Finland the research reactor must have a nuclear waste management plan, which must include, among others issues, considerations for the spent fuel management. The cost estimate of the nuclear waste management plan has to be updated annually and every fifth year the plan will be updated completely. The plan is based on the assumption that the final disposal site of the spent fuel will be at Olkiluoto. The final disposal facility is supposed to be operational by 2020.

For twenty years, we have had an agreement, in principle, about the possibility to use the final disposal facility of one of the Finnish nuclear power companies. Later this agreement was transferred to the joint nuclear waste management company Posiva.

The current operating license of our reactor will expire in 2011. A new license will be applied for in 2010. It is likely that there will be a delay between the final shut down of the reactor and the opening of the final disposal facility. Therefore, there has to be a sufficient storage capacity for the spent fuel at the reactor site for fuel awaiting transportation to the final disposal facility. So far, we have used the storage pits for dry storage, and since the renovation work of the reactor building in 1997, we have had sufficient storage capacity for all the fuel within the reactor building.

In addition to the domestic final disposal solution there is the option to return the spent fuel to the USA, under the USDOE take-back programme, which is available until 2016.

4. SPENT FUEL HANDLING AND STORAGE IN THE REACTOR BUILDING

4.1. Criticality safety

Criticality safety is a key issue, and must be ensured when planning a new storage configuration for the spent fuel. This is valid for the storage racks on the reactor tank walls, for the racks of the storage pits, and for the interim handling place for checking and drying the fuel elements. The basic assumption when performing the criticality analysis for a new configuration is that the storage is filled with water even in the case of dry storage.

4.2. Heat production of a spent fuel element

It is important to calculate the heat production of a spent fuel element especially, when the element is to be stored in a dry storage. In the case of a TRIGA fuel element with an average burnup of 6 %, which has been cooled for four years, the heat produced is around 1 W. This means that no special arrangement is needed to conduct the heat away from the fuel element in the dry storage.

4.3. Removing the fuel elements from the reactor tank

At present time there are 23 spent fuel elements, 10 elements stored in the racks on the walls of the reactor tank and 13 elements stored in the dry storage pits. Normally when one spent fuel element is removed permanently from the core and loaded to the fuel rack on the wall of the reactor tank it will stay for several years in the rack for cooling prior to removal. Allowing the spent fuel to cool for a sufficiently long period makes the transfer operations to the storage pit easier. A spent fuel element can now be lifted freely through the water and air to the nearby heavy transfer cask without immersing the heavy cask into the water of the reactor tank.

4.4. Interim handling place for drying and inspection of the spent fuel elements

Using the heavy transfer cask the fuel elements can be transferred to an interim handling place near the storage pits. The handling place is constructed using concrete blocks to provide sufficient radiation shielding around the fuel elements. The fuel elements can be left in the handling place for one or two weeks to dry. Visual checking of the fuel elements can be made at the same time through a lead glass window.

4.5. Dry storage pits for spent fuel elements

The storage pits were originally designed to be water filled. The pits are 3 m deep and the diameter is 30 cm. There are together 12 pits in total, and only two of them are occupied with spent fuel, all the others are empty. The pits are now used for dry storage, where lead provides the necessary radiation shielding. Twelve fuel elements fill one single layer rack in the storage pit. The elements are in vertical positions in the rack. The storage pits are connected to the outlet of the ventilation system. The covers of the pits can be sealed.

4.6. Preparation of a damaged fuel element for dry storage

A damaged fuel element, with a clad rupture, was first stored on the wall of the reactor tank for four years for cooling. The element was then dried, and put into an airtight capsule. In order to remove internally bound water, the storage capsule was connected to a vacuum pump, and vacuum dried at 1 mbar vacuum for three days. After this final drying the capsule was closed tightly and put in the dry storage pit.

5. CONCLUSIONS

The options available for the management of FiR TRIGA spent fuel are either to utilise the domestic final disposal solution or the USDOE return policy. In either case, the time for interim storage at reactor is relatively short. The use of the dry storage option is reasonable, and preferred when compared to the water filled storage option, which demands water circulation and purification systems. The downside is that dry storage option makes item counting and visual inspections more difficult to arrange.

RESEARCH REACTOR SPENT NUCLEAR FUEL, NATIONAL PRACTICE FOR INTERIM STORAGE IN THE NETHERLANDS

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Abstract

In the Netherlands radioactive waste is managed by COVRA, the Central Organisation for Radioactive Waste. COVRA was set up in 1982 as a joint initiative of the government and the larger waste producers at that time. COVRA N.V. is a private company with limited liability. The statutory task of COVRA is to take care of all kinds and categories of radioactive waste in the Netherlands on the basis of governmental policy. The shares of the company are held by the State of the Netherlands, i.e. the Ministry of Finance. In the national management system for radioactive waste in the Netherlands an integral approach has been chosen to process all kinds and categories of waste. For the small volume, but broad spectrum of radioactive waste, a management system has been developed based on the principle to isolate, to control, and to monitor the waste. Long term storage is an important element in this management strategy. The policy implies that there is one integral solution for all kinds of radioactive waste, regardless of its origin and composition. The decision to store for a long time was well considered and was not taken as a 'wait and see' option. This is clearly demonstrated by the fact that integral parts of the policy are: the establishment of a capital growth fund and a clear choice for the ownership of the waste, i.e. a complete transfer of liability to COVRA. This policy does not leave an undue burden of waste generated today to future generations. Only the execution of the disposal is left as a task for the future. All the waste will be kept at one site, well isolated from the environment, well controlled and well monitored. The waste is fully retrievable, a disposal solution is available in principle (geological disposition), and the money needed will become available in the capital growth fund. Also new disposal options, multilateral solutions or even complete new techniques can be applied when available and feasible.

1. INTRODUCTION

In the Netherlands one nuclear power plant, two nuclear research centres, a uranium enrichment plant and a medical isotope production facility are in operation. In addition, there is a widespread use of radioactive materials in other areas. The nuclear power plant Borssele (PWR, 450 MWe) is in operation since 1973 and is scheduled to remain operational until 2033.

The annually produced quantity of radioactive waste in the Netherlands is small and very heterogeneous. All radioactive waste is managed by COVRA, the Central Organisation for Radioactive Waste. Its task is to execute the policy of the government. The policy lays down that all radioactive waste will be stored above ground in engineered structures allowing retrieval at all times for a period of at least 100 years. After this period of long term storage final disposal is foreseen. The policy is based on a step-wise decision process in which all decisions are taken to ensure safe disposal in a repository, but without excluding unforeseen alternative solutions in the future.

Figure 1 shows the COVRA facilities. Buildings for the treatment and storage of all categories of waste have been constructed, commissioned and are now in full operation. COVRA has a site available of about 25 ha at the industrial area Vlissingen-Oost. There is a facility for the treatment of low and medium level waste as well as a modular storage building for the cemented waste packages. A separate storage building is available for the very low level radioactive waste from ore processing industries. This calcined waste product is stored in 20 ft standard size containers. Depleted uranium is stored in another building; this material is stored as the uranium oxide in DV70 containers. The storage facility for high level waste, mainly spent research reactor fuel and reprocessing waste was commissioned in 2003 and operates to full satisfaction. Information on the siting process, licensing, construction and practical experience in the Netherlands can be found in the literature and in the NEWMDB[7] of the IAEA [1–6].

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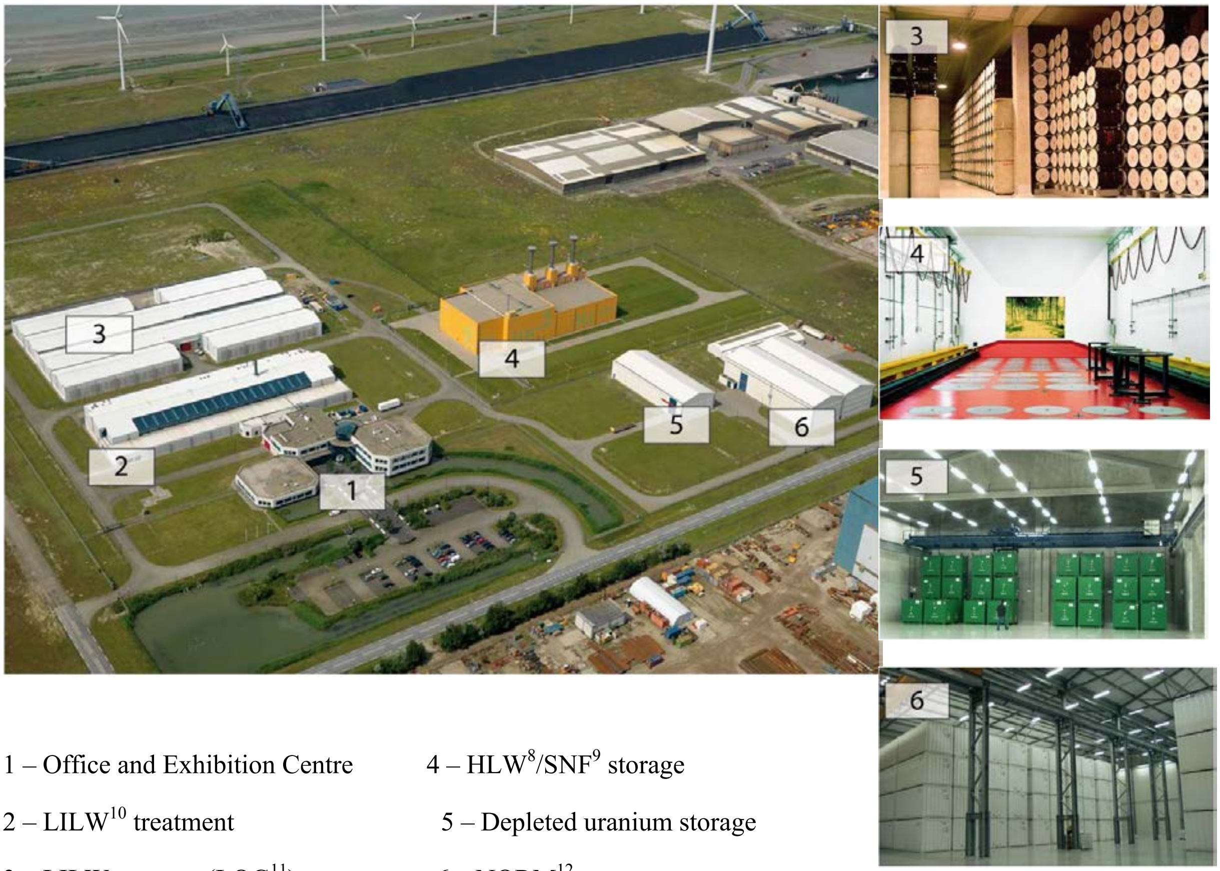
[6] COVRA: Centrale Organisatie Voor Radioactief Afval (Central Organisation for Radioactive Waste).

[7] NEWMDB: Net-enabled Waste Management Database of IAEA.



1 – Office and Exhibition Centre 4 – HLW[8]/SNF[9] storage

2 – LILW[10] treatment 5 – Depleted uranium storage

3 – LILW storage (LOG[11]) 6 – NORM[12] storage

FIG. 1. The COVRA facilities.

2. THE SITUATION OF RRSNF IN THE NETHERLANDS

2.1. National legislation and legal issues

The following are the main laws to which nuclear installations are subject in the Netherlands:

- The Nuclear Energy Act (1963, as amended 2004); (Kew[13]);
- The Environmental Protection Act (1979, as amended 2002); (Wm[14]);
- General Administrative Law Act (1992, as amended 2003); (Awb[15]).

The basic legislation governing nuclear activities is contained in the **Nuclear Energy Act**. The Nuclear Energy Act has historically been designed to encourage the use of nuclear energy and radioactive techniques, as well as to lay down rules for protection of the public and workers against the risks. The Act sets out the basic rules on nuclear energy, makes provisions for radiation protection, designates the various competent authorities and outlines their responsibilities.

Licences for nuclear facilities are granted jointly by the Minister of Housing, Spatial Planning and the Environment, the Minister of Economic Affairs, and the Minister of Social Affairs and Employment (plus, where relevant, some other ministers whose departments may be involved). Together, these

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[8] HLW: High Level Waste.
[9] SNF: Spent Nuclear Fuel.
[10] LILW – Low and Intermediate Level Waste.
[11] LOG: Laagradioactief afval Opslag Gebouw (Low-level Waste Storage Building).
[12] NORM: Naturally Occurring Radioactive Material.
[13] Kew: Kernenergiewet (Nuclear Energy Act).
[14] Wm: Wet Milieubeheer (Environmental Protection Act).
[15] Awb: acronym for Algemene wet bestuursrecht (General Administrative Law Act).

ministers form the competent authorities as defined by the Nuclear Energy Act and are jointly responsible for assessing the licence applications and granting the licences. The Minister of Housing, Spatial Planning and the Environment acts as the coordinator.

With regard to nuclear energy, the purpose of the Act is to regulate (Article 15b):

- The protection of people, animals, plants and property;
- The security of the State;
- The storage and safeguarding of fissionable materials and ores;
- The supply of energy;
- The payment of compensation for any damage or injury caused to third parties;
- The observance of international obligations.

A number of decrees have also been issued containing additional regulations. The most important of these in relation to the safety aspects of nuclear installations are:

- The Nuclear Installations, Fissionable Materials and Ores Decree (Bkse[16]);
- The Radiation Protection Decree (Bs[17]);
- The Transport of Fissionable Materials, Ores, and radioactive Substances Decree (Bvser[18]).

The Nuclear Installations, Fissionable Materials and Ores Decree regulates all activities (including licensing) that involve fissionable materials and nuclear installations. The Radiation Protection Decree regulates the protection of the public and workers against the hazards of all ionising radiation. It also establishes a licensing system for the use of radioactive materials and radiation emitting devices, and prescribes general rules for their use. The Transport of Fissionable Materials, Ores and Radioactive Substances Decree deals with the import, export and inland transport of fissionable materials, ores and radioactive substances by means of a reporting and licensing system.

The Nuclear Energy Act and the above mentioned decrees are fully in compliance with the relevant Euratom Directive laying down the basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionising radiation. This Directive (96/29/Euratom) is incorporated in the relevant Dutch regulations.

The **Environmental Protection Act**, in conjunction with the Environmental Impact Assessment Decree, stipulates (in compliance with EU Council Directive 97/11/EC; see also the section on Article 8) that an Environmental Impact Assessment must be presented when an application is submitted for a licence for a nuclear installation.

In the case of non-nuclear installations, this Act regulates all environmental issues (e.g. chemical substances, stench and noise); in the case of nuclear installations, the Nuclear Energy Act takes precedence and regulates both conventional and non-conventional environmental issues.

The **General Administrative Law Act** sets out the procedure for obtaining a licence, and also describes the role played by the general public in this procedure (i.e. objections and appeals).

For additional information see also the text under Article 4 (iv).

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[16] Bkse: Besluit Kerninstallaties, Splijtstoffen en Ertsen (Nuclear Installations, Fissionable Materials and Ores Decree).

[17] Bs: Besluit Stralingsbescherming (Radiation Protection Decree).

[18] Bvser: Besluit Vervoer Splijtstoffen, Ertsen en Radioactieve stoffen (Transport of Fissionable Materials, Ores, and Radioactive Substances Decree).

2.1.1. Main elements of the Acts and Decrees

2.1.1.1. Nuclear Energy Act (Kew)

Within the framework of the Nuclear Energy Act, fissionable materials are defined as materials containing up to a certain percentage of uranium, plutonium or thorium (i.e. 0.1% uranium or plutonium and 3% thorium by weight) and used for purposes of fission or breeding. All other materials are defined as radioactive materials.

As far as nuclear installations are concerned, the Nuclear Energy Act covers three distinct areas relating to the handling of fissionable materials and ores: (a) registration, (b) transport and management of such materials, and (c) the operation of sites at which these materials are stored, used or processed.

The registration of fissionable materials and ores is regulated in Sections 13 and 14 of the Nuclear Energy Act; further details are given in a special Decree issued on 8 October 1969 (Bulletin of Acts and Decrees 471). The statutory rules include a reporting requirement under which notice must be given of the presence of stocks of fissionable materials and ores. The Central Import and Export Office, part of the Tax and Customs Administration of the Ministry of Finance, is responsible for maintaining the register.

A license is required in order to transport, import, export, be in possession of or dispose of fissionable materials and ores. This is specified in Section 15a of the Act. The licensing requirements apply to each specific activity mentioned here.

Licenses are also required for building, commissioning, operating, modifying or decommissioning nuclear installations (Section 15b), as well as for nuclear driven ships (Section 15c). To date, the latter category has not been of any practical significance.

Under item (c), the Nuclear Energy Act distinguishes between construction licences and operating licences. In theory, a licence to build a plant may be issued separately from any licence to actually operate it. However, the construction of a nuclear power plant involves much more than simply building work. Account must be taken of all activities to be conducted in the plant. This means that the government needs to decide whether the location, design and construction of the plant are such as to afford sufficient protection from any danger, damage or nuisance associated with the activities that are to be conducted there. In practice, therefore, the procedure for issuing a licence to operate a nuclear power plant will be of limited scope, unless major differences have arisen between the beginning and the completion of construction work. For example, there may be a considerable difference between the Preliminary Safety Analysis Report (which provides the basis for the construction licence) and the Final Safety Analysis Report (for the operating licence). Views on matters of environmental protection may also have changed over the intervening period.

Amendments to a licence will be needed where modifications of a plant invalidate the earlier description of it.

The decommissioning of nuclear installations is regarded as a special form of modification and is treated in a similar way. In 2002 the Nuclear Installations, Fissionable Materials and Ores Decree (Bkse) was amended to meet the requirements set by Council Directive 96/29/Euratom with regard to the protection of workers and members of the public from the hazards of ionising radiation. The Directive had introduced a new licence requirement for the shutdown and decommissioning of nuclear installations. The amendment of Bkse had the effect of incorporating these regulations in Dutch legislation.

Where modifications are only minor, the licensee may make use of a special provision in the Act (Section 18) that allows such modifications to be made without amendment of the licence. In such cases, the licensee need only submit a notification describing the planned modification. This notification system can be used only if the consequences of the modification for man and environment are within the limits of the licence in force.

Licences for nuclear installations are issued under the joint responsibility of the Minister of Housing, Spatial Planning and the Environment, the Minister of Economic Affairs and the Minister of Social Affairs and Employment (plus other ministers, where relevant).

Bkse sets out additional regulations in relation to a number of areas, including the licence application procedure and associated requirements. Applicants are required to supply the following information:

- A description of the site where the plant is to be located, including a statement of all relevant geographical, geological, climatological and other conditions;
- A description of the plant, including the equipment to be used in it, the mode of operation of the plant and the equipment, a list of the names of the suppliers of those components which have a bearing on the assessment of the safety aspects, and a specification of the plant's maximum thermal power;
- A statement of the chemical and physical condition, the shape, the content and the degree of enrichment of the fissionable materials which are to be used in the plant, specifying the maximum quantities of the various fissionable materials that will be present in the plant at any one time;
- A description of the way in which the applicant intends to dispose of the relevant fissionable materials after their use;
- A description of the measures to be taken either by or on behalf of the applicant so as to prevent harm or detriment or to reduce the risk of harm or detriment, including measures to prevent any harm or detriment caused outside the plant during normal operation, and to prevent any harm or detriment arising from the Postulated Initiating Events (PIEs) referred to in the description, as well as a radiological accident analysis concerning the harm or detriment likely to be caused outside the installation as a result of those events (Safety Analysis Report);
- A risk analysis concerning the harm or detriment likely to be caused outside the installation as a result of severe accidents (Probabilistic Safety Analyses);
- A global description of plans for eventual decommissioning and its funding.

In addition to these regulations on the handling of fissionable materials, the Nuclear Energy Act includes a separate chapter (Chapter VI) on intervention and emergency planning and response.

2.1.1.2. Environmental Protection Act (Wm)

In compliance with this Act and the Environmental Impact Assessment Decree, the licensing procedure for the construction of a nuclear plant includes a requirement to draft an environmental impact assessment. In certain circumstances, an environmental impact assessment is also required if an existing plant is modified. More specifically, it is required in situations involving:

- A change in the type, quantity or enrichment of the fuel used;
- An increase in the release of radioactive effluents;
- An increase in the on-site storage of spent fuel;
- Decommissioning;
- Any change in the conceptual safety design of the plant that is not covered by the description of the design in the safety analysis report.

The Environmental Protection Act states that an independent Commission for Environmental Impact Assessments must be established and its advice must be sought whenever it is decided that an environmental impact assessment needs to be submitted by anyone applying for a licence. The regulations based on this Act stipulate the type of activities for which such assessments are required.

The general public and interest groups often use environmental impact assessments as a means of commenting on and raising objections to decisions on nuclear activities. This clearly demonstrates the value of these documents in facilitating public debate and involvement.

2.1.1.3. General Administrative Act (Awb)

Notice must be given in the Government Gazette (digital form) of the publication of a draft decision to award a license to a plant as defined by the Convention. At the same time, copies of the draft decision and of the documents submitted by the applicant must be made available for inspection by the general public.

All members of the public are free to lodge written objections to the draft decision and to ask for a hearing to be held under the terms of the General Administrative Act. Any objections made to the draft version of the decision are taken into account in the final version. Anybody who has objected to the draft decision is free to appeal to the Council of State (the highest administrative court in the Netherlands) against the decision by which the licence is eventually granted, amended or withdrawn. If the appellant asks the court at the same time for provisional relief (i.e. a suspension of the licence), the Decree (i.e. the licence) will not take effect until the court has reached a decision on the request for suspension.

3. INSTITUTIONAL AND POLITICAL ASPECTS

3.1. General

Article 58 of the Nuclear Energy Act states that the Ministers responsible for licensing procedures should entrust designated officials with the task of performing assessment, inspection and enforcement. The Decree on Supervision identifies the bodies that have responsibilities in this connection. Since 1 March 2004 the national regulatory body for supervision of Dutch nuclear installations is the Nuclear Safety Service (KFD[19]) of the Inspectorate of the Ministry of Housing, Spatial Planning and the Environment (VI[20]: VROM[21] Inspectorate).

A separate section of the KFD is responsible for supervision of nuclear security and safeguards (NBS[22]). At the same ministry, the Chemicals, Waste and Radiation Protection Directorate (SAS) is responsible for assessing whether the radiological safety objectives have been met. It should be noted that this directorate is responsible for policymaking and licensing, and does not perform inspections. SAS[23] has also responsibility for the implementation of international regulations and guidelines in the national legislation and for any other adjustments of the regulations deemed necessary.

With regard to nuclear fuel cycle installations and nuclear power plants in particular, almost all inspection tasks are carried out by the KFD, which possesses the technical expertise needed for the inspection of nuclear safety, radiation protection, security and safeguards.

3.2. Regulatory assessment

The regulatory body reviews and assesses the documentation submitted by the applicant.

This might be the Environmental Impact Assessment Report and Safety Report with underlying safety analyses within the framework of a licence renewal or modification request, proposals for design changes, changes to Technical Specifications, etc.

The KFD assesses whether the NVR[24] (i.e. requirements and guidelines for nuclear safety and environment), BRK93[25] (requirements and guidelines for security) and regulation for non-nuclear

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[19] KFD: Kernfysische Dienst (Nuclear Safety Service - the Netherlands).

[20] VI: VROM Inspectie (VROM Inspection).

[21] VROM: Volkshuisvesting, Ruimtelijke Ordening en Milieubeheer ((Ministry of) Housing, Spatial Planning and the Environment).

[22] NBS: Nucleaire Beveiliging & Safeguards (Nuclear Security and Safeguards).

[23] SAS: Stoffen, Afvalstoffen en Straling (Directorate for) Chemicals, Waste and Radiation Protection - the Netherlands).

[24] NVR: Nucleaire Veiligheids-Richtlijn (Nuclear safety rule - the Netherlands).

[25] BRK93: Beveiligingsrichtlijnen kerninstallaties (Requirements and guidelines for security and regulation for non-nuclear environment protection).

environment protection have been met and whether the assessments (methods and input data) have been prepared according to the state of the art etc. SAS assesses the waste and radiation safety aspects of spent fuel or radioactive waste management facilities.

3.3. Regulatory inspections

The function of regulatory inspections is:

- To check that the licensee is acting in compliance with the regulations and conditions set out in the law, the license, the safety analysis report, the Technical Specifications and any self-imposed requirements;
- To report any violation of the license conditions and if necessary to initiate enforcement action;
- To check that the licensee is conducting its activities in accordance with its quality assurance system;
- To check that the licensee is conducting its activities in accordance with the best technical means and/or accepted industry standards.

All inspections with regard to nuclear safety, radiological protection of personnel and of the environment around nuclear sites, security and safeguards, including transportation of fresh and spent nuclear fuel and related radioactive waste to and from nuclear installations are carried out by the KFD.

To check that the licensee is acting in compliance with the Nuclear Energy Act, the licence and the associated safety analysis report, there is a system of inspections, audits, assessment of operational reports, and evaluation of operational occurrences and incidents. An important piece of information for inspection is the safety evaluation report, conducted at 2–5 years periods. In this report the licensee presents its self-assessment of all the relevant technical, organizational, personnel and administrative matters. Every ten years a major assessment of the accomplishments in the area of safety and radiation protection is performed by the staff of the spent fuel and radioactive waste management facility and compared with new developments.

The management of inspection is supported by a yearly planning, the reporting of the inspections and the follow-up actions. On an annual or quarterly basis, dependent on the type of facility, a meeting between facility management and KFD management is held devoted to inspections and inspection findings, during which any necessary remedial actions are established and the progress made with their execution discussed.

4. MANAGEMENT OF RRSNF IN THE NETHERLANDS

4.1. Research reactors in the Netherlands

In the Netherlands there are three operational research reactors, High Flux Reactor (HFR), HOR and LFR. Two other research reactors were built in the Netherlands and have already been decommissioned, the Biological Agricultural Reactor Netherlands (BARN) and KEMA Suspension Test Reactor (KSTR). BARN, a 100 W pool type reactor reached first criticality on April 1963, was shut down on January 1980 and its dismantling was completed in 1999. KSTR, a 1 MW aqueous breeder type research reactor reached first criticality on May 1974, was shut down on May 1977 and had its site returned to “Greenfield” in 2003 [7].

A 6th reactor is being planned in the Netherlands, Pallas, which is being designed by NRG as the replacement reactor for HFR. The official target date for Pallas is 2016, a probable date in which HFR will be stopped.

High Flux Reactor (HFR) is the most prominent nuclear facility in the Netherlands. Located in Petten, it is owned by the Joint Research Center (JRC) of the European Union in Petten, and operated by Nuclear Research and Consultancy Group (NRG), the licensee. It is a tank in pool type reactor with nominal thermal power of 45 MW, and reached criticality for the first time in November of 1961. In the beginning the reactor was primarily used for research of materials, focussed in research of resistance of materials to neutron irradiation, especially materials for fusion and fission reactors.

Together with a set of hot cells available at the Petten site, HFR provided for over four decades, irradiation and post-irradiation examination services, using its 20 in-core and 12 poolside irradiation positions, plus 12 horizontal beam tubes, having registered a recognized record of consistency, reliability and high availability. Irradiation and post irradiations examinations was a traditional activity and service available for users of HFR, who used the reactor in the 1970's and 1980's to perform extensive tests on the behaviour of nuclear fuel (both FBR and LWR) under accident conditions like overpower tests, loss of coolant accident simulation tests and iodine redistribution tests [8]. Despite the decrease in this type of tests, HFR has never decreased its operational programme. With the increasing market of radioisotopes for nuclear medicine, HFR started to be a major radioisotope supplier, and actually it is an indispensable facility for the production of radio-isotopes for the medical sectors, covering about 60% of European demand. HFR was originally fuelled with highly enriched uranium (HEU) (UAlx) fuel, in 2005 it started a conversion process, to start using LEU fuel. In October of that year the first LEU fuel elements were inserted in the core, and in May 2006 the conversion process, from HEU UAlx fuel to LEU U3Si2 fuel was completed [8]. So far all the LEU silicide fuel has been kept stored in the reactor facility. First shipment is being planned for 2012.

Technical University Delft's 'Hoger Onderwijs Reactor' (HOR/Higher Education Reactor) is a 2 MWth pool type research reactor, situated in the Interfaculty Reactor Institute in Delft. The reactor reached criticality for the first time in April 1964, and is used as a scientific facility for academic research and training; especially to supply neutrons for use in neutron scattering experiments and neutron activation analysis. HOR is an open pool-type research reactor originally fuelled with 93% enriched fuel MTR-fuel assemblies, In 1980 researchers started feasibility studies for the conversion of the reactor, from UAlx-Al HEU to U3Si2-Al LEU. Physical conversion started in 1998, when the first LEU fuel elements were inserted into the core, and finished in 2004. The conversion process resulted in a more compact core. The pre-conversion core had 32 HEU fuel elements and the converted core had only 20 LEU fuel elements, plus four additional control assemblies, and surrounded by beryllium reflector elements on three sides. As for HFR, all LEU silicide fuel used in the reactor core has been stored in the reactor facility.

The Low Flux Reactor (LFR) of the Netherlands Energy Research Foundation (ECN) in Petten, is a small Argonaut type research reactor. The reactor that operates at a power up to 30 kW, consists basically of a ring-shaped core surrounded by an inner cylindrical graphite reflector and an outer graphite reflector. The core contains ten fuel elements arranged in two rows of 5 in an asymmetric configuration and cooled by light water [9]. Due to its low power, the facility still has not produced spent fuel.

4.2. Facilities to handle spent fuel in the Netherlands

In the Netherlands the following spent fuel management facilities can be distinguished.

Borssele	Dry storage in vaults	- COVRA facility for treatment and storage of HLW and SF (HABOG[26])
Borsssele	Fuel storage pond	- Pond associated with nuclear power station where spent fuel is stored temporarily before shipment to La Hague for reprocessing
Petten	Fuel storage pond	- Belongs to the HFR[27] research reactor; fuel is stored temporarily awaiting shipment to COVRA
	Dry storage in drums	- NRG[28] Waste Storage Facility; spent fuel samples from HFR irradiation experiments; stored in concrete-lined vaults
Delft	Fuel storage pond	- Belongs to HOR[29] research reactor.

The “production” of spent fuel is 2 spent fuel assemblies per year for Delft and 66 assemblies for Petten. This equals in average 1 canister for transport to COVRA every 16 years for the Delft reactor and 2 canisters per year for transport to COVRA for the Petten reactor.

Table I gives an overview of the research reactor SNF storage capacity.

TABLE I. OVERVIEW OF STORAGE CAPACITY FOR RESEARCH REACTOR SNF

Facility	Type of fuel	Stored (2009)	Maximum capacity
COVRA	Spent fuel	23 canisters, 33 assemblies each	70 canisters, can be extended to 175 canisters
	Uranium filters	3 canisters, 99 filters each	
Petten	Spent fuel/control assemblies HEU[a]	About 20 assemblies	About 500[c] assemblies
	Spent fuel/control assemblies LEU[b]	About 200 assemblies	
Delft	Spent fuel/control assemblies HEU	About 20 assemblies	About 100 assemblies
	Spent fuel/control assemblies LEU	About 30 assemblies	

[a] HEU: Highly Enriched Uranium [b] LEU: Low Enriched Uranium [c] Limited by license

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[26] HABOG: Hoogradioactief AfvalBehandelings- en Opslag Gebouw (High-level Waste Treatment and Storage Building).

[27] HFR: High Flux Reactopr – Research Reactor of JRC at Petten.

[28] NRG: Nuclear Research and Consulting Group – The Netherlands.

[29] HOR: Hoger Onderwijs Reactor – Research Reactor at the Technical University Delft.

5. NATIONAL PRACTICES

5.1. Policy

The Netherlands' policy on radioactive waste management is based on a report that was presented to parliament by the Government in 1984. This report covered two areas. The first concerned the long term interim storage of all radioactive wastes generated in the Netherlands, and the second concerned the Government research strategy for eventual disposal of these wastes.

Consideration of this report led, in regard to the first area, to establishment of the Central Organisation for Radioactive Waste (COVRA) in Borssele, and in regard to the second, to establishment of a research programme on disposal of radioactive waste.

Pending the outcome of research into disposal, and assurance of political and public acceptance, it was decided to construct an engineered surface-storage facility with sufficient capacity for all the radioactive wastes generated in a period of at least 100 years.

5.2. Long term storage

The policy in the Netherlands is that all hazardous and radioactive waste must be isolated, controlled and monitored. In principle this can be done by storage in buildings and institutional control. It can also be achieved by shallow land burial and maintenance of a system of long term institutional control, or by deep geologic disposal, in which case institutional control is likely to be discontinued at some moment. For the options mentioned the degree of institutional control is the highest for storage in buildings and the lowest for deep disposal. When containment is required over periods of time longer than the existence of society, doubt may be raised on the capacity of society to fulfil the control requirement.

The Netherlands has a very high ground water table and under these circumstances shallow land burial is not acceptable for the low and medium level waste. As a consequence deep geologic disposal will be required for all waste categories as a final solution under the assumption that disposal is the preferred management option.

Also it should be realized that the cumulative waste volume that is actually in storage right now is approximately 10.000 thousand m^3 (excluding temporary shielding). For such a small volume it is not economically feasible to construct a deep geologic disposal facility. The waste volume collected in a period of 100 years can be judged as large enough to make a disposal facility viable. So a period of at least 100 years of storage in buildings will be required. This creates at least six positive effects:

- Public acceptance is quite high for long term storage. The general public has more confidence in physical control by today's society than in long term risk calculations for repositories even when the outcome of the latter is a negligible risk.
- There is a period of 100 years available to allow the money in the capital growth fund to grow to the desired level. This brings the financial burden for today's waste to an acceptable level.
- During the next 100 years an international or regional solution may become available. For most countries the total volume of radioactive waste is small. Cooperation creates financial benefits, could result in a higher safety standard and a more reliable control.
- In the period of 100 years the heat generating waste will cool down to a situation where cooling is no longer required.
- A substantial volume of the waste will decay to a non-radioactive level in 100 years.
- A little bit more than 100 years ago, mankind was not even aware of the existence of radioactivity. In 100 years from now new techniques or management options can become available.

Consequently, it was concluded in the policy report of 1984 that a dedicated solution for the Netherlands is to store the waste in buildings for a period of at least 100 years and to prepare financially, technically and socially the deep disposal during this period in such a way that it can really be implemented after the storage period. Of course at that time society has the freedom of choice between a continuation of the storage for another 100 years or to realize the final disposal.

5.3. Disposal of radioactive waste

The geological conditions in the Netherlands are in principle favourable from the perspective of disposal of radioactive waste. In the northern part of the country there are deep lying, large salt formations with a good potential as disposal site. Clay formations are ubiquitous at varying depth in the whole country. Extensions of the Boom clay, which qualifies as potentially suitable host rock for a repository in Belgium, also abound in the south west of the Netherlands (see Figs 2 and 3).

In 1993 a radioactive waste disposal research programme was completed, and it was concluded that there are no safety-related issues that would prevent the deep underground disposal of radioactive waste in salt. However, the level of public acceptance of underground waste disposal remained low. Progress of the disposal programme was stalled by lack of approval for site investigations in salt formations considered suitable for this purpose and, hence, the prospect of a waste disposal facility being available within the next few decades is remote. In 1993 the government adopted, and presented to parliament, a position paper on the long term underground disposal of radioactive and other highly toxic wastes. This forms the basis for further development of a national radioactive waste management policy, which now requires that any underground disposal facility be designed in such a way that each step of the process is reversible. This means that retrieval of waste, if deemed necessary for whatever reason, would always be possible.

The reasons for introducing the concept of retrievability came from considerations of sustainable development. Waste is considered a non-sustainable commodity whose generation should be prevented. If prevention is not possible, the preferred option is to reuse and/or recycle it. If this in turn is not practical at present, disposal of the waste in a retrievable way will enable future generations to make their own decisions about its eventual management. This could include the application of more sustainable management options if such technologies become available. The retrievable emplacement of the waste deep underground would ensure a fail-safe situation in case of neglect or social disruption.

Although waste retrievability allows future generations to make their own choices, it is dependent upon the technical ability and preparedness of society to keep the facility accessible for inspection and monitoring over a long period. It also entails a greater risk of exposure to radiation and requires long term arrangements for maintenance, data management, monitoring and supervision. Furthermore, provision of retrievability in disposal deep underground is likely to make the construction and operation more complex and costly.

5.4. Treatment and storage of RRSNF

The research reactors in the Netherlands used HEU fuel with enrichment up to 93% ^{235}U. In the past the RRSNF was sent back to the country of origin, the United States of America. After the USA refused to take back the RRSNF, unless the research reactors switched to LEU, the reactor managers agreed to store the HEU assemblies temporarily in the reactor pool and decided to take part in the construction of the HABOG facility at the COVRA site. As shown in Figs 4 and 5, the spent fuel assemblies were loaded (wet or dry) and transported in a Castor MTR2 cask to the COVRA site. In the MTR2 cask 33 spent fuel assemblies are placed in a basket.

During the construction phase of the HABOG facility the MTR2 casks were temporarily stored in the storage facility for LILW. After finishing the construction of the HABOG facility and the respective commissioning phase, the facility went into active operation at the end of 2003.

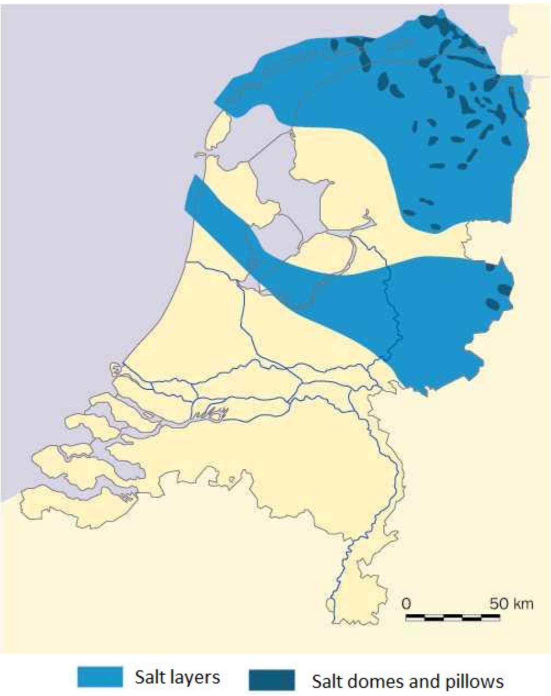

FIG. 2. *Distribution of salt formations.*

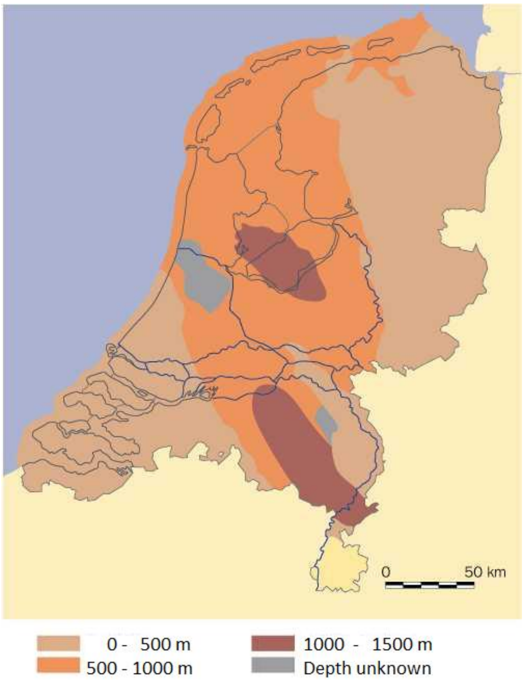

FIG. 3. Distribution and depth of the Boom Clay.

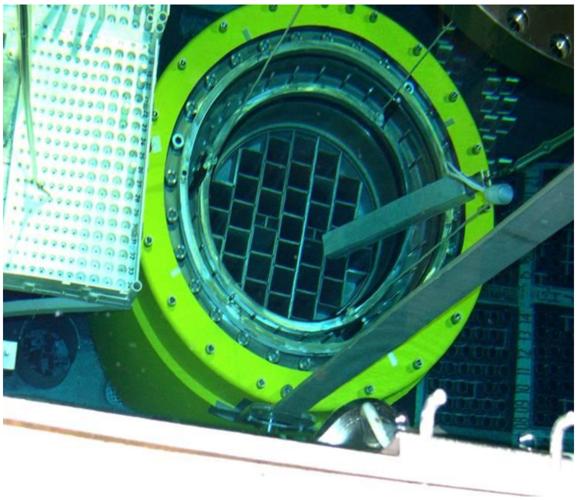

FIG. 4. Wet loading.

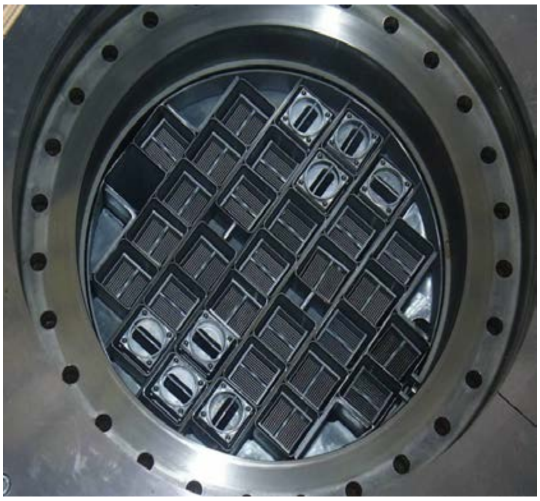

FIG. 5. Loaded basket in cask.

In the HABOG facility the basket is unloaded from the MTR2 cask (Fig. 6), placed into a stainless steel canister (Fig. 8) and welded tight (Fig. 7). The canister is filled with the inert Helium gas up to 1.54 bar and checked for leakage. The leakage criterion is 10^{-6} mbar.l.s^{-1}. After checking for contamination the canister is placed in a double jacket stainless steel well in a vault. The well is filled with the inert Argon gas. A storage vault is suitable for the storage heat-generating waste coming from reprocessing of spent nuclear fuel from commercial reactors and RRSNF. The cooling of the vaults is natural cooling so complete independent of mechanical systems.

FIG. 6. Basket to welding station.

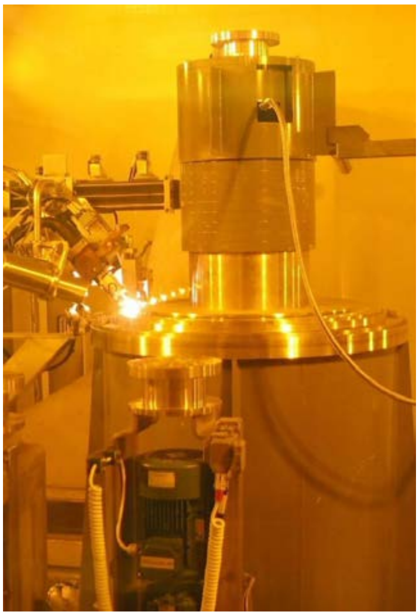

FIG. 7. Lid welding.

FIG. 8. Canister.

6. NATIONAL FACILITIES AND CAPABILITIES

As already mentioned in the introduction, all radioactive waste is managed by COVRA, the Central Organisation for Radioactive Waste. Its task is to execute the policy of the government. The policy lays down that all radioactive waste will be stored above ground in engineered structures allowing retrieval at all times for a period of at least 100 years. After this period of long term storage final disposal is foreseen. The policy is based on a step-wise decision process in which all decisions are taken to ensure safe disposal in a repository, but without excluding unforeseen alternative solutions in the future. As shown in Figs 9 and 10, buildings for the treatment and storage of all categories of waste have been constructed, commissioned and are now in full operation. Table II shows the current and maximum capacity of each facility.

FIG. 9. The COVRA facilities.

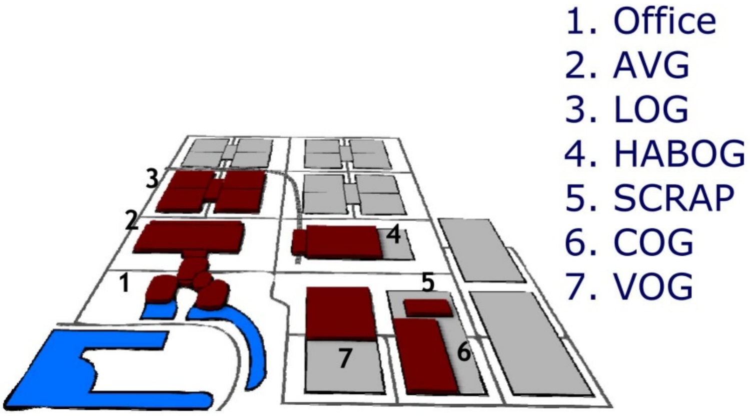

FIG. 10. The COVRA facilities[30] *and possible extensions (grey colour).*

TABLE II. STORAGE CAPACITY OF COVRA SITE

Storage building	Type of waste	Current capacity m^3	Maximum capacity m^3
LOG	LILW	35000	140000
HABOG	RRSNF and vitrified HLW	100	250
	Non-heat generating HLW	750	2000
COG	TENORM[31]	7000 (336 20 ft containers, 30 ton each)	8000 (408 20 ft containers, 30 ton each)
VOG	Depleted uranium	7000 (1950 containers , 11 ton U_3O_8 each)	14000 (3900 containers , 11 ton U_3O_8 each)

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[30] AVG: Afvalverwerkingsgebouw (LILW treatment facility).
COG: Container Opslag Gebouw (Container Storage Building).
VOG: Verarmd uranium Opslag Gebouw (Storage Building for Depleted Uranium).

[31] TENORM: Technologically Enhanced Naturally Occurring Radioactive Material.

7. LOGISTIC AND SUPPORTING INFRASTRUCTURE (*local on national basis*)

The COVRA site is located in an industrial site in the South-West of the Netherlands in the province of Zeeland and in the neighbourhood of the town Flusshing (Figure 11). The site has good transport connections with public roads and has a railway connection. A harbour facility is located at a few kilometres from the site.

FIG. 11. Nuclear and coal fired plants, showing the COVRA site in the background.

8. PUBLIC ACCEPTANCE

As mentioned before, public acceptance is quite high for long term storage. The general public has more confidence in physical control by today's society than in long term risk calculations for repositories even when the outcome of the latter is a negligible risk.

The level of public acceptance of underground waste disposal remains low. Progress of the disposal programme was stalled by lack of approval for site investigations in salt formations, considered suitable for this purpose and, hence, the prospect of a waste disposal facility being available within the next few decades is remote.

9. CONFORMANCE WITH NUCLEAR SAFETY REQUIREMENTS

Safety was a matter of great concern in the design of the HABOG. The criteria for a safe design can be divided into two groups, relating to the source of possible disturbance of the safe situation:

(a) External influences;
(b) Internal influences.

External influences taken into account were earthquake, flooding, gas cloud, whirlwind and aircraft crash. The design criterion for these events, with a frequency of occurrence of 10^{-6} (i.e. once in a million years) is that they should not create any radiological effect in the surroundings of the facility.

The internal influences considered were: high radiation levels and heat production.

The design should be such that the radiation impact on the environment (and the employees) is ALARA.

10. QUALITY ASSURANCE

For all types of radioactive waste specifications are available. The specifications have to be approved by the Dutch authorities (KFD). COVRA has the right to audit all types of waste coming from clients to convince herself that the waste fulfils the criteria of the specification. In case of the waste forms coming from reprocessing of spent nuclear fuel (commercial reactors) the specifications are approved worldwide by the base load customers of the reprocessing utilities (AREVA and Sellafield). Also the different national authorities and the national waste organizations have to approve the specifications.

Besides the quality assurance programme of the reprocessing facilities the base load customers hired an independent company for quality control of the waste to be returned to the country of origin. For the reprocessing of AREVA this is Bureau Veritas and for Sellafield it is Lloyds.

Besides all these external controls COVRA does its own quality control before the reprocessed waste is transported. This control starts with the control of the delivery documentation, the production documentation, witness of the waste before loading and a control of the transport and final documentation before the transport takes place.

For the acceptance of the RRSNF the data of the fuel is sent to COVRA. COVRA checks if these data fulfils the requirement of the specification and the certificate of the transport container. During the loading of the fuel representatives of the IAEA, Euratom and COVRA are present to check the fuel, the number of the assembly and the position in the basket.

11. SAFEGUARDS

As mentioned in 2.2.1 the sector NBS of the KFD is responsible for the safeguards. On behalf of them these inspections are carried out by Euratom which is accompanied by representatives of the IAEA. Due to the fact that fissile materials are stored with a very high content of ^{235}U, the storage facility HABOG is equipped with cameras and gamma and neutron measurement devices from Euratom/IAEA, and mass storage devices to store all the data.

The storage wells in which the fissile material is stored have, besides the camera survey, additional seals in the form of Cobra seal metallic seals and Vacoss seals.

Until now the administrative controls and controls of the mass storage devices are done on a three month basis and at arrival of RRSNF. In the next future Euratom/IAEA have the intension to implement an online connection by SDSL line to check the health of the equipment and to select one or more camera’s to have a direct view. They should respect a 24 hour delay in the sending of the camera pictures.

12. NATIONAL PLANS/OPTIONS RELATED TO STORAGE OF RRSNF FOR THE NEXT 100 YEARS

As mentioned in the introduction the national plan is to store all kinds of radioactive waste for the next 100 years in above ground structures.

12.1. Comparison between wet and dry storage of RRSNF

12.1.1. Wet storage

Wet storage has the following advantages:

- Storage pools are already available at the reactor site;
- Proven technology. All spent fuel is (temporarily) stored in pools for cooling;
- Cheap storage.

On the other hand, wet storage has the following disadvantages:

- Corrosive environment;
- Possibility of damaging the fuel due to corrosion;
- Cleaning of the water by filtering, radioactive residues;
- Necessity of control of water quality;
- Safeguards is not easy;
- Mechanical equipment such as coolers and pumps to maintain over a long period;
- Possible occurrence of leakage of the pool;
- Possibility of spreading contamination to the environment.
- Before transportation of the fuel it has to be dried and packed into a cask.

12.1.2. Dry storage:

Dry storage has the following advantages:

- Storage in inert atmosphere, no corrosion problems of the fuel;
- Natural cooling possible, no running equipment, no maintenance;
- Contamination of the environment unlikely;
- The fuel package can be transported;
- Safeguards is easy.

On the other hand, dry storage has the following disadvantages:

- Expensive buildings or containers for storage. This depends of the amount of fuel or the combination with storage of reprocessing waste like in HABOG;
- Repair equipment and hot cell in case of damaged canisters or leaking seals of the storage casks;
- Surveillance with pressure devices is necessarily to detect leakage of a seal;
- A valid certificate of the cask for transport after a long time of storage is difficult.

13. CONCLUSIONS

The size of the nuclear programme in a country strongly influences the choice of a radioactive waste management system. In the Netherlands a tailor made system has been developed that now is in operation for 25 years.

In the Netherlands the policy, for spent nuclear reactor fuel from commercial reactors up to now, is reprocessing. For the storage of this reprocessing waste a choice was made to construct a storage facility, HABOG, for the storage of heat generating reprocessing waste and non-heat generating waste. The heat generating waste from reprocessing is stored in natural cooled vaults, so it was a logical choice to store the heat generating RRSNF, welded tight in canisters, in the same vaults instead of in a separate building in storage containers. As COVRA is the only national storage facility for radioactive waste, wet storage will never be an option.

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FORTY-NINE YEARS OF SAFE STORAGE OF RESEARCH REACTOR SPENT FUEL AT ANSTO

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Abstract

ANSTO permanently shut down its 10 MW research reactor (HIFAR) in January 2007 following 49 years of operation. The shutdown followed the earlier closure of a smaller 100 kW research reactor (MOATA) in 1995. The spent fuel resulting from the operation of the HIFAR and MOATA reactors (2281 elements) was stored in wet and dry storage facilities. Of the 2281 spent fuel elements produced only 19 incurred some degree of "damage", either physical or chemical. Until recently (2007) some of these elements were still kept in dry storage, awaiting removal and preparation for final shipment. The damaged fuel elements were then removed from the wet storage ponds, some placed inside special sealed cans, and then deposited in the dry storage holes for long term storage. The management of spent fuel remained a very important aspect of the operation of research reactors for ANSTO. For disposition of UK-origin spent fuel arising from the operation of the HIFAR reactor, ANSTO initially elected to ship the irradiated fuel assemblies to the UKAEA in Dounreay, Scotland. With the closure of Dounreay, alternatives were evaluated, and reprocessing of the spent fuel at the La Hague reprocessing plant was selected as the option for the disposition of ANSTO's UK origin spent fuel. Between 1999 and 2004, a total of 1288 fuel assemblies were sent in four shipments to the La Hague reprocessing plant. For the remaining HIFAR fuel assemblies containing US origin uranium, ANSTO decided to exercise its option to return the fuel assemblies to the USA under the Foreign Research Reactor Spent Nuclear Fuel (FRR SNF) acceptance programme. ANSTO's remaining spent fuel was shipped to the USA in 2006 and 2009 respectively. This paper describes ANSTO's management of its spent fuel inventory.

1. INTRODUCTION

ANSTO has safely operated research reactors for over 50 years (Table I). The longest running research reactor, HIFAR, was a heavy water moderated, light water cooled high neutron flux reactor which operated from 1958–2007. The reactor was powered by 25 MTR type fuel elements and operated at 10 megawatts. Following removal from the reactor core, the spent fuel elements were initially wet stored for a minimum period of 21 months and then transferred to dry storage tubes, mostly under a nitrogen atmosphere. The storage and monitoring methods employed were generally highly successful in ensuring the integrity of the spent fuel elements (SFEs). Of the 2281 fuel elements used, the majority (2262) were maintained in sound condition, with all cladding having total integrity and with no fission products being released.

TABLE I. ANSTO RESEARCH REACTORS

Reactor (power)	Type	Started	Shutdown
HIFAR (10 MW)	DIDO Reactor	1958	2007
MOATA (100 kW)	ARGONAUT	1965	1995
OPAL (20 MW)	Open Pool Light Water	2006	

ANSTO shipped its spent fuel to Dounreay (Scotland) in 1963 and 1996, Savannah River (USA) in 1998, 2006 and 2009 and Cogema (France) in 1999, 2001, 2003 and 2004. In all of the overseas shipments there were only 19 'compromised' elements with some evidence of corrosion. The 19 compromised elements were returned to the USA in the 2009 and final HIFAR spent fuel shipment, after the fuel elements were placed in specific designed over pack canisters, approved by US DOE for reception and storage at Savannah River. The low number of compromised elements demonstrated that the wet and dry storage system used to store the spent fuel during nearly 50 years of reactor operation at ANSTO provided the necessary long term safe storage for the fuel.

2. FUEL TYPES USED

During HIFAR's operating life, a number of different MTR fuel assembly designs were employed, each of which comprised of an enriched uranium-aluminium cast alloy fuel meat (95% volume Al) dispersed in aluminium metal within high purity aluminium cladding [1]. Each assembly consisted of curved rectangular fuel plates, with the fuel meat surrounded by a picture frame and metallurgically bonded between two sheets of Grade 1050 (99.5% Al) aluminium by hot rolling. Each fuel plate was tested at 600°C for 20 minutes [2] and examined for blisters after cooling, to check the integrity of the bond between the meat and cladding. The cladding method prevented the release of uranium and fission products during burnup, and ensured that in the event of a minor breach to the cladding, the only part of the fuel meat exposed to water was directly under the breach. Highly enriched uranium elements (up to 93% ^{235}U) were used until 2005, when HIFAR was converted to use low enriched uranium elements (20% ^{235}U) for the final years of operation.

Figure 1 depicts the variations of fuel types used by HIFAR. There have been three main geometric variations:

- The parallel plate box type (Mark II);
- The annular involute type where the plates form spiral fins between two aluminium cylinders (Mark III);
- Concentric tubular elements consisting of four tubes, each made from three curved plates electron beam welded to form the tube (Mark IV). The figure shows the tubes after cropping the end pieces and held clipped together in preparation for storage or shipment.

For each of the geometric variants, there were several different levels of enrichment, different ^{235}U loading, different country of origin obligations, and after being loaded in the reactor core, they also reached different levels of burnup. All these factors had implications for the subsequent handling, storage, transportation and ultimate disposition of the spent fuel. Table II shows some characteristics of the fuel used in ANSTO's research reactors

3. SAFETY CONSIDERATIONS

Fundamental to the safety of spent fuel storage systems is the maintenance of an accurate detailed record of the location, movements, irradiation history and composition of every individual spent fuel element. Safety assessments and operating procedures must ensure that every conceivable combination of possible fuel variants, enrichments, uranium loadings and burnup is taken into account. This depends on very precise knowledge of the spent fuel being handled in any given operation.

The safety objective with the design of spent fuel storage facilities is to ensure that spent fuel will be received, stored and retrieved without undue risk to health and safety, or to the environment. To achieve this objective, the design must incorporate features that will be effective for the lifetime of the facility under normal operating conditions, anticipated operational occurrences and accident conditions. Relevant features include:

- Maintaining sub-criticality;
- Removing spent fuel decay heat;
- Providing for radiation protection;
- Maintaining isolation of radioactive material from the environment.

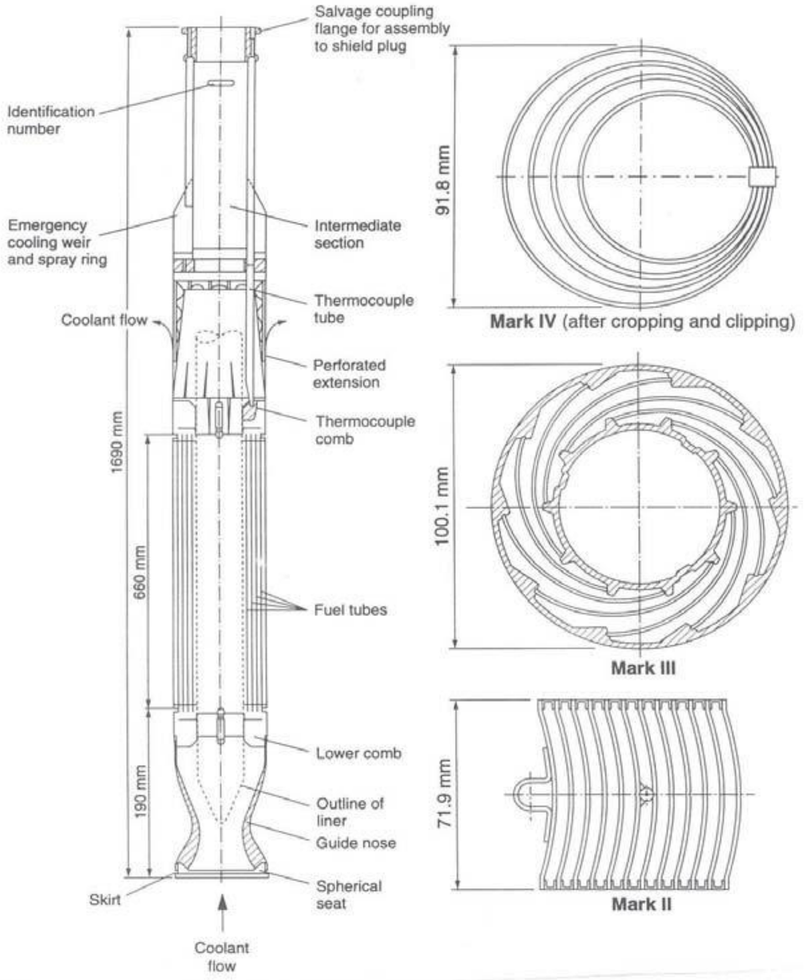

FIG. 1. HIFAR fuel type.

TABLE II. CHARACTERISTICS OF FUEL ELEMENTS USED IN ANSTO'S RESEARCH REACTORS

Reactor	Fuel characteristics
MOATA	60% – 90% HEU fuel assemblies with 12 aluminium clad fuel plates containing about 22–23 g of ^{235}U in an aluminium/uranium alloy
HIFAR	60% – 90% HEU fuel assemblies consisted of a uranium-aluminium alloy in an aluminium matrix with ^{235}U ranging from 115 g to 170 g until 2006. converted to <20% LEU fuel in 2006 until its closure in January 2007
OPAL	<20% LEU fuel consisting of uranium-silicide dispersed in aluminium and clad in aluminium

Whilst fuel cladding has been shown to be capable of retaining its integrity over periods of several decades in storage, very careful attention must be paid to the storage conditions when long term storage is contemplated if unacceptable degradation of the fuel cladding is to be avoided.

4. WET STORAGE

Following the removal of burnt spent fuel elements from HIFAR, they were placed in a sealed storage block, containing de-ionised water, and located within the HIFAR containment building. The initial storage block period is 3–6 months and allows sufficient cooling to occur for the spent fuel to be safely transferred to a larger capacity open pool wet storage facility outside of the HIFAR containment building. The full wet-dry storage cycle is shown in Fig. 2.

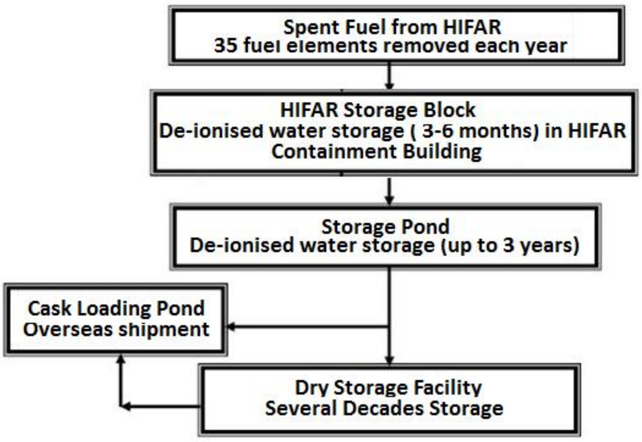

FIG. 2. Wet/Dry Spent Fuel Storage cycle at ANSTO.

Figures 3 and 4 show the ANSTO spent fuel cropping and storage ponds (medium term storage – up to three years), respectively. In the medium term wet storage facility, the spent fuel elements are first cropped, to remove the non-fuel aluminium waste material ends, and then the cropped fuel element is passed through a transfer port (connecting the cropping pond to the storage pond) for medium term wet storage. The storage pond has storage racks that can hold up to 394 spent fuel elements at any time. Both ponds are stainless steel lined, and have a common ion exchange treatment (purification) system, for water circulation and removal of any impurity arising from the spent fuel elements, via corrosion or environmental dust.

Water is pumped from the cropping pond, passed through filters, a fully sealed UV treatment system (for disinfection) and an ion exchange unit, before returning to the cropping pond. An overflow pipe connects the storage pond to the cropping pond. The pond water circulation system includes cartridge filters for removal of particulates and ion-exchange resin beds for removal of dissolved ions.

FIG. 3. Spent Fuel Cropping Pond.

FIG. 4. Spent Fuel Storage Pond.

Cropped spent fuel elements are transferred to the adjacent storage pond via an underwater connecting gate transfer port as highlighted in yellow arrows in Figs 3 and 4.

5. DRY STORAGE

Just as the water quality must be controlled and monitored to preserve spent fuel in wet storage, the condition of the storage atmosphere in dry storage must also be carefully controlled and monitored. The dry facility should be designed and operated so as to minimize the possibility for water (moisture) to enter the facility. From our experience at ANSTO the issue of radiolytic ionisation and chemical reactions in the moist air have been shown to produce oxides of nitrogen in the atmosphere. When dissolved in any free water these lead to weak solutions of nitric and nitrous acids which attack aluminium cladding aggressively. In the absence of water no such attack has been seen even though some oxides of nitrogen must be present in the dry atmosphere. Therefore, provision should be made to detect and remove any water that inadvertently enters the facility and to maintain the humidity of the atmosphere at very low levels. The atmosphere is usually dry air. The recommended practice is to replace the air with dry nitrogen or inert gas (Argon) which further limits the potential for radiolysis reactions.

ANSTO's Dry Storage Facility (Figs 5 and 6) is a typical vault type dry storage and was one of the earliest such storages built. It consists of fifty stainless steel tubes, each one about 16 m long and placed in holes drilled into sandstone bedrock. Two spent fuel elements are placed in transfer canisters (stainless steel) and the canisters lowered into the holes. Each storage hole takes eleven canisters for a total capacity of the store being 1100 spent fuel elements or approximately 30 years of HIFAR operations.

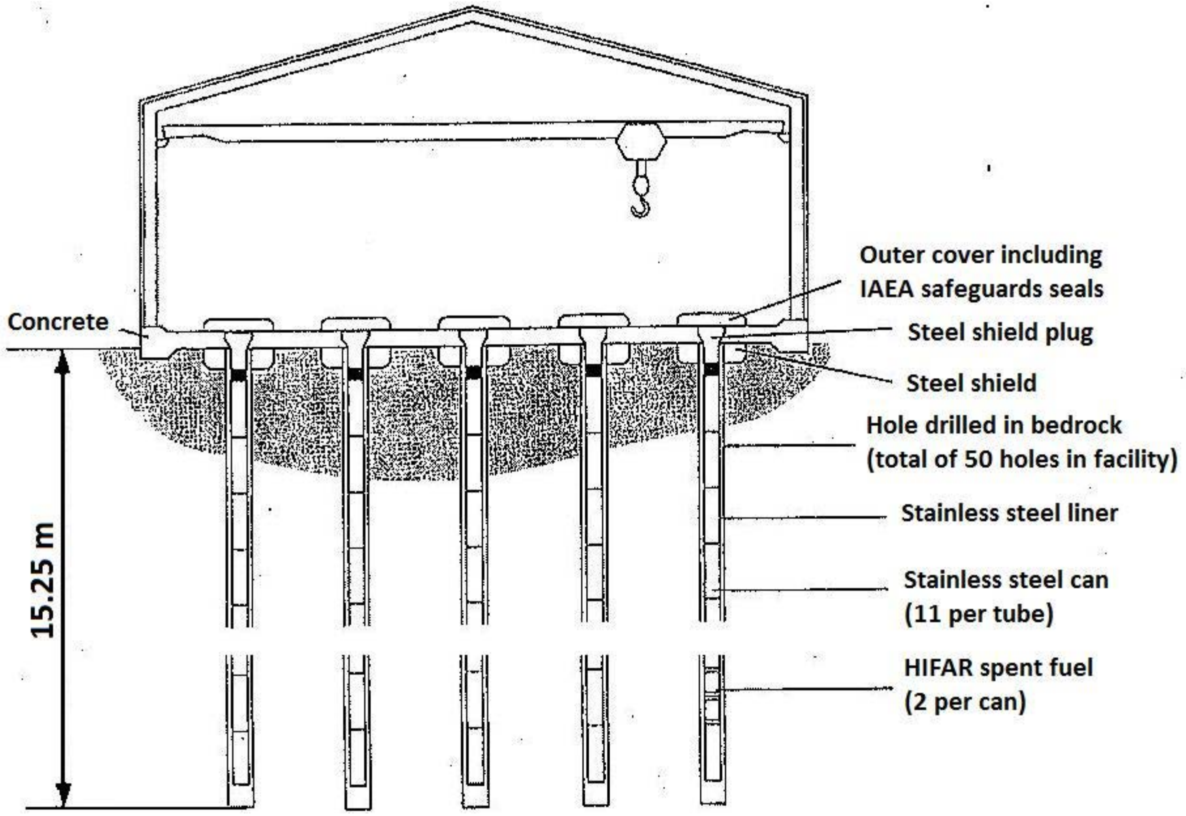

FIG. 5. Schematic of ANSTO's Dry Storage Facility.

FIG. 6. External view of ANSTO's Dry Storage Facility.

The transfer of spent fuel elements between the wet and dry storage facilities is carried out using a shielded spent fuel general purpose transfer flask, shown on the right side of Figure 7. On the left side of Fig. 7 is shown the medium term wet storage pond, used for loading spent fuel assemblies into overseas shipment casks, and for extraction of spent fuel elements using the general purpose transfer flask.

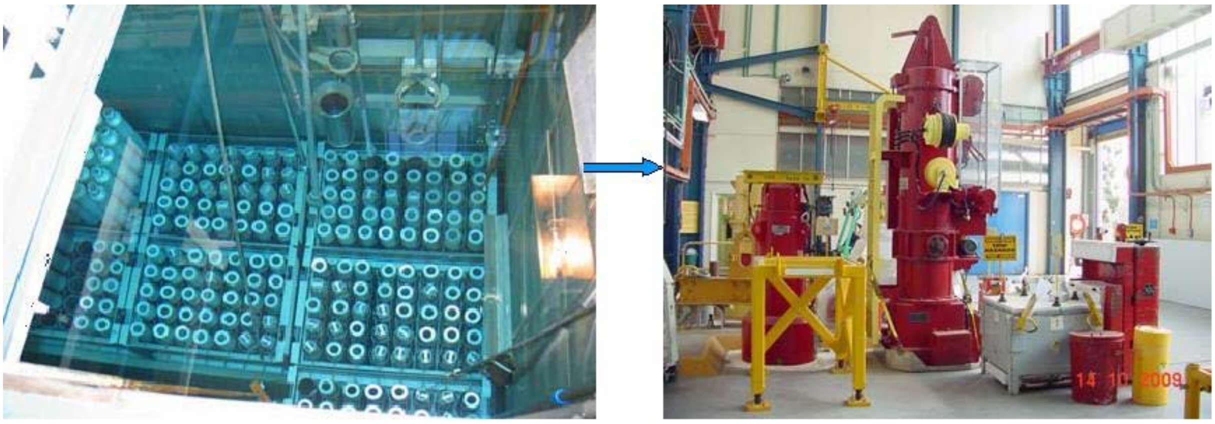

FIG. 7. Medium term wet storage pond (left), and a general purpose transfer flask (right).

6. MONITORING CONTROLS IN WET AND DRY STORAGE FACILITIES

6.1. Wet storage

For aluminium-clad research reactor fuel the predominant type of corrosion occurring is pitting corrosion which can lead to direct exposure of the fuel meat to water. In wet storage, pitting corrosion is promoted by poor water chemistry which includes too high or too low pH, high conductivity or high chlorine ion concentration which can all lead to the onset of corrosion. Other factors promoting corrosion are stagnant water which allows buildup of ions or conductivity in crevices, surface impurities on the cladding and contact with dissimilar metals (galvanic corrosion). The presence of biological films on the cladding has also been reported to increase corrosion.

To prevent the occurrence of aluminium cladding corrosion in wet storage it is recommended that the pool water be frequently (or continuously) circulated through an ion exchange column and filter. At ANSTO the chemical control is implemented through pumping the pond water through a bed of ion-exchange resin that removes virtually all the dissolved ions. Any particulate material is removed in the cartridge filters, which are located before the ion exchange resin beds. In addition to the removal of solids in suspension and ions, UV light is used to control biological activity in the water. The parameters for control of the water condition are listed below in Table III. The parameters are monitored through analysis of water samples taken at least monthly. During flask movement operations involving spent fuel, samples are measured more frequently.

TABLE III: THE DESIGNATED RANGES AND MINIMUM TESTING FREQUENCIES FOR CHEMICAL PARAMETERS IN WET STORAGE PONDS

Parameter	Normal Range	Notification Level	Minimum Frequency
pH	5 – 8	<5 – >8	Weekly
Conductivity µS/cm	1 – 10	> 20	Weekly
Gross α activity, Bq/mL	<0.02	>0.02	Weekly
Gross β activity, Bq/mL	<0.50	>1.0	Weekly

Where activities outside the normal range are measured, follow up γ spec measurements are initiated. If fission products are suspected a determination of ^{137}Cs leaking rate should be the most sensitive indicator [3]. Where the conductivity or activity (α, β) is above the normal operating range, the ion exchange column may need replacing. In this instance, samples from the Ion Exchange Column return are analysed to determine the effectiveness of the resin bed.

It is common that no firm limit is imposed on the radioactive content of the pond water but rather an "action level" is established which, if reached, requires an investigation and corrective action to be taken. This action level is generally well below any level that would represent a direct radiation hazard to operating personnel. The action level is established at a value that ensures a good measure of integrity of the fuel and the proper operation of the pond water cleanup system. For example, at ANSTO for the fuel storage and loading cask ponds the normal operating levels of radioactivity are in the range of 50 Bq/L to several hundreds of Bq/L depending on the frequency of spent fuel movements in the ponds. The action level is established at 1000 Bq/L. These are very low levels compared with some other spent fuel ponds overseas. For example, in the Receiving Basin for Off-site Fuels (RBOF) of Savannah River Site, the normal steady-state basin activity level is about 4,000 Bq/L [3], and the normal operating levels in the power reactor fuel storage ponds at La Hague, France and the CLAB Facility, Sweden are 12 000 Bq/L and 10 000 Bq/L respectively [4] Ongoing monitoring of the wet storage ponds is routinely carried out at ANSTO. Figures 8 and 9 provide the results over the past 14 years. The spikes in alpha and beta activity during 2002 resulted in the breach of the fuel meat of one spent fuel element during routine cropping operations, when the spent fuel element has its metal waste ends cut off with in-pool circular saws. The investigation of the event led to the root cause being the non-placement of an end stopper on the cropping saw mechanism. This led to non-alignment of the cropping saw blades resulting in the saws partially cutting though the fuel meat and releasing fission product (swarf) into the pond water.

Figure 10 shows the setup of the cropping pond saw cutting equipment with the missing spacer.

6.2 Dry storage

The aim of monitoring the spent fuel in dry storage is to detect for moisture and humidity probes. A well sealed, dry storage compartment should maintain relative humidity below 20% and humidity above 50% is an indication of a poor seal. Where dry nitrogen or inert gas is used as the normal storage atmosphere (as at ANSTO) the two possibilities can be distinguished by means of a concurrent oxygen probe measurement. High humidity and oxygen suggest air leakage into the storage. High humidity and oxygen (<5%) suggest the seals are working but water has been inadvertently introduced into the storage.

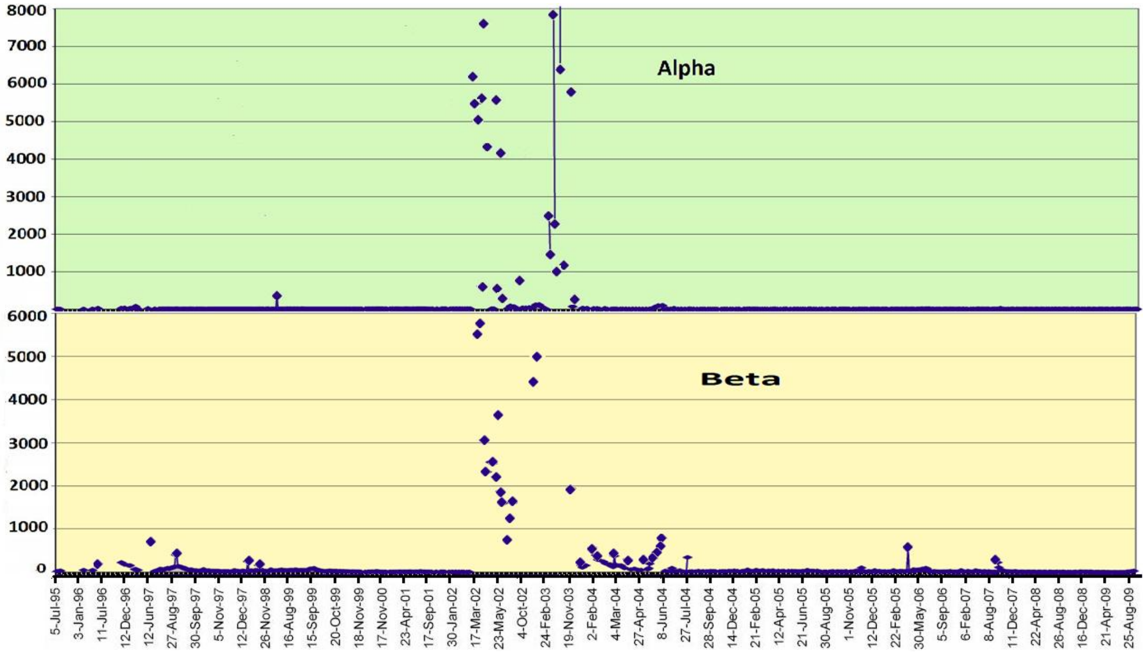

FIG. 8. Alpha and Beta levels (KBq/m3) for B23 cropping pond.

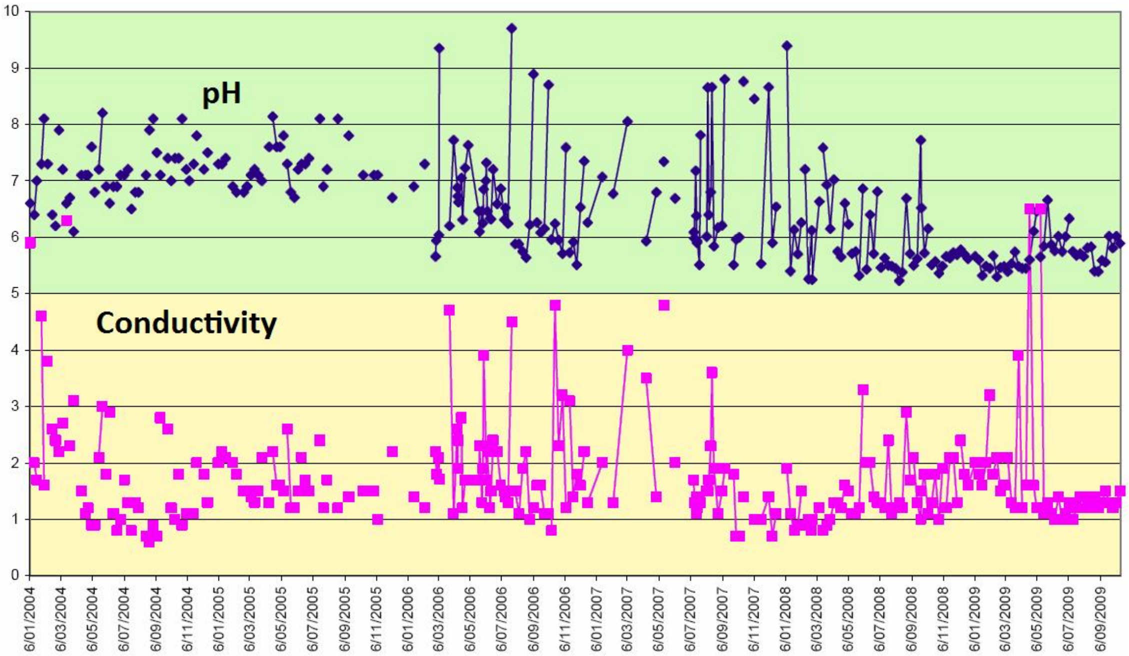

FIG. 9. pH and conductivity levels for B23 cropping pond.

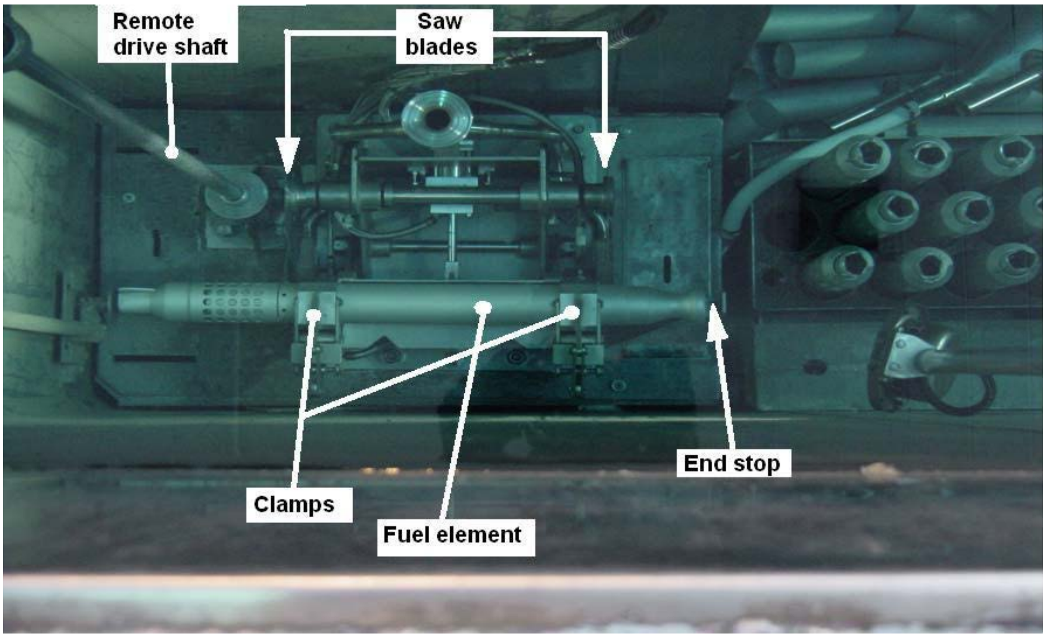

FIG. 10. Cropping pond saw with end stop spacer missing.

Radioactivity checks can be made by measuring the beta-emitting gaseous fission product krypton-85. Experience at ANSTO shows that the detection limit for this test is around 0.15 MBq/m^3. However krypton-85 is not considered a sensitive early warning sign of the onset of a corrosion problem in research reactor fuel dry storage. It can be considered more useful as a backup measure in the sense that a negative result gives confidence that no major problem has already developed in storage. The more reliable measure is provided by the combination of humidity measurements and a regular programme of using randomly selected fuel elements for visual inspection. Figure 11 shows the dry nitrogen purging and a filling unit used on the dry storage facility at ANSTO.

FIG. 11. Dry nitrogen purging of dry storage holes.

6.3 Hot cell inspection of spent fuel

As indicated in Section 6.2 a more reliable measure of spent fuel integrity testing is provided by a regular programme of using randomly selected fuel elements for visual inspection. At ANSTO random inspection of spent fuel from wet and dry is carried out in dedicated hot cells.

The hot cells have been extensively used to carry out physical non-destructive testing of ANSTO's spent fuel inventory over the past 50 years. As mentioned earlier, of the 2281 spent fuel elements produced during the operation of the HIFAR reactor only 19 elements were considered as being affected by some degree of corrosion during the storage period. These "compromised "spent fuel elements were assessed in the hot cells, as illustrated in Fig. 12.

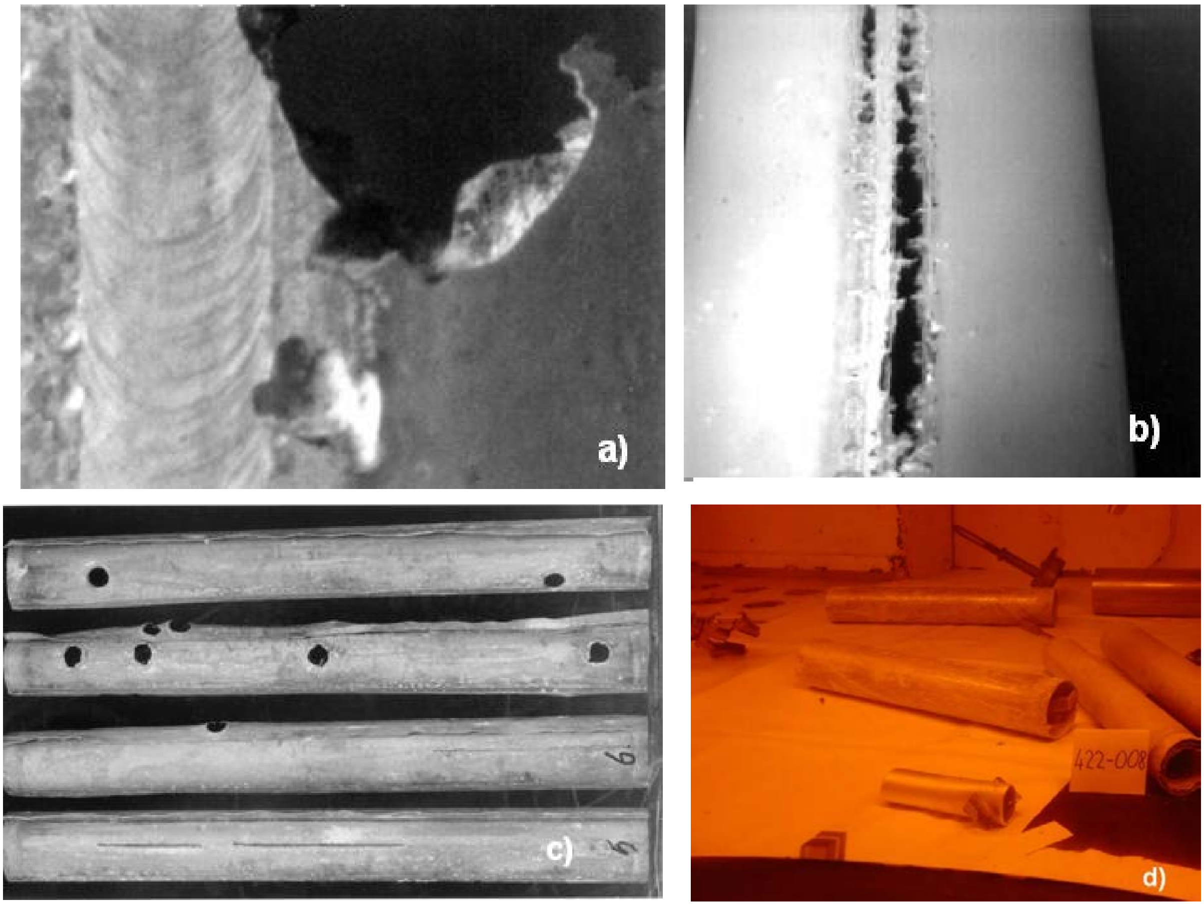

FIG. 12. (a) The corner of the fuel meat has been exposed at the top of a plate from assembly; (b) Extensive pitting corrosion in the non-fuel section, resulting in compromised structural integrity; (c) Punched plates (not corrosion but holes physically made to sample the fuel meat; (d) Spent fuel undergoing inspection in hot cell.

7. SPENT FUEL SHIPMENTS

In April 2009 ANSTO shipped its 9th and final spent fuel shipment to the USA (Table IV). The final shipment resulted in all of the spent fuel produced from the operation of the HIFAR and MOATA research reactors to being shipped overseas for reprocessing or as part of the Foreign Research Reactor (FRR) Spent Nuclear Fuel (SNF) Acceptance Programme. As mentioned earlier the successful shipment of the spent fuel (2281 spent fuel elements) can be attributed to the effective storage and monitoring programme at ANSTO over the past 50 years.

TABLE IV. ANSTO SPENT FUEL SHIPMENTS

Year	Quantity	Destination
1963	150	Dounreay (UK)
1996	114	Dounreay (UK)
1998	240	Savannah River (USA)
1999	308	Cogema (France)
2001	360	Cogema (France)
2003	344	Cogema (France)
2004	276	Cogema (France)
2006	330	Savannah River (USA)
2009	159	Savannah River (USA)
Total of Spent Fuel Elements shipped	2281	

For the loading operation, a cask loading/buffer storage pond, shown in Fig. 13, and previously in the left side of Fig. 7, was used.

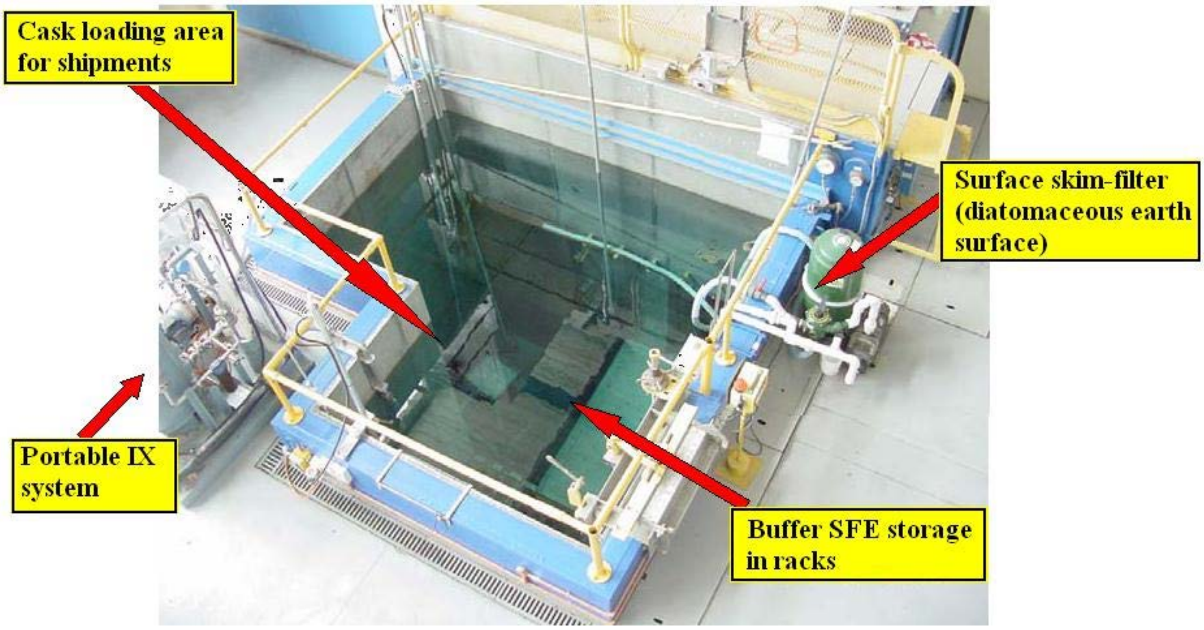

FIG. 13 Cask Loading/ Storage pond.

Figures 14 to 16 provide practical evidence of the preparation for the spent fuel shipments using both wet and dry cask loading systems.

FIG. 14. (a) Preparation for spent fuel shipment: wet loading; (b) drying; (c) inspection; (d) cask loaded and ready for shipment.

FIG. 15. (a) Preparation of spent fuel shipment using dry loading cask (basket in pond; (b) shielded sleeve; (c), basket removal system; (d) dry loading cask being set up for loading.

FIG. 16. Spent fuel shipment ready for transport to dock.

8. ILW RETURN FROM REPROCESSING

The Australian Government, through ANSTO, has contractual agreements in place for the reprocessing of research reactor spent fuel in the United Kingdom and France and the subsequent return of reprocessing residues in flasks as shown in Fig. 17. At the time of entering into these contracts, it was assumed that the Australian Government would have a licensed intermediate level waste (ILW) storage facility in operation before the scheduled return of the ILW reprocessed wastes.

The current lack of a licensed ILW storage facility in Australia and the absence of firm plans to site and license a facility prior to the scheduled return of the ILW residues are of major concern to ANSTO.

Under the contractual agreements, ANSTO must provide details for the repatriation of residues by the end of 2010. These details necessarily include the return destination in Australia — a facility that is appropriately designed and licensed for operation.

The complexity of the return logistics and the lengthy national and international regulatory approval process requires that detailed planning and cost estimation be carried out in the near future. In order to progress that planning, it is necessary for ANSTO to have a suitable backup option for the receipt of ILW residues. The only backup option that is compatible with the tight time schedule is to construct and licence an interim ILW storage facility at ANSTO. This interim storage facility would operate only until such time that the national ILW storage facility is constructed and licensed for reception and storage of these wastes.

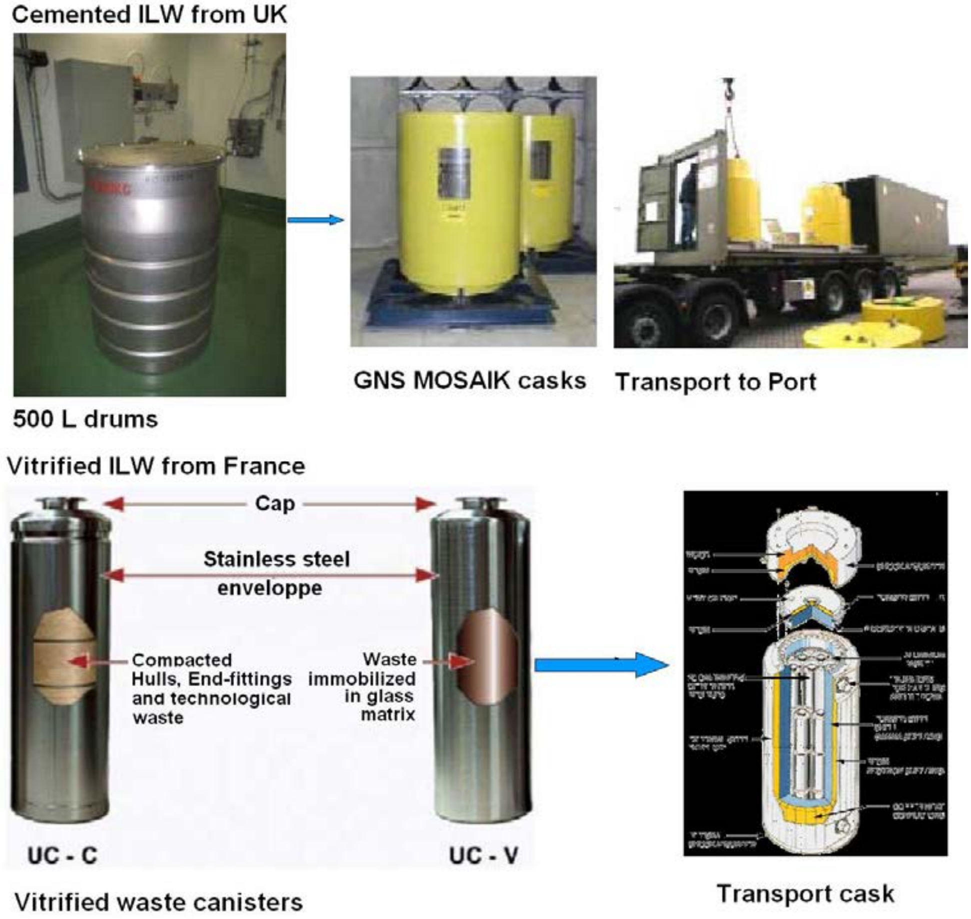

FIG. 17. Reprocessed ILW packages and transport casks for cemented ILW from Dounreay (top) and vitrified ILW canisters from France.

9. OPAL SPENT FUEL MANAGEMENT

OPAL, the new Australian research reactor, is a 20 MW open pool, heavy water reflected and light water cooled reactor, which started operation in August 2007. OPAL generates a nominal 36 spent fuel elements per year and operates typically in cycles of 30–35 days, followed by a short refuelling outage to remove two or three spent fuel elements, that are replaced with new fuel elements. Figure 18 shows the OPAL Reactor Service Pool (front) and the reflector vessel (core) in the background.

The service pool has a nominal storage capacity for 10 years of OPAL operation (up to 360 spent fuel elements), and under the current OPAL operating license, the storage of spent fuel is restricted to the OPAL Service Pool (wet storage). ANSTO’s current plan is to send two shipments of OPAL spent fuel to the USA, as part of the Foreign Research Reactor Spent Nuclear Fuel Acceptance Programme. The programme is scheduled to accept fuel discharged from eligible reactors before May 2016, to be received in the USA by May 2019. After the expiration of the programme, ANSTO will send its used research reactor spent fuel overseas for reprocessing, with ultimate return of reprocessing wastes to Australia.

FIG. 18. OPAL reactor service pool (rectangular pond) showing the storage racks for spent fuel.

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WET STORAGE PRACTICES AND EXPERIENCE

(Part 2)

WET INTERIM STORAGE OF RESEARCH REACTOR SPENT FUEL IN INDONESIA

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Abstract

Multipurpose research reactor G.A. Siwabessy started to be built in 1983, and reached criticality for the first time in July 1987. Official opening ceremonial was attended by president of RI on 20 August 1987. The first operation of the reactor at full nominal power was made in March 1992. RSG-GAS has a nominal power of 30 MW, and an average neutron flux of 10^{14} n/cm^2.sec. It is an open pool type reactor with 19.75% enriched U_3O_8Al/U_3Si_2Al fuel, and burnup ratio up to 56%. The reactor has one service pool that is used for temporary wet storage of spent fuel. The storage capacity of the service pool is for 288 spent nuclear fuel (SNF) elements. Usually the SNF is kept in the service pool for about three months to allow a significant decrease in radiation level and in decay heat production, before it is transferred to an Interim Spent Fuel Storage Facility (ISFSF), or repatriated to the origin country, such as the USA. A SNF corrosion surveillance programme is carried out since 2007 in both places, in the service pool and in the interim storage facility. The purpose of the programme is to monitor any type of corrosion that may occur. The test coupons used in the programme were discs made with samples of stainless steel, aluminium-cladding and aluminium-tank liner. A water chemistry management is applied to maintain parameters of cooling water within specified limits. Most of the recent fuel elements used in the reactor are non US origin, and cannot be shipped back to that country, neither repatriated to the country of origin; therefore the ISFSF is being prepared to be used as a facility for wet interim storage of the spent fuel from reactor RSG-GAS, and if necessary, as the storage for the spent fuel from the other two research reactors.

1. INTRODUCTION

Indonesia has three research reactors (RR) that are mainly utilized for radioisotope production, research and development: Kartini (100 kW) in Yogyakarta (operated since March 1979), TRIGA-2000 reactor in Bandung (originally a TRIGA MARK II 250 kW, upgraded to 1 MW in 1972, and upgraded again to 2 MW in 2000) and RSG-GAS reactor in Serpong (a 30 MW reactor operated since 1987).

RSG-GAS is the biggest research reactor in Indonesia. It is an open pool type reactor, light water cooled and moderated. The fuel elements used in the reactor core are Low Enriched Uranium – Material Test Reactor type (LEU-MTR), in the form of U3O8-Al dispersion, which are gradually being replaced with U3Si2-Al since 1998. The reactor core, with active length equal to 60 cm, is placed in the bottom of a 12.45 m water column. The core is arranged on top of a 10 X 10 grid plate with 40 standard fuel elements, eight control elements, eight Ag-In-Cd control absorbers, one central irradiation position, four additional irradiation positions, in the fuel region, 5 rabbit systems and other irradiation positions within the reflector region. In spite of having a 30 MW nominal power, actually the reactor is operated at 15 MW for four cycles a year, which represents a consumption of six fuel elements [1].

These spent nuclear fuels are classified as high level radioactive waste. This paper describes the routine management procedures to handle these spent nuclear fuels, the current status and the plans that are being made for the future.

2. FUEL TYPE

Table I shows the specification of the fuel used in RSG-GAS reactor, and Table II shows the specification of the fuel used in both TRIGA reactors — Bandung and Yogyakarta. Special consideration must be given for the spent fuel from RSG-GAS which is having a slight modification to achieve higher performance and reactor utilization [2].

TABLE I. FUEL ELEMENT SPECIFICATION FOR RSG-GAS RESEARCH REACTOR [1]

Dimension (mm x mm x mm)	77.1 x 81 x 600
No. of plates per standard Fuel Element (FE)	21
No. of plates per Control Element (CE)	15
Clad material	AlMg2
Clad thickness(mm)	0.38
Meat dimension (mm x mm x mm)	0.54 x 62.75 x 600
Meat material	U2Si3Al
^{235}U Enrichment (w/o)	19.75%
Uranium Density (g/cm3)	2.96
^{235}U weight per FE (g)	250
^{235}U weight per CE (g)	178.6

TABLE II. FUEL ELEMENT SPECIFICATION FOR TRIGA TYPE RESEARCH REACTORS [1]

Name	Specification			
Catalogue number	102	104	106	108
Total length tube (cm)	75.5	75.5	75.5	75.5
Outer diameter (cm)	3.65	3.56	3.75	3.75
Fuel length (cm)	35.56	35.56	38.1	38.1
Fuel composition	U Zr H	U Zr H	U Zr H	U Zr H
Weight of ^{235}U (g)	37	38	55	99
Weight % of ^{235}U (%)	8.5	8.5	12	20
Enrichment (%)	20	20	20	20
Graphite reflector at the end cm	10.2	10.2	10.2	10.2
Tube material	Al	SS304	SS304	SS304

3. SPENT FUEL MANAGEMENT

3.1. Current status

The RSG-GAS reactor has a service pool that is used for temporary wet storage of the spent fuel removed from the core. It is located adjacent to the reactor pool and is separated from the main reactor pool by a gate that is normally closed during the reactor operation. Figure 1 shows the isometric view of the RSG-GAS reactor and service pools [2].

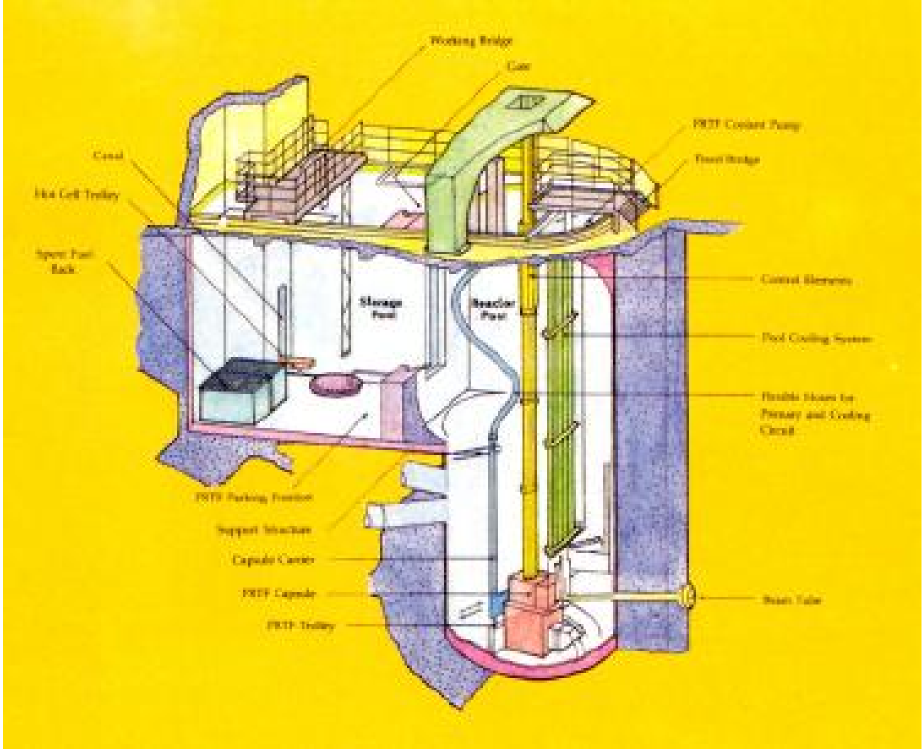

FIG.1 Isometric view of RSG-GAS reactor and service pools [2].

The service pool has a capacity for 288 spent fuels which are stored intentionally for at least three months to allow a significant decrease of the radiation level, and also a decrease in the decay heat produced by the spent fuel before it is transferred to the Interim Spent Fuel Storage Facility (ISFSF). The ISFSF is located apart from the RSG-GAS building but it is connected to it by a transfer channel, as shown in Fig. 2.

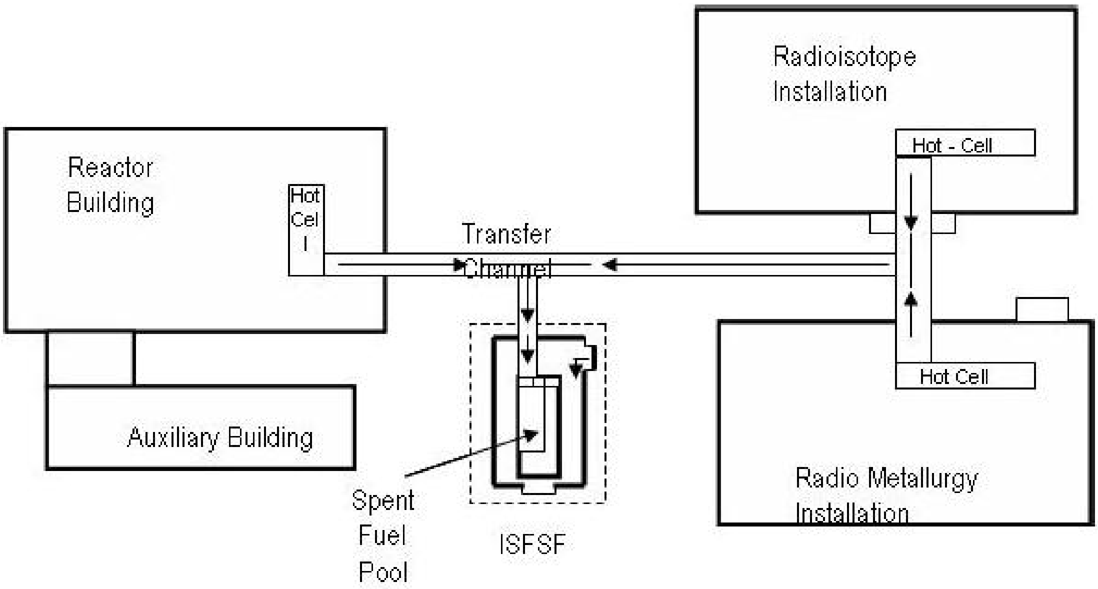

FIG. 2. Diagram showing the ISFSF and the transfer channel that connects ISFSF to the RSG-GAS reactor building [1].

In 1996, when the US Department of Energy issued a Record of Decision on Nuclear Weapons Non-proliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel, with the purpose to recover enriched uranium exported from the USA, a decision was made in Indonesia to repatriate all the US origin spent fuel. BATAN organized the first shipment operation in March 1999. It consisted of 47 spent fuel elements and 1 plate of MTR fuel from RSG-GAS reactor. The second shipment operation was done in 2004, when a total of 111 spent nuclear fuel elements and 1 plate of MTR, from all 3 Indonesian research reactors were returned to the USA. The third and last shipment was done in June 2009, when 42 spent fuel elements from RSG-GAS returned to the USA.

In September 2001 Indonesia started buying low enriched uranium from France and the Russian Federation, so at present all the spent fuel of RSG-GAS reactor is from non US sources, and, according to a recommendation made by Indonesian Regulatory Body, it needs to be transferred soon to ISFSF, together with some fuel from Radio-metallurgy Laboratory. Yogyakarta and Bandung Reactors still have some US origin fuel, that is planned to be shipped back to the USA in 2016.

3.2. Future plan

By Indonesian law, spent fuel is defined as high level radioactive waste, so since 2006 the control of ISFSF was handed over from Serpong Reactor Operator to the Centre for Radioactive Waste Technology (PTLR).

The total water volume of ISFSF pool is about 700 m3. The walls have an aluminium liner, the water is occasionally circulated and purified, and its quality is precisely the same as RSG-GAS primary pool water. The pool has some storage racks that were installed for the previous shipment operations Fig. 3 shows some of these storage racks.

FIG.3. The Interim Spent Fuel Storage Facility (ISFSF) [3,4].

As mentioned before, the present fuels in RSG-GAS are non US origin, and considering that the US return programme will allow to return the spent fuel from the two other reactors only up to 2016, then ISFSF is being envisaged as the only option to be used as national long term storage facility, where the nuclear spent fuel from Indonesian research reactors, including the higher density nuclear fuels from R&D, can be safely stored. Considering this, some effort is being done to prepare the ISFSF to be this national interim wet storage facility. We also expect that the present R&D work being done with higher density nuclear fuels will result in more efficient research reactor nuclear fuel elements, with reduced volume, and making a significant contribution to increase the storage capacity of ISFSF.

The work done so far includes mainly leakage tests by using non-destructive test techniques, and re-welding. Leakage suspicion was motivated by the increase in the amount of water necessary to refill the pool. Calculations showed that the amount (necessary to refill the pool) is much higher than the amount of water lost by evaporation.

Other actions to consolidate ISFSF as an interim storage facility are discussed below.

3.2.1 Surveillance Programme

In 2008 BAPETEN, the Indonesian Regulatory Body issued the operating license for ISFSF facility to cover a 10 year period of operation. Since then, a surveillance programme was established, as part of a water quality management programme, with the purpose to evaluate the present water quality conditions, and to study the resistance of spent fuel cladding to corrosion.

The surveillance programme started in 2007 for the two pools, the RSG-GAS reactor pool and the ISFSF pool. As part of the surveillance programme, the effects of primary water on homogeneity, galvanic and crevice corrosion have been observed using various racks of coupons, made of aluminium and stainless steel, in order to better understand and control the corrosion process in both facilities. The standard procedures established for the corrosion monitoring and surveillance programme, provided valuable technical guidelines for the continued wet interim storage of spent fuels. After having defined the current water chemistry parameters, experiments were conducted using the coupons made of the same grade materials as used for the construction of the fuel cladding (aluminium), the reactor tank (aluminium), and fuel storage racks (stainless steel). The test coupons were made of a series of discs (external diameter 95 mm with a 15 mm diameter centre hole; 1 mm thick for stainless steel and 5.5 mm thick for aluminium) assembled using a stainless steel centre pin and ceramic bushing isolators and 5 mm thick spacers between the discs. The coupons were assembled according to two orientations, to allow vertical and horizontal installation in the pool. Fig. 4 shows the visual result before and after immersing one of the racks in the service pool of RSG-GAS reactor.

FIG. 4. The horizontal coupon racks: before exposure to the water pool (left) and after one year expose to service pool of RSG-GAS reactor (right).

4. CRITICALITY SAFETY ASSESSMENT FOR RSG-GAS SERVICE POOL AND ISFSF

Criticality assessment is one aspect that needs to be considered to safely operate a spent nuclear fuel interim storage facility. The criticality assessment of the RSG-GAS service pool, with existing storage racks, was done to support the reactor core conversion programme, from uranium oxide to uranium silicide [2]. For the preliminary calculations the main parameters considered were fuel densities, water density as a function of temperature, and fuel orientation. First results showed that the existing spent fuel racks can be safely used for storage of the higher density uranium fuels. Additional calculations are being made, to consider the increase in storage rack temperature, caused by radionuclide γ-heating, and the water flow rate, which causes a decrease in the rack temperature. Both effects may influence criticality results.

The criticality safety assessment will also be made to the existing racks in ISFSF. This is necessary to anticipate the possibility of using the facility as an interim storage for the spent fuel from the three Indonesian research reactors, RSG-GAS, TRIGA Bandung and TRIGA Yogyakarta. These calculations are more complex, because as can be seen in Tables I and II, fuel used in TRIGA Bandung and TRIGA Yogyakarta are different from RSG-GAS fuel. In addition, the calculations need to consider the modifications proposed for the RSG-GAS fuel element which had a slight modification to achieve higher performance and improve reactor utilization [2].

5. DESIGN AND MANUFACTURE OF OWN RACKS AND TRANSFER CASK

A new rack's design may be an output of the above described criticality safety assessment, especially for the TRIGA spent fuel. Also, consideration must be given to the possibility of manufacturing it, and considering that the transfer cask used to load the TRIGA spent fuel in the 2004 shipment operation was supplied only for that operation, under US DOE contract, we anticipate the necessity to develop a national project with the purpose do design and construct a new transfer cask, to transport TRIGA Bandung and TRIGA Yogyakarta spent nuclear fuel to ISFSF in Serpong.

6. CONCLUSION

MTR Spent nuclear fuels and a single MTR fuel plate from RSG-GAS have been shipped back to the USA together with TRIGA Bandung and TRIGA Yogyakarta spent fuels. The last shipment of spent fuel from Indonesia is scheduled to be carried out on 2016, however this shipment operation will have no spent fuel from RSG-GAS research reactor, because since 2001 the reactor is fuelled with uranium supplied by France and the Russian Federation. Therefore, ISFSF is being prepared to be used as the national long term spent fuel storage facility. For this purpose, an aluminium corrosion surveillance programme was introduced. Furthermore, a criticality safety assessment programme is being carried out for the facility, and new racks and a new transfer cask design are being planned to the near future.

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US PRACTICE FOR INTERIM WET STORAGE OF RRSNF

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Abstract

Aluminium research reactor spent nuclear fuel is currently being stored or is anticipated to be returned to the United States of America and stored at Department of Energy storage facilities at the Savannah River Site and the Idaho Nuclear Technology and Engineering Center. This paper summarizes the current practices to provide for continued safe interim wet storage in the US aluminium fuel stored in poor quality water is subject to aggressive corrosion attack and therefore water chemistry control systems are essential to maintain water quality. Fuel with minor breaches is safely stored directly in the basin. Fuel pieces and heavily damaged fuel is safely stored in isolation canisters.

1. INTRODUCTION

Spent fuel from research and test reactors worldwide and domestic research reactors is consolidated by the DOE at its US storage sites. The foreign and domestic research reactor fuel is primarily aluminium-based, aluminium-clad, spent nuclear fuel that is being stored in wet storage in L-basin at the Savannah River Site (SRS) and in dry storage at facilities at the Idaho Nuclear Technology and Engineering Center (INTEC) at the Idaho National Laboratory. This paper provides an assessment of the current practices of the USA with respect to wet storage of aluminium-based research reactor fuel.

2. SAFE MANAGEMENT OF RR SPENT NUCLEAR FUEL IN WET STORAGE

Safe management of fuel in continued wet storage involves several key functions, namely:

- Initial condition characterization;
- Water quality;
- Water chemistry control system;
- Corrosion surveillance;
- Handling heavily damaged fuel.

The following sections provide, for each one of the key functions listed above, a discussion of the current practices for the management of spent nuclear fuel in wet storage at the Savannah River Site, highlighting the key technologies involved.

2.1. Initial condition characterization

Aluminium-clad nuclear fuel stored in water basins is subject to corrosion attack if the water quality and storage conditions are aggressive to aluminium. The three primary forms of corrosion observed in aluminium spent fuel, when stored in low quality water, include pitting corrosion i.e. localized corrosion; crevice corrosion, which typically occur in crevices formed at joints, and galvanic corrosion, which occur when dissimilar metals contact each other. Figure 1, shows an example of a fuel assembly of Materials Test Reactor design (multiple fuel plates in a “box” assembly) that has minimal corrosion attack following its use in reactor and basin storage; Figure 2 shows a fuel assembly that has undergone corrosion attack via several corrosion modes. Corrosion can lead to through-clad penetrations of the fuel, and approximately 7% of the total number of aluminium fuel assemblies worldwide is expected to contain through-clad penetrations [1]. Continued interim storage must consider this initial condition of the fuel.

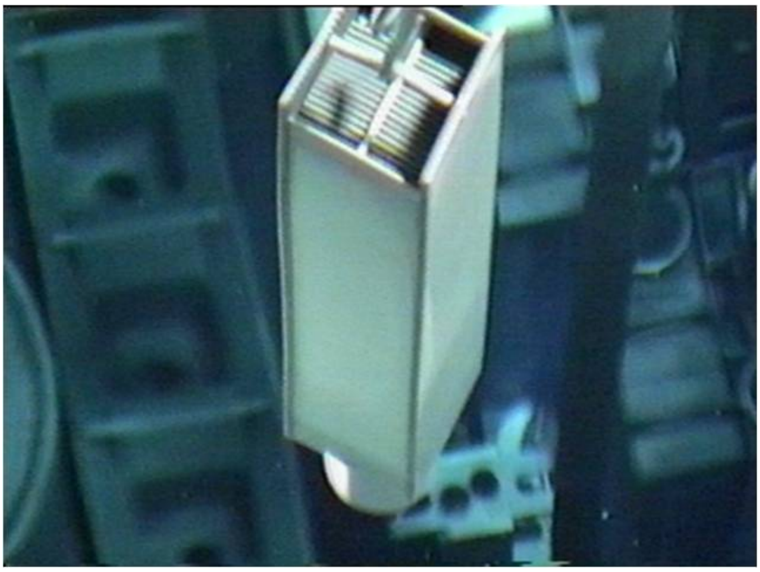

FIG. 1. Aluminium-based, aluminium-clad spent nuclear (plate) fuel assembly in a good physical condition, post-irradiation and stored in a basin with good water quality.

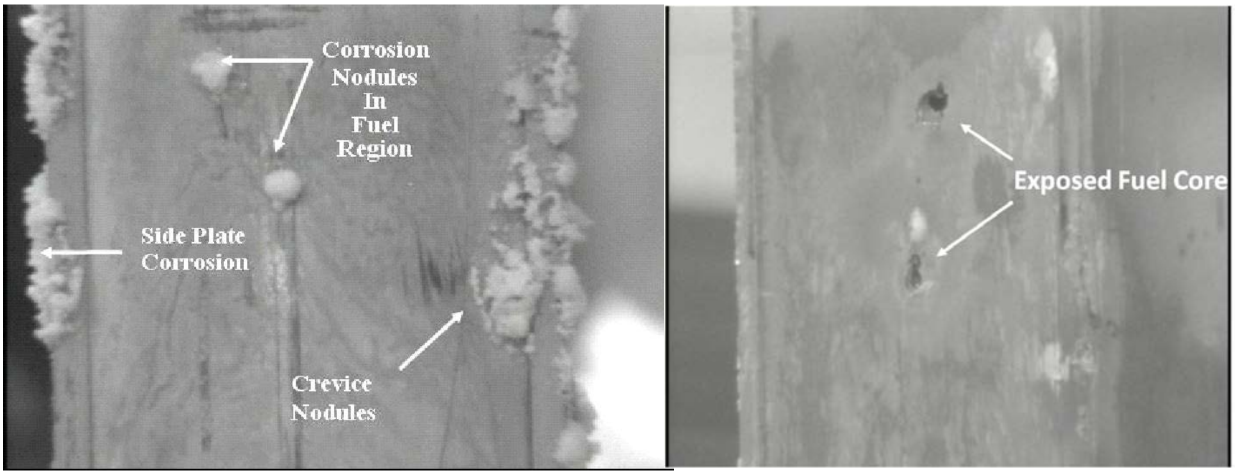

FIG. 2. Aluminium-based, aluminium-clad spent nuclear fuel in moderately degraded condition post-irradiation and stored in a basin with poor water quality and storage conditions.

2.2. Water quality

Key to the successful control of fuel corrosion in extended wet storage is the effective control of the basin water quality. During the late 1960s to mid-1990s the K, C, and P reactor basins of SRS used portable deionizer systems that helped with aggressive ion and activity removal, but were far less effective than the current permanent system installed in L-basin. Through previous work and operating experience at SRS storage basins, it has been shown that the presence of aggressive species (i.e. chloride, nitrate, and sulphate), as well as high water conductivity, causes extensive fuel corrosion issues.

The effects of exposure to a range of aggressive ion levels were investigated in order to map the water chemistry envelope for establishment of optimal water chemistry limits [2]. Cyclic polarization testing was used to perform accelerated corrosion testing on coupons of 6061 and 1100 aluminium. The corrosion potential E_{corr} was measured and used to evaluate the potential impact of the water chemistry on the pitting of the material. Cyclic polarization studies were performed on Al 1100 and Al 6061 coupons under a range of solution conditions. Results for Al 1100 and Al 6061 were obtained at various chemistries as shown in Figures 3 and 4, respectively, indicating that the chloride ion strongly promotes corrosion of the aluminium fuel alloys.

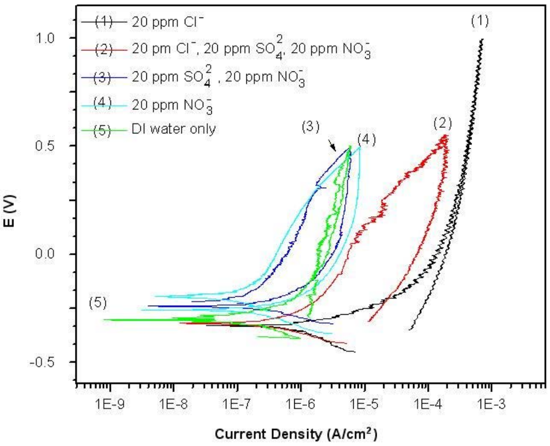

FIG. 3. Cyclic polarization results from Al1100 coupons in aggressive water chemistry. From Ref. [2].

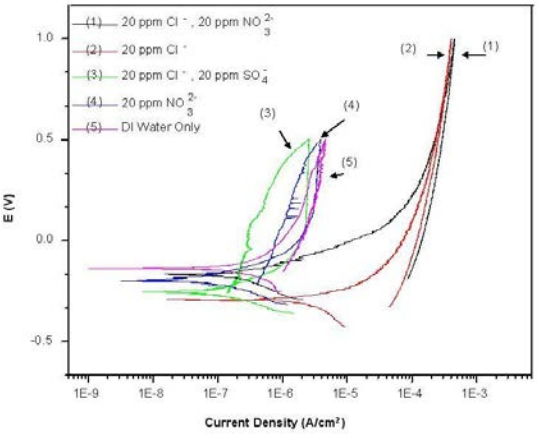

FIG. 4. Cyclic polarization results from Al6061 coupons in aggressive water chemistry. From Ref. [2].

These results also indicate that while the chloride is aggressive, it may be tolerable at higher than current concentration limit of 0.1 mg/L. Additionally, the data suggests that nitrate and sulphate additions have shown a tendency to inhibit pit initiation on Al 1100 and Al 6061 but further study is warranted to fully understand their effects in solution with respect to pitting and general corrosion.

Table I contains the recommended water physical-chemical parameter limits for aluminium fuel storage basins. A publication being prepared by the IAEA provides guidelines for avoiding excessive corrosion through water quality management [3].

2.3. Water chemistry control system

L-basin uses zeolite and ion exchange resins for basin chemistry control for control of corrosion of the fuel. The system consists of two sand filter trains, one zeolite train, and two deionizer resin trains. The sand filter system contains sand and anthracite (coal) to remove insoluble particles. The recirculation flow through the sand filter is 1,800 gpm.

A portion of the flow (200 gpm) from the sand filter enters the zeolite and ion exchange resin systems as indicated in Figure 5. The zeolite train is a single pass, molecular sieve and ion exchange system that effectively removes cesium and strontium ions from the water after is passes through the sand filters, and before it enters the anion and cation resin trains. The zeolite train has two vessels which consist of 100 ft^3 total of zeolite, 50 ft^3 on each vessel.

The anion and cation trains are specifically designed to minimize corrosion by removing and replacing any deleterious ions that have not been removed by the zeolite. The ion exchange resins used in the L-basin system are porous polystyrene/divinylbenzene copolymers. The resin is mostly polystyrene with some divinylbenzene cross-linking for strength.

TABLE I. RECOMMENDED PHYSICAL-CHEMICAL PARAMETERS LIMITS, AND MONITORING FREQUENCIES FOR WATER IN FUEL DECAY AND STORAGE BASINS [3]

PARAMETER	VALUE (LIMIT)	MONITORING FREQUENCY
pH	4.5 to 7	weekly
Conductivity	< 10 µS/cm	weekly
Solids	< 5 mg/l	Every 6 months
Cu Concentration	< 0.1 mg/l	Every 6 months
Cl Concentration	< 0.1 mg/l	Every 6 months
Nitrate (NO_3^-), mg/l	< 10 mg/l	Every 6 months
Sulphate (SO_4^{2-}), mg/l	< 10 mg/l	Every 6 months
Fe Concentration	< 1.0 mg/l	Every 6 months
Al Concentration	< 1.0 mg/l	Every 6 months
Temperature	< 45°C	monthly
Radioactivity level[a]	(see note below)	Weekly
Turbidity[b]	(see note below)	

[a] Water Radioactivity level and the presence of radioisotope species should be measured each time a water sample is drawn or at least one time per week. A gamma scan is recommended to measure the presence of radioisotopes that would have come from failed fuel (e.g. ^{137}Cs). No specific limits are set. The presence of radioisotope species should be evaluated on case-by-case basis. Measurement of the activity from filters and resin columns should be performed to detect the presence of leaking fuel.

[b] Turbidity should be reduced as necessary, to provide visual clarity in the water system.

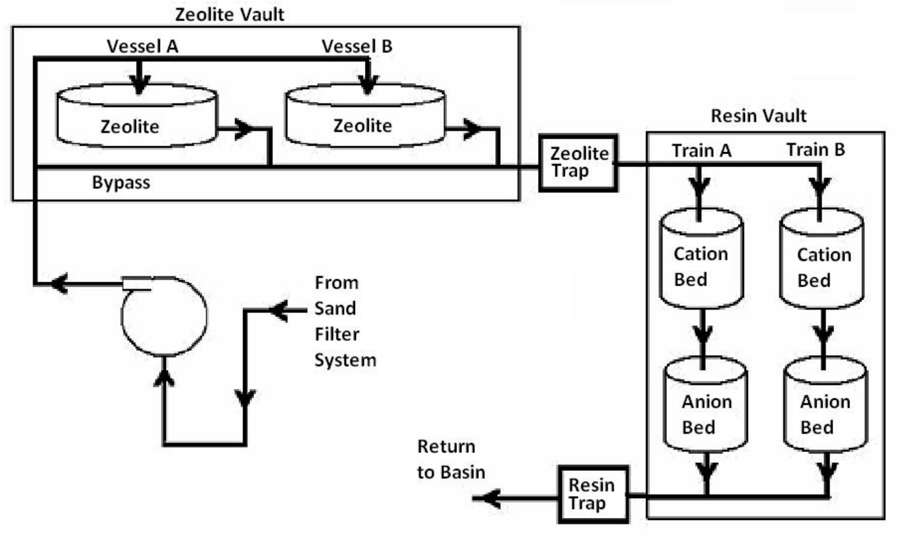

FIG. 5. Water Chemistry Control System Layout for L-basin.

The resins are similar, except for the amount of divinylbenzene and the attached functional groups. The cation resin contains 8% divinylbenzene and the functional group is $SO_3^-H^+$. The anion resin contains 4% divinylbenzene and the functional group is $CH_2\text{-}N^+(CH_3)_3OH^-$.

Each of the two anion vessels contains approximately 70 ft3 of resin. Each of the two cation vessels contains approximately 50 ft3 of resin. The dual anion and cation train configuration provides redundancy in ion exchange capability and the resin ratio ensures that both resin vessels will be spent at approximately the same time.

2.4. Corrosion surveillance

The practices recommended for water quality [3] includes corrosion surveillance. The purpose of a corrosion surveillance programme (CSP) in a research reactor facility is to provide early detection of corrosion of components, structures and/or the nuclear fuel in contact with the water. Typically a CSP involves the exposure of a set of test coupons, or rack9 to the basin water for a predetermined period, followed by its detailed metallurgical evaluation for corrosion. Figure 6 shows a design of a galvanic couple subset of a corrosion surveillance assembly. Figure 7 shows a photograph of an assembly following removal from the SRS L-basin.

Techniques which utilize electrochemical noise sensors have also been used to conduct corrosion surveillance, but to a much less extent. The CSP also involves the determination of water parameters at periodic intervals. The combined results from the evaluation of corrosion coupons and the water quality parameter measurements provide the basis for evaluation of corrosion on the fuel cladding and other structural materials due to exposure to the water during the time of immersion of the materials. The results also include effects of any transients in water parameters, something that often goes unnoticed in the absence of a CSP.

The IAEA publication on practices recommended for water quality [3] includes protocols for corrosion surveillance.

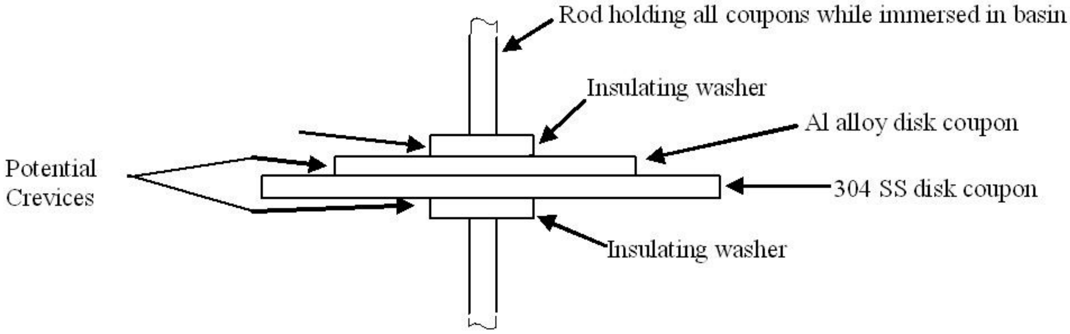

FIG. 6. Schematic diagram of a galvanic coupon subassembly showing the crevice between the coupons and between the coupons and washers.

FIG. 7. A coupon assembly used in a research reactor corrosion surveillance programme with 32 and 70 mm diameter coupons. The white area between the coupons is a PTFE separator, used to isolate the central stainless steel rod from the coupon and the coupons from one another.

2.5. Handling heavily damaged fuel

2.5.1. *Storage of breached fuel in water basins*

The L-basin activity is dominated by the concentration of 137Cs in the basin water as determined by chemical analysis of basin water samples. Therefore, modelling release rates from aluminium spent nuclear fuel (Al-SNF) into the basin water concentrates on the release rate of 137Cs. In general, the release of radioactivity from Al-SNF with breached cladding into water is dependent on several factors:

- Area of exposed fuel;
- Environment (temperature and quality of the water);
- Radioisotope content of fuel (enrichment, burnup, and decay time);
- Fuel meat material (post-irradiation composition and microstructure);
- Clad material.

At the low temperatures typical of basin storage (approximately room temperature), corrosion is the primary mechanism whereby species from the fuel core are released into the water. That is, diffusion transport of species from regions in the fuel core to the exposed fuel surface and direct release is not significant.

The radionuclides are assumed to be fully soluble and free to disperse into the water, and not bound in the corrosion product. A simple model to estimate the release from fuel core has been developed by considering general corrosion of the fuel core region directly exposed to the environment [4]. The release model is given by (1):

$$R = A \times B \times C \tag{1}$$

where:

R is the ^{137}Cs release rate [Ci/hr];

A is the ^{137}Cs activity density in the fuel meat material at the decay time of interest [Ci/cm^3];

B is the area of fuel exposed to the environment (area of breach) [cm^2]; and

C is the general corrosion rate of the fuel core material in the environment of exposure [cm/hr].

2.5.2. Steady state release into the basins

The impurity specie will ultimately reach a steady-state concentration in a basin that has a constant source of impurity specie addition, or release rate, into the water with an online deionization system for the basin. That is, at t = infinity, the concentration of an impurity specie $C(t) = C_{SS}$. For a closed loop system the steady-state concentration of radioactivity in the water, C (in Bq/L), is given by (2):

$$C_{SS} = R/(Q\ \varepsilon) \tag{2}$$

Where Q is the volumetric flow rate in L/s into and out of a closed loop deionization system, R is the total release rate of radioactivity into the water in Bq/s, ε is the efficiency of the deionizer system and C_{SS} is the activity of the water in Bq/L at steady-state.

Once again, this method can be applied to any specie of interest in a reactor water system, recalling that the efficiency of most deionization systems is 100% ($\varepsilon = 1$) until the resin has reached its capacity.

2.5.3. Mitigating severely breached fuel in wet storage

The DOE complex has a long history of handling a wide variety of damaged fuel types. Placing a damaged fuel assembly in an isolation canister has been a common solution to contain radioactive isotopes that can be released from breached-cladding (primarily ^{137}Cs). Experience has shown that isolation canisters with special design features are effective in isolating breached or damaged SNF from the bulk basin water.

Isolation canisters for damaged SNF are designed to allow underwater storage of the highly damaged fuel while sequestering the radionuclides that is released from it. Containing the released radioactive material is important from the standpoint of radiological protection of basin operations personnel and waste reduction realized from lower frequency of regeneration/replenishment of basin deionizer resin beds. A common design feature in the early SRS-design isolation canisters was a J-tube in the lid of the canister, as shown in Fig. 8. The J-tube acts in combination with evolved gas within the oversized containers to separate the internal water environment from bulk basin water. Gases released from damaged SNF build up at the top of the can and into the J-tube. Figure 9 shows a schematic of the new lid design that incorporates the inverted J-tube. The improved design of the inverted J-tube isolation canisters were developed to mitigate problems with pluggage due to silt build up within the early J-tubes [5]. In employing isolation canisters for the storage of damaged/degraded spent nuclear fuel, the canister provides the necessary and critical design features that were once performed by the fuel structure and cladding.

Historically, SRS chose isolated storage in water for the significantly damaged and destructively tested fuel, separating it from the remainder of the basin water by loading it into large water filled canisters. The method of storing damaged fuel underwater was to place fuel pieces in small diameter canisters (some aluminium and some stainless steel) which were grouped with others and placed in larger aluminium vessels referred to as oversized (OS) canisters (Figure 8).

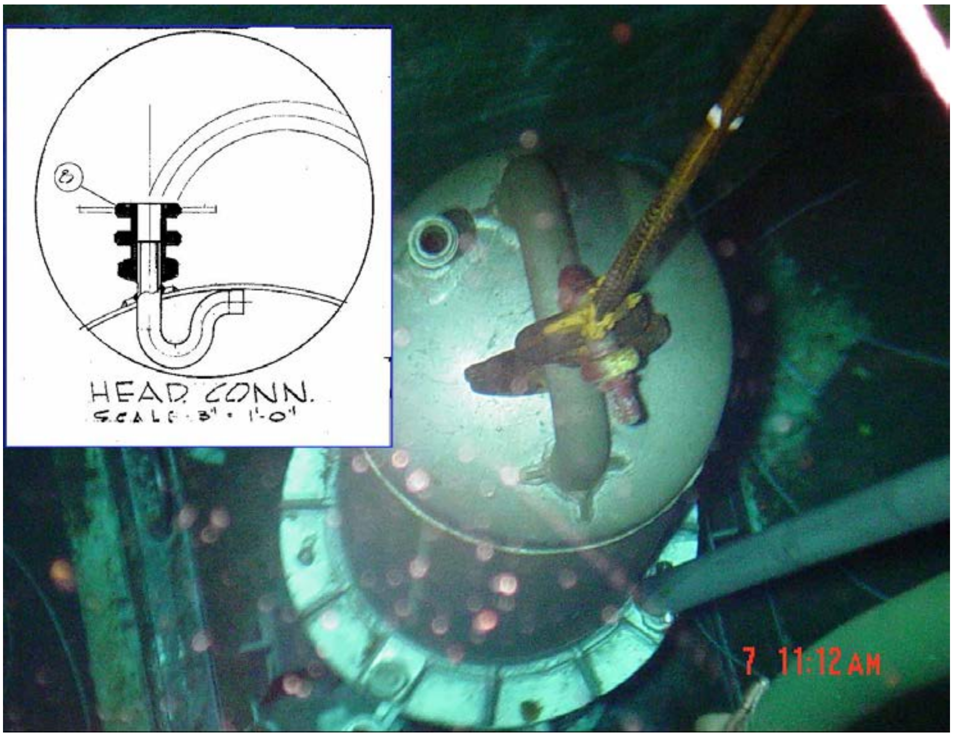

FIG. 8. Schematic illustration of a J-tube common to early SRS isolation canisters.

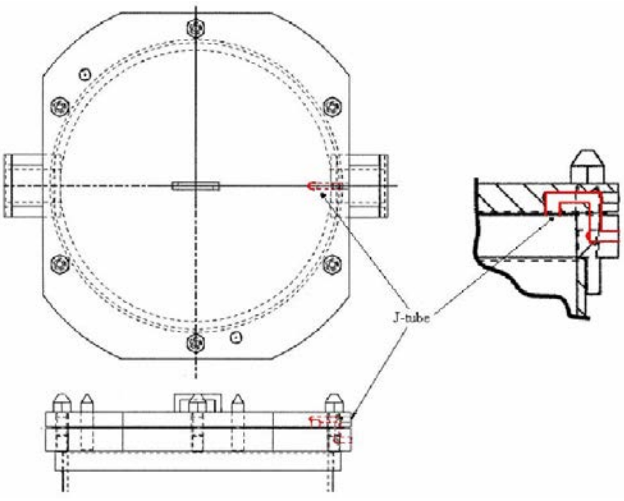

FIG. 9. Later inverted J-tube design on damaged fuel storage can.

The SRS experience with these oversized isolation canisters illustrates the benefit and the potential liability with the employment of isolation canister technology. The isolation canister effectively isolates the bulk basin water from the water within the isolation canister. However, the water within the isolation canister may become increasingly more contaminated with fission products from the ongoing fuel degradation process. This becomes significant if there comes a need to open the isolation canister as in the case of the SRS oversized isolation canisters. This need arose during the deinventory activities at the SRS Receiving Basin for Offsite Fuels (RBOF). To remove locally the highly contaminated water and minimize the impact on the bulk basin water activity upon opening the OS canisters, SRS developed and deployed a system to clean the water within the isolation canister through a submersible filtration and deionization system.

3. UNDERWATER DEIONIZER

The required parts of the underwater deionizer in its simplest form were: a pump and motor, filter, and an ion-exchange column, as sketched in Figure 10a. The system footprint was minimized to maximize flexibility and mobility of the deionizer system. The system was uniquely designed with the appropriate amount of cation resin to ensure maximum ion exchange efficiency. Positive displacement, water lubricated gear pumps (redundant) were powered by non-lubricated air motors, allowing variable speed operation and protection against locked-rotor situations. The design was checked for code compliance, fabricated, and fully tested before delivery to RBOF.

The redundant pumps and their corresponding motors were fixed to a 0.71 x 0.91 m skid and plumbed together with stainless steel pipe, having a common suction from the filter. Each pump was followed by a check valve before joining to a common flow meter. The check valves allowed either pump to operate singly, without recirculating water to the suction, or simultaneously, if desired. Figure 10b shows the underwater deionizer as it was positioned in the SRS RBOF basin, 4.6 m below the surface of the water. The site personnel planned and performed, OS canister flushing using the submersible deionizer with excellent results. During the operation, dose rates in the general basin water were maintained below the limit of 0.02 mGy/hr. The deionizer removed radioactive material from OS canister water allowing damaged fuel to be moved into more modern storage containers with the improved inverted J-tube and designed to fit the racks in the destination facility.

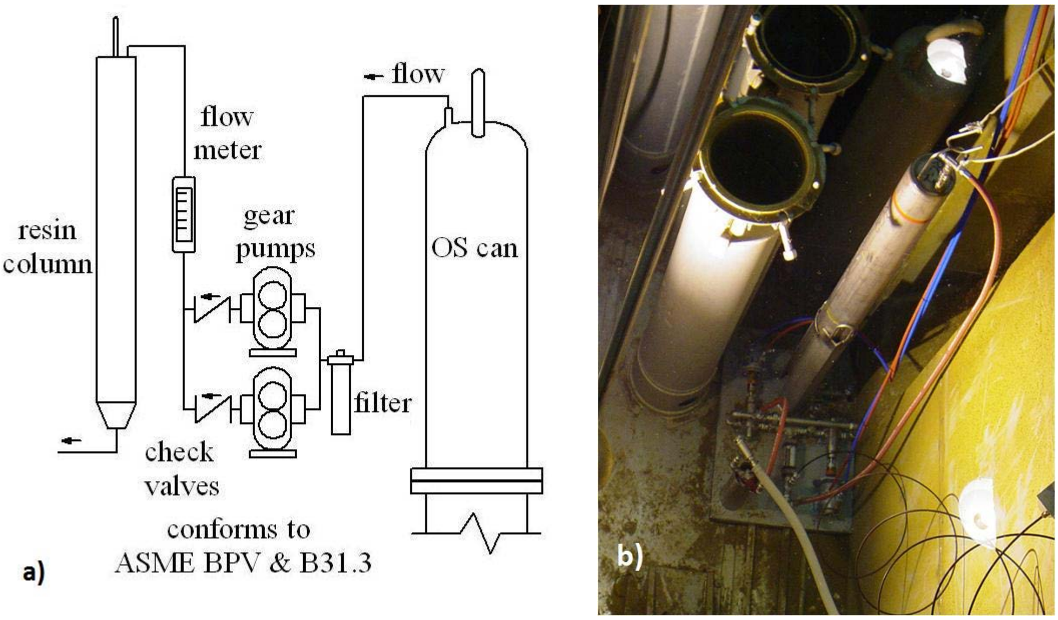

FIG. 10. Underwater deionizer flow diagram (a) and on RBOF with empty oversized canister halves (b).

4. CONCLUSIONS

The aluminium fuel storage experience to date in the USA, supported by the understanding of the effects of environmental variables on materials performance, demonstrates that storage systems that minimize degradation and provide full retrievability of the fuel up to several decades can be achieved in wet storage facilities. Continued surveillance and evaluation of the fuel and storage system materials to verify the predicted impact of the environment on materials behaviour remains paramount to ensure reliable safe storage throughout the storage period.

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THE STORAGE OF POWER DEVELOPMENT AND RESEARCH REACTOR FUEL AT SELLAFIELD

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Abstract

Sellafield Limited has extensive experience of building and operating spent nuclear fuel storage facilities on the Sellafield site. Since the first operation in 1952, a total of six storage facilities have been built in support of reprocessing spent fuel. Currently, four of these facilities are operational and two are undergoing decommissioning activities. Whilst the routine spent fuel operations are primarily associated with managing Magnox, Advanced Gas Reactor and LWR fuel from power generation reactors, management services to other fuel types are offered. Examples of these services include the storage of British naval training reactor fuel; the reprocessing of two skips of aluminium clad uranium metal fuel from Swedish AB SVAFO and the management of fuel from the UK Power Development Programme. The current paper provides an account of the management of the UK's Power Development Programme fuel stored on the Sellafield site. The fuel has been pond stored for up to 42 years and periodic inspection during this time has revealed no significant deterioration of the fuel, particularly that which has been containerised during its storage period. The paper also outlines some of the issues associated with the recovery and transfer of long stored fuel and assessment of the fuel storage can longevity if the material is not reprocessed.

1. INTRODUCTION

Sellafield Limited has extensive experience of building and operating spent nuclear fuel storage facilities on the Sellafield site. Since the first operation in 1952, a total of six storage facilities have been built in support of reprocessing spent fuel. Currently, four of these facilities are operational and two are undergoing decommissioning activities.

Whilst the routine spent fuel operations are primarily associated with managing Magnox, Advanced Gas Reactor and LWR fuel from power generation reactors, management services to other fuel types are offered. Examples of these services include the storage of naval training reactor fuel [1]; the reprocessing of two skips of aluminium clad uranium metal fuel from AB SVAFO [2–3] and the management of fuel from the UK Power Development Programme.

Currently, there are four operational spent fuel storage facilities on the Sellafield site used in support of reprocessing activities; First Generation Oxide Fuel Storage Pond (FGOFSP), First Generation Advanced Gas Reactor Storage Pond (FGAGRSP), Fuel Handling Plant (FHP) and Thorp Receipt & Storage (TR&S).

Spent fuel from the UK's power development programme was/has been received and stored primarily in the First Generation Oxide Fuel Storage Pond (FGOFSP).

The current paper provides an account of the management of the UK's Power Development Programme fuel stored on the Sellafield site, more specifically in the First Generation Oxide Fuel Storage Pond (FGOFSP).

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2. THE FIRST GENERATION OXIDE FUEL STORAGE POND (FGOFSP)

The First Generation Oxide Fuel Storage Pond (FGOFSP) is the oldest operational spent fuel storage facility at Sellafield. It was commissioned in 1967/68 and according to the current Long Term Plan (LTP) it is due to be decommissioned by 2020.

The original facility comprised a single receipt building and an outdoor storage pond. Early fuel receipts were in wet transport flasks; containing individual fuel assemblies in transport frames or baskets. Upon receipt each fuel assembly was transferred into open storage skips. In the case of post irradiation examination (PIE) material (whole or part pins and residues in cans), this was transferred dry to the facility where a flask to flask dry transfer was undertaken before the unloading flask was flooded and unloaded wet. Fuel assemblies or PIE cans were stored in open fuel skips.

In the 1970s the FGOFSP was progressively extended with the addition of a new receipt facility and three further outdoor ponds, or storage bays. During this period problems were encountered with 'crud' migration[33] during LWR flask discharge. To combat this problem Sellafield Limited developed over a number of years, a containerised transport and storage system for LWR fuel [4]; commonly known as the multi-element bottle (MEB), shown in Fig. 1. The interim step in this process was individual bottles used in transport and a rack variant for storage.

FIG. 1. Multi-element bottle (MEB).

Later, in 1980, a larger fifth outdoor storage bay, shown in Fig 2., was added to accommodate LWR fuel destined for the Thermal Oxide Reprocessing Plant (Thorp). As the interface, Thorp Receipt & Storage was based on handling MEBs therefore these new facilities were primarily designed around the handling of MEBs. The FGOFSP ceased fuel receipts from NPPs in 2004.

3. POWER DEVELOPMENT AND RESEARCH REACTOR FUELS STORED AT SELLAFIELD

The principal research/prototype fuels stored at Sellafield are those arising from the UK Power Development Programme, namely the Windscale Advanced Gas Cooled Reactor (WAGR), Sellafield and Steam Generating Heavy Water Reactor (SGHWR), Winfrith. This programme also brought in smaller quantities of fuel which had been irradiated in research reactors and from international collaboration projects; for example the BR3 project.

The fuel cycle policy for UK origin/title fuel is a commercial judgement of the owners of the spent fuel [5]. In the case of spent fuel covered by the UK power development programme the original owners, the UK Atomic Energy Agency, signed a contract with British Nuclear Fuel plc for storage and reprocessing services; this contract has since been inherited by the Nuclear Decommissioning Authority.

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[33] Crud (Chalk River Unidentified Deposits) – common name used to describe any corrosion product/material adhering to the outside of a water reactor fuel assembly.

FIG. 2. 5th Storage Bay FGOFSP.

The current plan, as reflected in the Sellafield Lifetime Plan, is still to reprocess all this fuel, however, it is recognized that for the following reasons a proportion of the material may yet be deemed as non-reprocessable for the following reasons:

- Difficult to separate chemically (e.g. MOX, carbide);
- Impact on plant throughput;
- Difficult to put into a reprocessable format/cost of conditioning;
- Plant modifications would be required to enable it to be reprocessable;
- Volume of material is too small;
- Cannot make an economic safety case.

For contingency planning purposes an assessment of the longevity of these fuels and the compatibility to store alongside AGR fuel, which is planned to be interim stored in Thorp Receipt & Storage facility, have been carried out.

3.1 Windscale Advanced Gas Cooled Reactor (WAGR)

The Windscale Advanced Gas Reactor (WAGR) was a carbon dioxide cooled, graphite moderated reactor, fuelled with slightly enriched uranium dioxide fuel clad in stainless steel. The reactor was built as a prototype to study the advanced gas cooled power reactor system, and to serve as a test bed for the development of fuel element designs and other components. It operated from 1963 to 1981 with the first fuel being committed to pond around 1967.

The fuel irradiation experiments conducted in WAGR covered a wide range of variables including fuel elements of alternative designs, fuel cladding with thinner walls, grooved pellets or plutonium-bearing fuel [6]. The work directed at improving fuel cladding included studies of:

- The effects of manufacturing processes on the ductility of the cladding;
- The resistance of the cladding to thermal cycling;
- Influence of neutron irradiation on the mechanical properties at elevated temperatures.

By 1980 around five main variants of cladding including different cladding finishes had been investigated. The early variants were cold worked and suffered from thermal sensitization/carbon pick-up. It is noted that the commercial AGR is now at the Mark 53 fuel pin. There were also variations to the annular bore diameter and to the enrichment (0.4–12.5%).

All WAGR fuel has been dismantled with the large majority of pins subjected to some form of post irradiation examination. The fuel was originally packaged either as intact pins in slotted cans (open to the bulk pond water) or sealed screw top cans. Part pins/fuel debris were accommodated in fully welded cans (see Fig. 3). There are around 900 dismantled fuel cans currently stored in the FGOFSP.

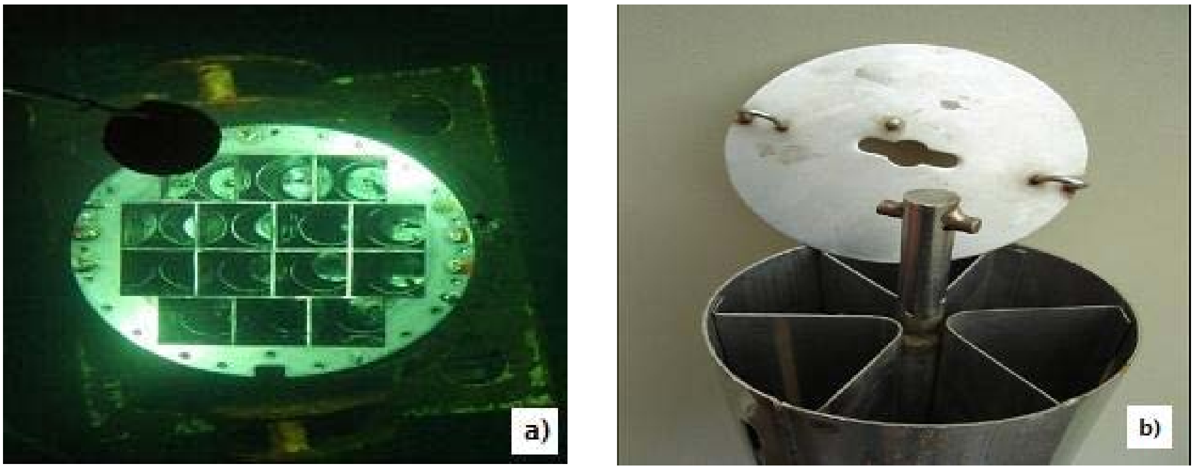

FIG. 3. PIE stored in an MEB (a), PIE storage can (b).

3.2 Steam Generating Heavy Water Reactor (SGHWR)

The Steam Generating Heavy Water Reactor (SGHWR) used a pressure tube construction, cooled by light water, moderated by cool un-pressurized heavy water, and fuelled with rod cluster elements. The fuel was clad in Zircaloy-2. The reactor operated from 1967 to 1990 and the first SGHWR was received at Sellafield in 1972. This fuel and subsequent receipts were committed to storage in the First Generation Oxide Fuel Storage Pond with the last shipment being received in 1994.

The majority of the fuel has been received as intact fuel assemblies and was transferred into Multi-Element Bottles (MEBs) upon receipt (Fig. 4). The older receipts are accommodated either in open fuel skips or individual bottle racks (IBRs); these fuels are in the process of being transferred to MEBs in readiness for reprocessing. Both MEBs and IBRs isolated the fuel from the bulk pond water.

Intact SGHWR fuel is in the form of 36 pin, 57 pin and 60 pin assemblies. Variations which were made to SGHWR fuel assemblies during development were largely limited to changes in fuel pin length and fuel pin enrichment with little variation in the cladding material. One variant included MOX fuel.

3.3 Other power development programme fuels

Examples of other fuels stored from the power development programme include: Whiteshell (Fig. 5) from the WR-1 reactor (Zircaloy clad, 2.2% ^{235}UO2 fuel) and Vulcain from the BR3 programme in Belgium; the latter is stainless steel clad fuel with enrichments up to 9% ^{235}U.

FIG. 4. Top nozzle of a SGHWR rod cluster.

FIG. 5. Top nozzle of a Whiteshell fuel assembly.

4. CHARACTERISTICS OF SPENT FUEL STORAGE AT SELLAFIELD

The main objectives of spent fuel storage are:

- To maintain integrity of the primary containment barrier (i.e. the fuel cladding) to avoid the loss of fission products and fuel to the environment.
- To maintain the original fuel element (or bundle) geometry, fuel pin integrity or approved conditioned fuel package in order to avoid variations in fuel reactivity and to ensure that sub-criticality is maintained.
- To maintain the fuel element or conditioned fuel package structural integrity to avoid fuel drops during handling and to allow fuel to be safely removed from one facility to another.
- To maintain the fuel clad and/or fuel package in a condition that does not compromise future spent fuel conditioning.

These objectives are achieved through an understanding of:

- The end-of-life fuel properties or its condition at the beginning of storage.
- A knowledge of the degradation mechanisms that can occur during pond storage.
- The basic principles which should be adopted in plant safety cases to ensure that fuel or conditioned package integrity is maintained during storage.
- The importance of a good quality assurance programme, with specific instructions to assure the sub-critical condition of the facility.
- Clear guidance and recommendations on the water chemistry to be adopted.
- Recommendations, suggested monitoring programmes and pond conditions (for example temperature).
- Adoption of best practice.

Such guidance has been provided through the generation of Technical Standards for LWR, AGR and Magnox fuels. A technical standard to cover non-standard fuels is being compiled.

4.1 Nuclear safety (sub-criticality analysis)

The nuclear safety criterion adopted to ensure that all spent fuel storage ponds remains sub-critical, is:

$$K + 3\sigma \leq 0.95$$

All analysis is based upon pre-irradiation enrichments.

Assessments cover intact fuel assemblies, part fuel assemblies, dismantled assemblies as intact pins in a storage can and cut pins/debris in storage cans. Assessments are storage module specific against a set of hazards to determine if the threshold is exceeded.

Examples of the sort of hazards which have been considered for spent fuel in the FGOFSP safety case include:

- Omission of neutron absorber from the storage module[34];
- Displacement of the neutron absorber in the storage module3;
- Corrosion of the neutron absorber material;
- Out-of-specification neutron absorber material;
- Fuel assembly corrosion;
- Incorrect fuel/storage module3 combination;
- Introduction of a fuel assembly with an increased reactivity due to altered fuel assembly geometry;
- Dropping of fuel assembly on a PIE can;
- Fuel damage as a result of a mechanical handling accident;
- Explosion inside a storage module3 causing fuel damage;
- Fuel in a storage module3 becoming uncovered as a result of loss of pond water;

Assessments have been carried out for fuel up to 20 w/o ^{235}U which bounds the materials associated with the UK power generation programme stored at Sellafield.

4.2 Storage performance/behaviour

To date the majority of long stored fuel assemblies have been found to be in good condition. There are, however, a number of fuel assemblies either received as non-standard items or have been damaged during fuel handling operations that require remedial action to put them in an appropriated format for reprocessing.

4.2.1 WAGR

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[34] Storage module refers to either a multi-element bottle, Individual bottle racks or open fuel skips.

WAGR fuel was retrieved and inspected in hot laboratories at Sellafield between 1977 and the mid 1980s. The initial examination was carried out on fuel which had been stored in the FGOFSP for between 2 and 10 years. The sample of pins was selected from those most likely to have failed during storage either because of long storage times or because they had been irradiated at high clad temperature (~750°C). Earlier experience at Oak Ridge National Laboratories (USA) had identified that unstabilised stainless steel fuel clad in operating temperatures of 430–650ºC was susceptible to intergranular attack. In the case of the WAGR fuel inspected, leaks had developed in the cladding of some pins which had cold-worked clad and which had been stored in open skips for 3–4 years; two types of corrosion were noted: fine intergranular attack (fuel irradiated at low temperature) and broad front attack (fuel irradiated at high temperature). Failures were only observed in fuel with mean stringer irradiations >15GWd/tU; only a fraction of WAGR fuel experienced irradiations to this level.

From the early 1980s WAGR fuel was received in IBRs and/or MEBs which isolated it from the bulk pond water and protected it from the atmospheric chloride to which the older fuel in open skips is exposed. For commercial AGR, the fuel has been committed to caustic dosed ponds to protect the fuel from the environment.

There is no evidence from activity release modelling or bulk pond water sampling (<3Bq/ml) to suggest that further fuel degradation occurs or that any gross problems with leaking fuel exist.

4.2.2 SGHWR

As SGHWR fuel is zircaloy clad it would not be expected to degrade in the relatively benign storage environment of the FGOFSP, even in the presence of up to 3ppm chloride. Recent visual inspection of the assemblies has shown that their condition after ~30 years in wet storage is mainly very good with limited corrosion visible around the pin tops and very little evidence of corrosion along the rest of the assembly length. A very small quantity of assemblies have been found with individual fuel pins having ruptured cladding, hydride blisters (see Fig. 6) and/or other physical damage sustained at reactor (e.g. buckled spacer grids).

FIG. 6. SGHWR assembly showing hydride blisters or sun burst.

4.3 Transfer of long stored fuel to Thorp receipt & storage in support of reprocessing activities

Whilst the introduction of MEBs for the transportation and storage of LWR fuel has had many benefits, the longer term impact on FGOFSP operations has left a legacy of fuel in older storage containers/skips requiring rebottling. This has required fuel handling systems (which have become redundant with time) and procedures for individual fuel handling to be re-established. Secondly, as some fuels/dismantled fuel cans will have been subject to atmospheric chloride ingress, those mechanisms which could lead to failure of the fuel handling systems are of concern and warrant integrity checks prior to transfer; this is provided through CCTV inspection for signs of corrosion.

Thirdly, while detailed records exist regarding the irradiation experiments that were undertaken at WAGR and SGHWR, the condition of the fuel at the beginning of storage is less well documented.

Current inspections of SGHWR fuel have revealed a small proportion (~2%) of assemblies with mechanical damage such as damage to spacer grids or sparge tubes which were not documented and could present a problem to downstream plants (e.g. when feeding to shear). It is not known whether these examples of damage were sustained at reactor or have occurred as a result of pond storage. The inspection results are being documented in line with the current inspection programme to ensure downstream plants are aware of the issues (and can militate against them).

4.3.1 Thorp receipt and storage (TR&S)

The Thorp Receipt and Storage facility (TR&S) shown in Fig. 7, was commissioned in 1988 and was designed for the receipt and handling of containerised irradiated fuel. All LWR fuel stored in this facility is contained in Multi-Element Bottles (MEBs), and dismantled AGR fuel is stored in skips/containers. Fuel in PIE cans is currently excluded from the facility to prevent it from being fed to the dissolvers.

TR&S, which contains demineralized water at pH7, is currently being developed to act as a bridging option for AGR fuel storage while the consultation process on the management of fuel in the UK is completed. The design basis is to enable fuel storage for up to 80 years which will require dosing the pond water with caustic as a corrosion inhibitor, which will raise the pH to 11.5, or with sodium nitrate.

FIG. 7. Storage pond Thorp Receipt & Storage.

4.3.2 Nuclear safety assessment in support of transferring SGHWR fuel

An SGHWR fuel assembly is considered to be less capable of withstanding impact loads in comparison with standard PWR/BWR assemblies; early Zircaloy-2 clad. Finite Element Analysis (FEA) has shown that for the worst case hypothetical impact scenarios considered in these safety cases, 23% of the fuel pins could fail. Whilst this represents a significant departure from the position with PWR/BWR fuel assemblies where a much lower pin failure fraction has been substantiated, the radiological safety case for LWR fuel still bounds SGHWR movements as the fraction released from long cooled fuel is significantly lower than short cooled high burnup LWR fuel.

Criticality modelling of the SGHWR fuel assemblies in the event of a flask impact has been carried out and has confirmed that an impacted flask with 14 assemblies would remain safely sub-critical.

4.4 Long term storage assessment

4.4.1 SGHWR

At pond storage temperatures, fission product attack and any redistribution of zirconium hydrides can be discounted. Similarly, stress corrosion cracking of the zircaloy is discounted as no credible mechanism can be identified that would cause high concentrations of ferric chloride to accumulate in surface cracks caused by hydride precipitates.

The main degradation mechanism that remains in play during pond storage is general corrosion and at pond temperatures there is no distinction in corrosion behaviour between zircaloy-2 and zircaloy-4. As there is no corrosion data available on the performance of zircaloy in sodium nitrate, coupon trials and high resolution corrosion probes trials are currently under way and have been run for several months.

Irradiation hardening/hydride embrittlement has occurred due to in-reactor service. No further deterioration in mechanical properties during pond storage is expected.

4.4.2 WAGR

Assessment work regarding the longevity of fuel cans and WAGR material has been carried out which concluded that cans fabricated using 304L stainless steel would not be susceptible to corrosion. Older cans are fabricated from S321 stainless steel and it has been assessed that they may suffer knife edge attack.

As cans are over-packed into 304L stainless steel sheaths for reprocessing, the ability to retrieve cans in the longer term remains with the sheath which should remain unaffected. In addition movement to a dosed facility will minimize/inhibit any further corrosion.

5. SUMMARY

The current paper provides an account of the management of the UK's Power Development Programme fuel stored on the Sellafield site. The fuel has been pond stored for up to 42 years and periodic inspection during this time has revealed no significant deterioration of the fuel, particularly that which has been containerised during its storage period.

The paper has also outlined some of the issues associated with the recovery and transfer of long stored fuel and assessment of the fuel storage can longevity if the material is not reprocessed.

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STORAGE AND MANAGEMENT OF SPENT FUEL IN HANARO RESEARCH REACTOR

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Abstract

HANARO, the only operating research reactor in the Republic of Korea, generates 5 spent fuel assemblies after 28 day operation at 30MWth. After a visual inspection, the discharged fuel is stored in the spent fuel storage pool. As of August 2009, the spent fuel pool accumulates 326 assemblies, not including 299 spent fuel rods from 2 TRIGA reactors which operated before HANARO, and which were sent back to the USA in June 1998, as decommissioning projects of the reactors are underway. The spent fuel pool of HANARO can store 1032 spent fuels, equivalent to 20 years of operation of the reactor, plus test fuels, which have been locally irradiated. The spent fuels are stored in modules supported by racks and modules can be piled up in 3 layers. The storage pool is a heavy concrete structure with an internal stainless steel lining. Demineralized water is used for radiation shielding and cooling purposes. An independent cooling and purification system maintains the water temperature below 40 ^{o}C, electric conductivity below 5 µS/cm and pH between 5.5 and 6.5. The pool is equipped with a radiation monitoring system and an IAEA camera is used for surveillance. To accommodate our future needs we consider three options: To use the space previously available to accommodate the fuels from the shut down TRIGA type reactors; to expand the storage capacity, by changing the design of the storage module, or to return the spent fuels to the USA, taking advantage of the take-back programme. Another option under consideration is to store research reactor spent fuels together with the spent fuels from nuclear power plants (NPPs). This issue is under discussion and a new national policy for spent fuel management is expected to be defined soon, in a timely manner through national consensus by public consultation.

1. INTRODUCTION

HANARO is an upward flowing light water cooled, heavy water reflected open-tank-in-pool type research reactor of 30MWth. Its first criticality was achieved on Feb. 8, 1995 and normal power operation started in January 1996. Since 1996, HANARO has been widely used for radioisotope production, material and fuel irradiation tests, beam application research and NAA (neutron activation analysis). HANARO uses 19.75 % enriched U_3Si-Al fuel, and it has two types of fuel assemblies. One is a circular shaped assembly with 18 fuel rods, and the other is a hexagonal shaped assembly with 36 fuel rods.

The reactor core, shown in Fig. 1 has an inner part, with 23 hexagonal and eight circular flow channels, separated by flow tubes, and an outer part with 8 circular flow channels. Thirty-two fuel assemblies are loaded into the core; 28 in the inner core (in all positions of the inner core, except positions CT, IR1 and IR2), and four in the outer core (positions OR3 through OR6). These three additional hexagonal positions of the inner core, and four circular holes in the outer core are reserved for irradiation and experiments.

2. SPENT FUEL STORAGE SYSTEM

The spent fuel storage system, used to store the spent fuel from HANARO reactor is composed by a spent fuel storage pool, an independent cooling and purification system, and a water monitoring programme [1].

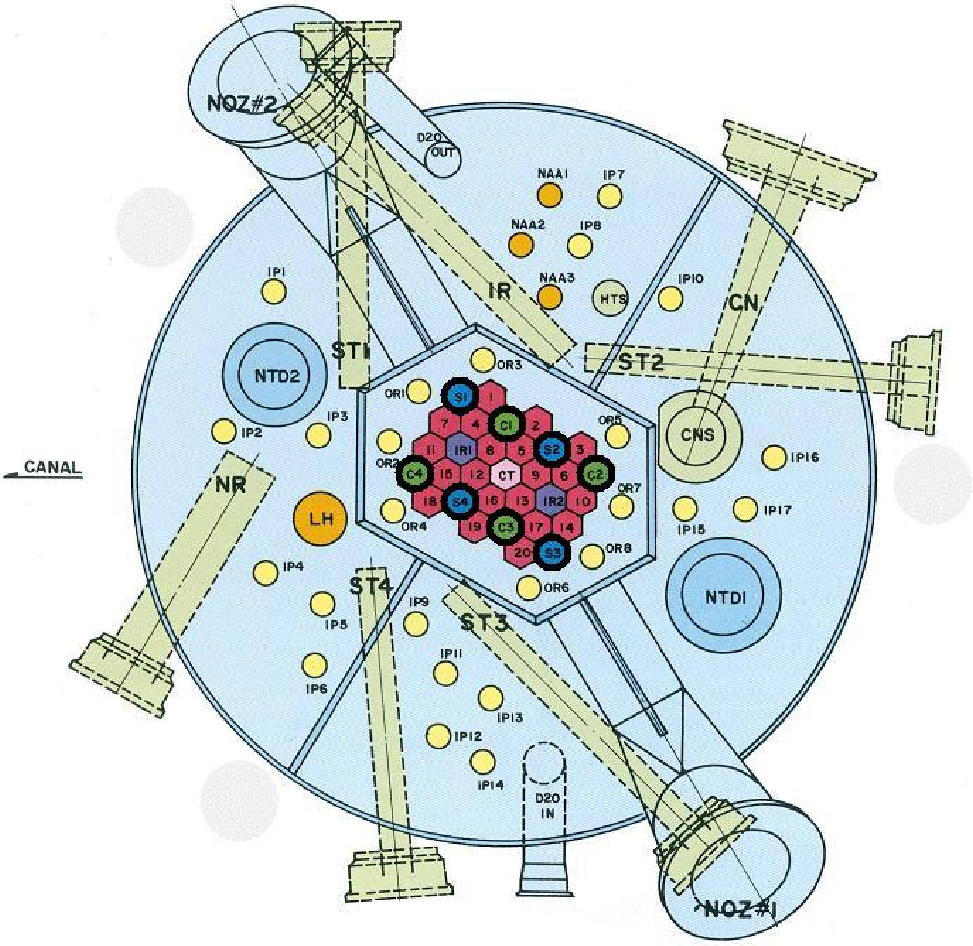

FIG. 1. Plan view of HANARO reactor.

2.1. Spent fuel storage pool

In HANARO reactor, the spent fuels are stored in the spent fuel storage pool, which is connected to the reactor pool by way of a service pool, as shown in Fig. 2a. At first, the spent fuel is transferred to the service pool for visual inspection. When the visual inspection is concluded, then the fuel is transferred to the storage pool. The dimensions of the spent fuel storage pool, shown in Fig. 2b, are 4 m x 4 m x 13.4 m. It is a heavy concrete structure, internally lined with stainless steel plates. The thickness of the stainless steel plates is 9 mm, and the thickness of the heavy concrete is 1.5 m. The concrete structure was designed and constructed as Seismic Category I.

Three storage racks are placed in the bottom of the storage pool, in order to support and pile up the storage modules. The storage rack is a freestanding type, and three layers of fuel storage modules can be piled up in each storage rack. The structures of the storage racks and fuel storage modules are made of stainless steel. The storage racks are divided into eight compartments for 36 rods fuel assemblies and six for 18 rods fuel assemblies, with three additional compartments for TRIGA type fuels. Storage modules for 36 rod assemblies can accommodate up to 25 elements; storage modules for 18 rod assemblies can accommodate up to 24 elements, and modules for TRIGA type fuel can accommodate up to 35 elements. Therefore, considering that we can have up to three layers of storage modules piled up, the total storage capacity is 600 for 36 rod assemblies, 432 for 18 rod assemblies, and 315 for TRIGA type fuels. This is enough capacity for spent fuels generated during normal operation of HANARO research reactor during 20 years.

The fuel is stored in "storage cells" placed in the storage module to protect spent fuels, and a funnel with some slope is attached at the upper part of the cell to allow easy insertion of the fuel. An aperture of 74 mm in diameter is installed at the lower part of the cell to allow cooling the fuel by natural circulation.

The pitches of the storage cells in a module, to maintain it in sub-critical condition are: 150 mm for 36 rod assemblies, 129 mm for 18 rod assemblies, and 95 mm for TRIGA type fuels. The distance between 2 consecutive layers is higher than 300 mm. The storage pool is filled with demineralized water to remove decay heat and to shield the radiation produced by the fuel. The storage pool has an independent cooling and purification system. Whenever it is needed, demineralized water from the reactor supply system is added to the spent fuel storage pool, to compensate eventual losses caused by evaporation. The quality of the water in the storage pool is always maintained within the specified limiting values, to prevent the fuels and the stainless steel lining plate from corrosion.

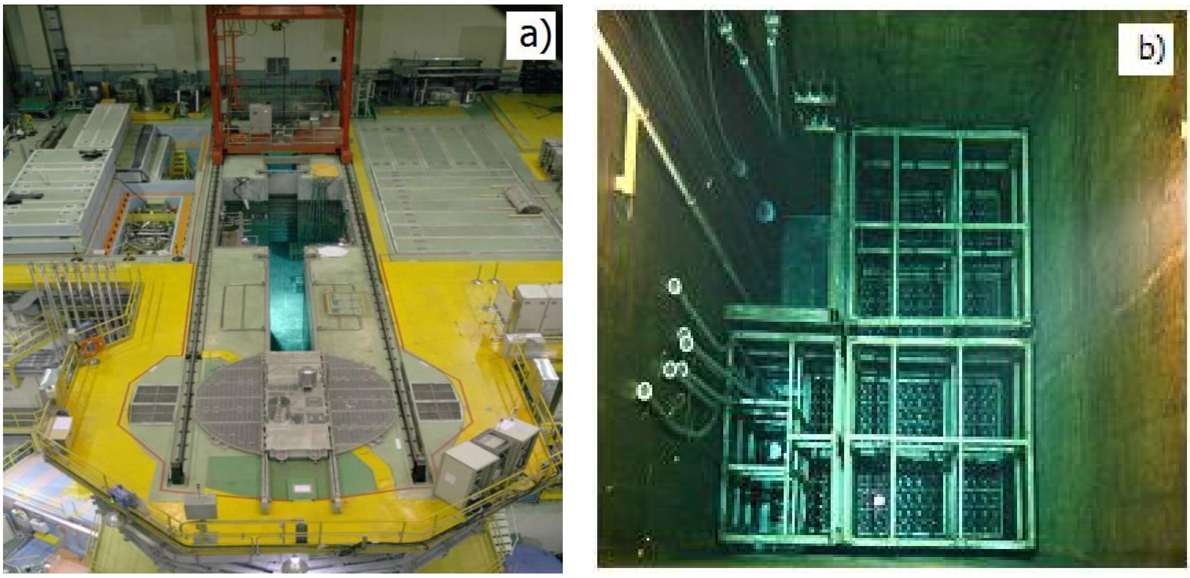

FIG. 2. (a) Reactor hal; (b) spent fuel storage pool.

2.2 Cooling and purification system

The cooling and purification system of the spent fuel storage pool removes decay heat and purifies the storage pool water to maintain the water quality as required. The system, as shown in Fig. 3, consists of two pumps, one heat exchanger, one ion exchanger, and one normal filter. The two pumps (one in standby) maintain the pool water circulation through the heat exchanger, in order to remove the maximum heat of 0.1MW, and keeping the water temperature below 40 ^{o}C

The water flow rate in the cooling circuit is 5 kg/sec. After passing through the heat exchanger, 50% of the water flow rate passes through the purification system, before returning to the storage pool.

The purification system is composed of a normal filter and ion exchanger beds. Ball type isolation valves are installed upstream and downstream from the filter and ion exchanger to take samples. When the normal filter and the ion exchanger need to be repaired, or replaced, the ball type valves allow the isolation of the purification system.

Water samples can be taken at the inlet and outlet of the ion exchanger and the normal filter. Performance and replacing time of filter and resin can be checked by online monitoring of conductivity and pressure. The water level of the storage pool is maintained using demineralized water supplied through a line directly connected to the storage pool. When the pool water level reaches a predefined low level, an alarm indicating the low level sounds in the control room, and the valve of the demineralized water supply line is automatically opened to allow demineralized water into to pool. When the water level reaches a predefined high level, the alarm for the high level is again sounded in the control room and the valve is automatically closed.

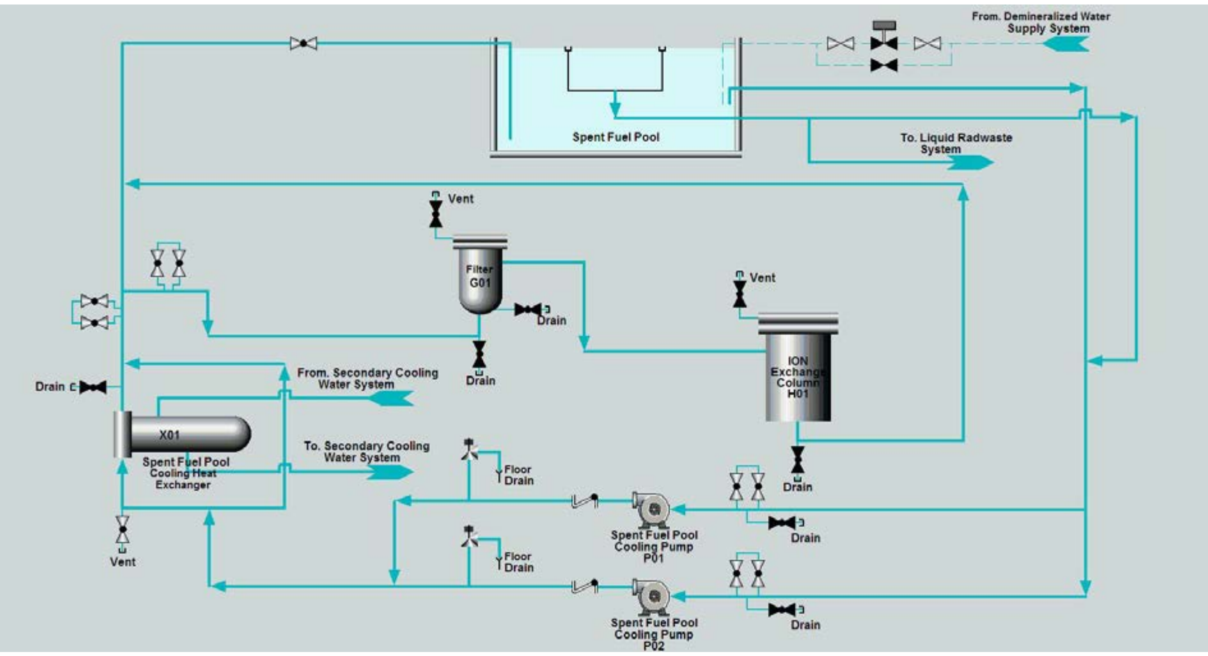

FIG. 3. Spent fuel storage pool cooling and purification system.

Instruments are installed in the cooling and purification system of the spent fuel storage pool for the following functions:

- Alarms for low and high levels of water in the storage pool;
- Measurement of water temperature in the storage pool and at the outlet of the heat exchanger;
- Measurement of coolant flow;
- Indication of coolant pump failure;
- Indication and alarm of eventual coolant leakage from the heat exchanger;
- Measurements of pressure drop at the filter the ion exchanger;
- Measurements of conductivity of water at inlet and outlet of heat exchanger.

2.3. Water monitoring programme

HANARO operates with a full time 6-shift system. During reactor operation, four operators, consisted of a shift leader and three operators, operate and inspect the reactor systems. The spent fuel storage pool is continuously monitored for some of the parameters listed in Table I, to maintain good water quality. The first six values of parameters listed in Table I are recorded and confirmed by the shift reactor operators every eight hours, together with other parameters of reactor systems. The chemical analysis group checks the pool water by sampling every six month for Cl^-, F^- and turbidity, as indicated in Table I. The nuclide analysis in the water of the storage pool is also performed monthly. Conductivity is maintained below 1.0 µS/cm even though allowable value is below 5.0 µS/cm. If conductivity and pressure differences are larger than 1.0 µS/cm and 143 kPa, resin of ion exchanger and filter are replaced.

TABLE I. MONITORED PARAMETERS OF WATER IN SPENT FUEL STORAGE POOL AND IN COOLING AND PURIFICATION SYSTEM

Parameters	Allowable value
Water level	12.28 ~ 12.5 m
Heat exchanger inlet and outlet temperature	10 ~ 40 °C
Pressure difference of ion exchanger in and out	≤ 140 kPa(g) + 10%
Conductivity of ion exchanger in and out	≤ 5.0 µS/cm
Pressure difference of filter in and out	≤ 130 kPa(g) + 10%
PH	5.5 ~ 6.5
Cl and F	< 0.2 µg/ml
Turbidity	≤ 1 NTU

3. SPENT FUELS

HANARO operates on the basis of a 28 day operation cycle. After every operation cycle 5 fuel assemblies are discharged from the core. In general, a fuel assembly remains in the core for about 180 days and then it is discharged with maximum burnup of 54 % ^{235}U. As for august 2009, 326 spent fuels have been discharged from the reactor, after a total reactor operation time equivalent to 49,407 MWD, and stored in the spent fuel storage pool. The average burnup of the stored fuels is about 44 %^{235}U. This burnup is very low compared to the maximum allowable discharge burnup of 54 %^{235}U. The reason is that normal operation of HANARO started in 1996, and in the early days of reactor operation, the fuel assemblies could not endure as designed because of the occurrence of abrasion of the spacer supporting and assembling the fuel rods caused by flow induced vibration. After some design changes the fuel assemblies started being discharged with designed burnup [2].

Actually, the 326 spent fuels from HANARO stored in the pool represent one third of its total capacity, not considering the modules reserved for TRIGA type. In some parts of the racks, as shown in Fig. 4, the modules already have three layers piled up, with IAEA installed safeguard seals to prevent moving any module.

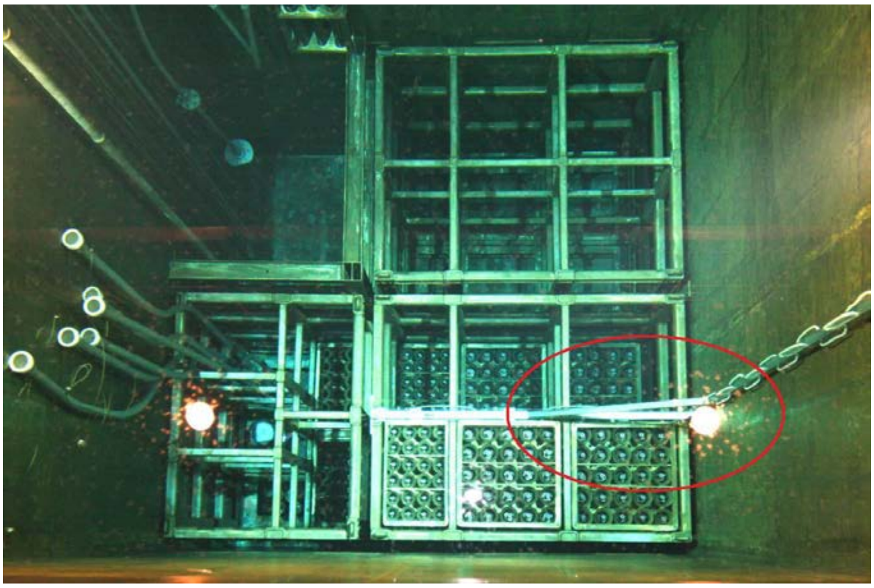

FIG. 4. IAEA safeguard sealed racks.

4. FUTURE PLAN

Considering the operating regime of HANARO research reactor, we anticipate that in 2023, the spent fuel storage pool will be filled with spent fuels, and since HANARO can operate beyond 2023, a study has been started to expand its spent fuel storage capacity. The main possibilities are:

(1) Use the space available for storage of TRIGA type spent fuel in the spent fuel storage pool: Before HANARO construction, two TRIGA reactors had been operated in the Repbulic of Korea. These reactors were shut down and are now under decommissioning. Space for 315 TRIGA reactor fuels was made available in the spent fuel storage pool of HANARO to accommodate the spent fuel from these TRIGA reactors, but all the 299 spent fuel rods from the two TRIGA reactors were sent back to the USA in June 1998. So, the space available for TRIGA type fuels can be used to store the HANARO spent fuels by simply changing the modules.

(2) Expanding the storage capacity by changing the design of the storage module to a new one with narrow pitch between cells containing spent fuel: Existing modules were designed using very conservative criticality calculations. Cells made of stainless steel tube with 3.0 mm thickness containing the spent fuel was not considered in the stage of design. Considering the thickness of the cell, the pitch of cell to cell can be reduced. The analysis results show that if this is done, the existing capacity to store the spent fuel in the pool can be multiplied by 1.6 [3].

(3) Take-back programme: Most of HANARO fuel use US origin enriched uranium, and considering that we participated in the shipment operation of the 299 TRIGA type fuel elements returned to the USA, this means that we already have experience in the take-back programme, and can use this experience to return HANARO spent fuels to the USA. It depends only having the necessary financial support to cover the transport operation.

Finally, it is important to mention that the Korean government considers spent fuel interim storage an important issue, especially for the spent fuels from nuclear power plants (NPPs). The current practice has been to store the spent fuel generated from NPPs in each plant, by expanding its local storage capacity. However, considering that this may not be the best solution, the government started discussing a national policy to deal with the nuclear spent fuel. Several options are been considered, including the construction of a specific facility to store the spent fuel. The final decision will be reached by national consensus, after a public consultation.

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MANAGEMENT AND STORAGE OF RESEARCH REACTOR MARIA SPENT FUEL

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Abstract

The description of management of highly enriched spent nuclear fuel of research reactors EWA and MARIA, of the Institute of Atomic Energy POLATOM in Świerk (Poland) is presented in this paper. Considering that EWA research reactor has been shut down and is under a decommissioning process, the paper emphasizes the spent fuel from MARIA, a research reactor that operates since 1974 is presented. The conditions of spent storage legislative requirements and methods of controls are also presented. Special attention is given on presentation of current methods utilized for protection of spent fuel damaged by corrosion. Finally, the internal plans related to storage of research reactor MARIA spent fuel for the near future is also presented.

1. INTRODUCTION

In the Institute of Atomic Energy POLATOM of Świerk, there are two research reactors. The first one is EWA, which started operation in 1957, was shut down in 1995 and is partially decommissioned. The second research reactor in IAE is MARIA, which is in operation since 1974.

Reactor EWA was originally a 2 MW pool-type reactor. In 1964 the reactor had its power level increased to 4 MW, in 1967 to 8 MW, and in 1972 to 10 MW. Operating until 1995, the reactor used 10% and 36% enriched fuel, having been used for scientific investigation mainly, and for irradiation of target materials for radioisotopes production. Originally the reactor used EK-10 fuel elements as nuclear fuel. After sufficient modernisation of the reactor core internals and cooling systems, in 1967, WWR-SM fuel assemblies started to be used. After 1990, the core became a mixture of fuel assemblies WWR-SM and WWR-M2. After discharge, the spent fuel was kept in the reactor pool for a few months, and then transported away from the reactor site to an external storage facility. In the beginning of 1995, reactor EWA was defueled, and in 1997, decommissioning operations began. Reactor EWA was operated for approximately 3500 hours per year during 37 years without any significant events [1].

Reactor MARIA is a water and beryllium moderated, graphite (canned in aluminium) reflected and water cooled reactor of pool type with pressurised fuel channels containing concentric multi-tube MR fuel assemblies. Maximum thermal power of each fuel assembly is limited to 1.8 MW, and the cooling capacity of the reactor is 30 MW, which is the nominal power of the reactor. The reactor is a multipurpose high flux reactor, operated mainly for research purposes, using neutron horizontal beam and irradiation of various materials in the reactor core. Maximum neutron thermal flux can reach 2.5×10^{14} n/cm^{2}sec^{-1} in the fuel region, and 4.05×10^{14} n/cm^{2}sec^{-1} in the beryllium region. The first criticality of the reactor, designed and constructed by Polish industry, occurred in 1974, and it was operated normally until July 1985, when it was shut down for modernisation. After a radical modernisation, reactor MARIA returned into operation in 1993. In 2009, it was decided to use the reactor for production of Molybdenum. The adjustments were made, and we expect to start the commercial production by the beginning of 2010. The reactor has a concrete biological shielding lined with stainless steel plate, and a core basket made of aluminium alloy. Figure 1 shows a schematic view of MARIA research reactor [1].

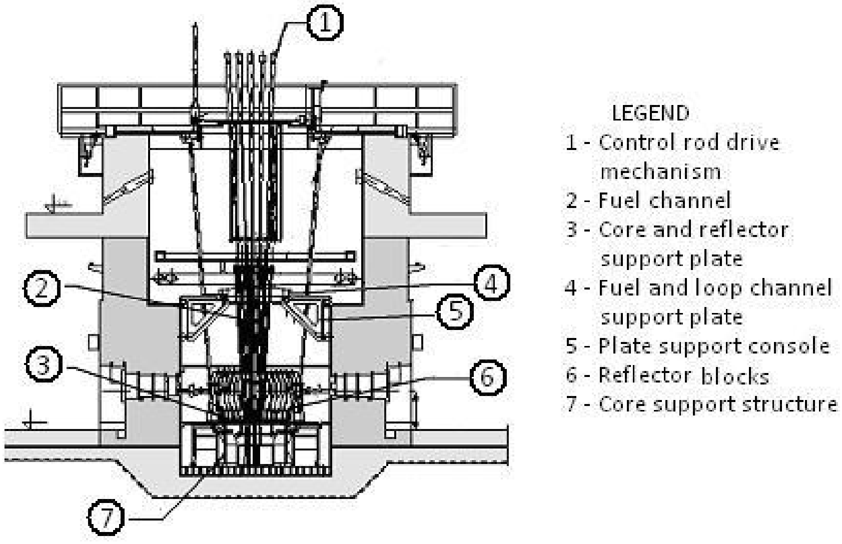

FIG.1. Schematic view of MARIA research reactor [2].

2. FUEL CHARACTERISTICS [1]

2.1. Fuel used in EWA research reactor

EWA reactor used three types of nuclear fuel, EK-10, WWR-SM and WWR-M2. EK-10 consists of an aluminium tube, with diameter 10 mm, filled with a dispersion of UO_2 in magnesium. Each fuel element contains 81.4 grams of uranium enriched up to 10% in ^{235}U. The active length of the fuel mixture is 500 mm. The overall length of the fuel rod is 595 mm. The thickness of the clad is in the interval 1.0 – 1.5 mm. The estimated burnup of each EK-10 rod is 1 MWd.

WWR-SM consists of three concentric fuel tubes. The outer tube has a hexagonal form, and the other two are cylindrical. The fuel tube is a “sandwich” type. Between two aluminium layers (thickness 0.9 mm), there is a meat of fuel consisting in a dispersion of uranium in aluminium with thickness 0.7 mm. Each fuel assembly contains 108 grams of 36% enriched uranium. The length of the fuel mixture is 600 mm and the overall length of the assembly is 865 mm. The reactor core was designed as a hexagonal lattice with pitch equal to 35 mm. The estimated burnup of a single fuel assembly is 14 MWd.

WWR-M2 has the same dimensions as WWR-SM assembly, except for the quantity of uranium in single assembly; in WWR-M2 it is equal to 124.5 grams of uranium enriched up to 36%. Thickness of the fuel mixture is 0.9 mm and thickness of the aluminium clad is 0.8 mm. The estimated burnup of a single fuel assembly is 16 MWd.

2.2. Fuel used in MARIA research reactor

MARIA reactor is fuelled with pressurised fuel channels. Each fuel channel, also known as “technological tube”, contains one multi-tube MR fuel assembly, with six (MR6) or five (MR5) concentric cylindrical fuel tubes. Total fuel channel length (field tube type) is 4.7 m, as shown in Fig. 2. The fuel assembly is only 1.38 m long, and it is located in the lower part of the fuel channel. The fuel tube is a “sandwich” with a fuel dispersion contained within aluminium cladding. The active length of fuel mixture is 1000 mm, and overall length of fuel assembly is 1.38 m. The outer diameter of fuel assembly is 70 mm. In the period of 1974 – 2000, the fuel was 80% enriched (this fuel is denoted as MR6 (80%) – which means six concentric fuel tubes with 80% enriched fuels or MR5 (80%) – which means five concentric fuel tubes with 80% enriched fuel). From 2000, the new fuel used was MR6 (36%) 550 – which means six concentric fuel tubes with enrichment up to 36% and

550 grams of ^{235}U in each fuel assembly. Since January 2005, MARIA has been fuelled with MR6 (36%) 430 – six fuel tubes with 36% enriched fuel and 430 grams of ^{235}U [1].

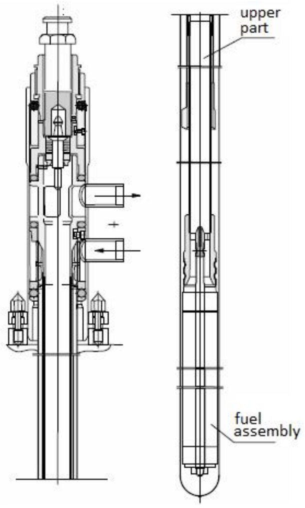

FIG. 2. View of MARIA reactor fuel channel.

3. SPENT FUEL INVENTORY

3.1. Spent fuel from shutdown reactor EWA

As explained before, EWA used three types of nuclear fuel, EK-10, WWR-SM and WWR-M2 fuel types. Table I lists the final inventory of spent fuel from the reactor operation. This spent fuel is temporarily stored in wet spent fuel storage pool 19A, described below, and actually it is been prepared for exportation to Mayak factory, in the Russian Federation, for reprocessing. According to the original plans, the first shipment was scheduled to be completed before the end of 2009.

TABLE I. SPENT FUEL INVENTORY FROM EWA RESEARCH REACTOR

Type of fuel	Number	Enrichment of fresh fuel [%]
EK-10	2595	10
WWR-SM	2095	36
WWR-M2	445	36

3.2. Spent fuel from operational reactor MARIA

MARIA reactor utilizes MR fuel type, and the fuel inventory produced to date is shown in Table II.

TABLE II. SPENT FUEL INVENTORY FROM MARIA RESEARCH REACTOR

Type	Number	Content of ^{235}U [g]	Enrichment [%]	Average burnup [%]
MR-5	15	330	80	35
MR-6 (350)	297	350	80	35
MR-6 (430)	52	430	36	55
MR-6 (540)	49	540	36	45

4. SPENT FUEL STORAGE FACILITIES

All spent nuclear fuel from EWA and MARIA is stored in two places of the Nuclear Centre of Świerk. One place is a wet spent fuel storage pool named 19A and the second place is the wet spent fuel storage pool of reactor MARIA, also known as "technological pool". Actually the storage pool 19A has all fuel from reactor EWA and also a part of the spent fuel from reactor MARIA. This storage is under supervision of the Radioactive Waste Management Plant. The storage pool of reactor MARIA has only spent fuel from reactor MARIA and this storage is under supervision of Reactor Operation Department, which is part of the Institute of Atomic Energy POLATOM.

Storage pool 19A became operational in 1968. The design of the storage facility was based on the assumption that the spent fuel could be received after a minimum of three months cooling following the discharge from reactor EWA. The storage facility consists of two parts: an underground part made of reinforced concrete and a part made of bricks situated above ground level. Two storage ponds are situated in the concrete block. Each of them has the following dimensions: length 300 cm, width 270 cm and depth 550 cm. The ponds are lined with 6 mm stainless steel (1H18N9T). The pool is filled with demineralized water. The cross section of storage pool 19A is shown in Fig. 3. Spent fuel assemblies are stored in racks made of aluminium. The storage facility contains a ventilation system and activity monitoring system. A movable water treatment plant is used at least once a year, or whenever the conductivity of water exceeds 3 µS/cm. There is no continuous water circulation in the storage pool [1]

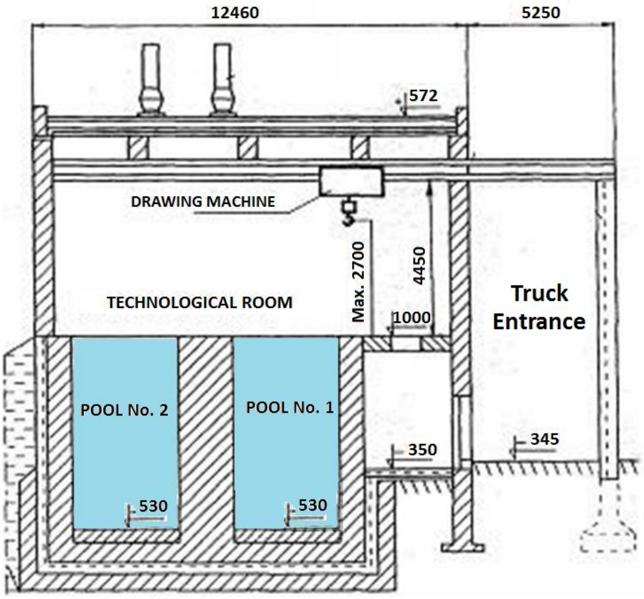

FIG. 3. Cross-section of spent fuel storage pool 19A.

The "technological pool", used to store spent fuels from research reactor MARIA, is a pool adjacent to the reactor pool. Dimensions of the pool, shown in Fig. 4, are: 650 cm of height, 350 cm of width and 1235 cm of length. For storage, the spent fuel elements are placed into storage racks, shown in Figure 5, that have a regular mesh (distance between two consecutive spent fuel elements) of 154mm, in both directions. This mesh assure sufficiently low multiplication factor. The neutronic calculation for this rack, done with utilization of WIMS-D4 computational code for MR-6/540g fuel is shown in Fig. 6 [3].

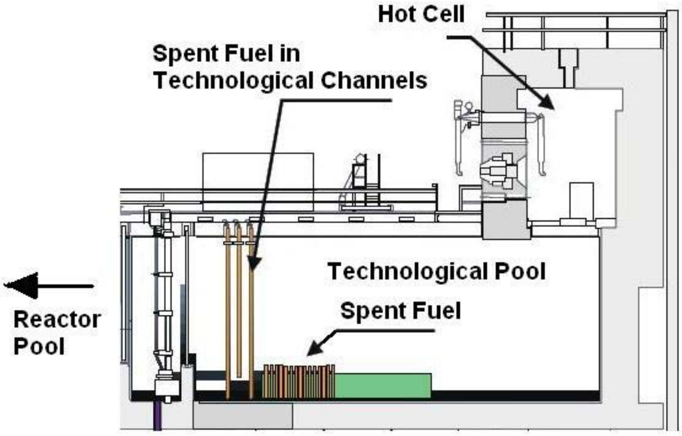

FIG. 4. View of Maria nuclear spent fuel storage pool.

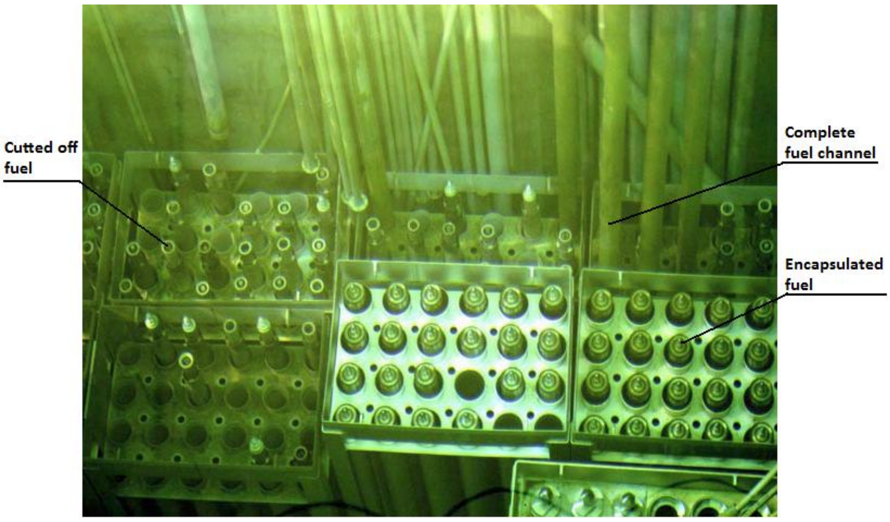

FIG. 5. View of Maria spent fuel located in spent fuel storage racks.

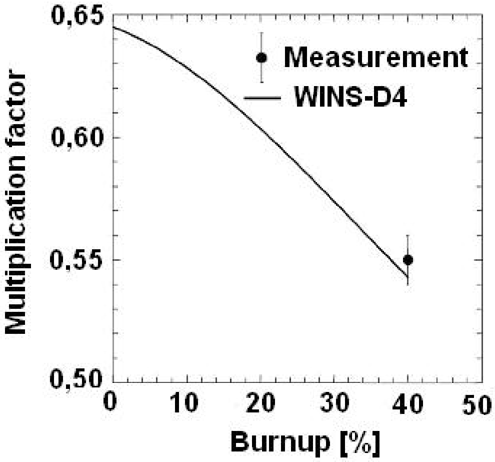

FIG. 6. Multiplication factor considering infinitive mesh for fuel MR-6/540g ^{235}U [3].

4.1. Standard procedure to store the spent fuel from MARIA research reactor

After being discharged from the reactor core, the spent fuel assembly (also known as fuel channel) is transferred into the spent fuel storage pool and situated in aluminium racks near the pool wall. They have to be kept under the water for at least 30 months to allow decay heat removal. After this time the temperature of fuel cladding is below 100°C in the air.

As mentioned before, total fuel channel length is 4.7 m, and the fuel assembly, located in the lower part of the fuel channel, is only 1.38 m long. After a few years of spent fuel cooling the upper part of the fuel channel has to be cut off. This action is performed to facilitate the fuel assembly maintenance and cooling by clean water from the reactor storage pool. In complete (long) fuel channel the water is stagnant and in this condition the fuel cladding corrosion is more intensive. The aluminium clad used in fuel assemblies for reactor MARIA is made of SAW alloy of two millimetres thickness. To cut off the upper part of the fuel channel, it is utilized a very specific tool, designed especially to allow the operation to be made underwater, as shown in Figure 7.

FIG. 7. View of process to cut off upper part of fuel channel.

4.2. Water quality in spent fuel storage pool

The storage pool is filled with demineralized water. To prevent corrosion of the aluminium clad fuel, the water in the pool has to be carefully controlled. Table III shows the water quality parameters of the deionised water in storage pool.

TABLE III. PARAMETERS OF DEIONISED WATER IN STORAGE POOL

Parameter		Value
conductivity of water		<3µs/cm
pH index		between 5.5 and 6
content of impurities:	chlorides	< 0.02 mg/l
	sulfides	< 0.05 mg/l
	copper	< 0.02 mg/l
	aluminium	< 0.05 mg/l
	iron	< 0.05 mg/l
	oxygen	< 0.05 mg/l

5. LEGAL ISSUES

In Poland, all activities related to nuclear fuel and materials are regulated by Atomic Law Act. It is an Act of the Parliament, dated 29 November 2000 – Atomic Law (O.J. of 2001 No 3, item 18), and in force since 2002-01-01. The Act enumerates all activities that require a license from (or should be at least notified to) the Regulatory Body. Those activities include manufacturing, conversion, storage, disposal, transport or use of nuclear materials, radioactive sources, radioactive waste and spent nuclear fuel. They encompass also the construction, operation, closure and decommissioning of disposal facilities for radioactive waste, and disposal facilities for spent nuclear fuel, as well as construction and operation of storage facilities for spent nuclear fuel.

Specific issues related to spent fuel management are described in: Directive of Council of Ministries about radioactive waste and spent fuel, issued 3 December 2002 (OJ Dz. U. 2002, no 230, item 1925), in force since 2003-01-01.

After the establishment of the Governmental Strategic Programme for "Management of Radioactive Waste and Spent Fuel in Poland", signed in 1997 a programme for systematic inspections of the fuel was implemented at Swierk. The programme defined several undertakings and research and development (R&D) projects, which resulted in specific activities. One of them was related to the management of the spent fuel from Polish research reactors. The R&D project conducted with reference to the spent fuel included the following activities: characterisation of the existing inventory of fuel as a function of fuel type, age in storage and burnup; characterisation of the physical conditions of the fuel (with underwater video records and eventually ultrasonic characterisation of pitting profiles); identification of leaking fuel elements (sipping tests): development of technology for encapsulation of damaged fuel elements: definition of criteria of extended interim storage, and recommendation to the Government on final solution [4].

6. CONDITIONS OF SPENT FUEL STORED IN SPENT FUEL WET STORAGE

Following the decision to characterize the spent fuel, the "Hot Cell" of the storage pool was adapted, and used to investigate of the condition of some chosen spent fuel elements used in EWA and MARIA reactors. The inspection showed negligible corrosion effects on fuel elements wet stored less than 15 years. For fuel elements with longer storage time, uniform and pit corrosion of the fuel cladding were observed. In the case of very old fuel elements, close to 30 years, advanced corrosion was evident [5]. Long term storage in water environment led to cladding surface degradation, caused by corrosion. In some fuel elements this process led to leaks of fission products into storage facility water environment. An example of MR type spent fuel surface with pitting corrosion is presented in Fig. 8 [6].

FIG. 8. Surface of MR type spent fuel (80%) cladding monitored in hot cell before encapsulation.

The main cause of corrosion was pitting at random points. Pits from the surface contact of ions in the water, particularly chlorine, copper, iron and aluminium and corrosion process had developed in the fuel cladding. An overall measure of the ions content is given by the conductivity. Freshly deionised water has a conductivity of 0.5 µS/cm, which rapidly rises to 1 µS/cm due to chlorine and carbon dioxide from air interactions witch the water. Pitting is enhanced by galvanic action between aluminium and other materials, such as iron, and also by scratches, which remove the oxide layer.

Once identified the degraded condition of the spent fuel elements stored in the pool for more than 15 years, a decision was taken to dry store failed fuel elements, or assemblies, in capsules filled with Helium [5].

6.1. Verification of spent fuel integrity

As explained in Section 5.1, about 30 months after being removed from the reactor core, the upper part of the fuel channel is cut off, and now, as a standard procedure we verify the integrity of the cut off spent fuel element, before placing it on the storage rack. Control of cut off spent fuel integrity is performed in a special installation in spent fuel storage pool. The assembly is enclosed in a measuring cavity and the water from this cavity is pumped out. Next, the cavity is filled up with demineralized water and then, after mixing a sample of the water is taken and spectrometric analysed. A second sample of water is taken and analysed 24 hours later. According to the criterion established for fuel integrity, the fuel is considered:

- "integral" if it shows activity of Cs137 < 10 kBq/day and activity of Eu155 < 1 kBq/day;
- leaking if it shows 100 kBq/day > activity of Cs137 > 10 kBq/day;
- damaged if it shows activity of Cs137 > 100 kBq/day.

6.2. Encapsulation of leaking spent fuel

Once identified a leaking spent fuel, it is closed inside a hermetic stainless steel can, as shown in Figure 9. The encapsulation process indicates the leak-tight closing of the dried out spent fuel assembly in the stainless steel capsule and isolating the spent fuel aluminium clad from water. In the process of encapsulation of the MR type spent fuel it is necessary to distinguish the two locations of operations needed to be accomplished with the fuel, i.e. the storage pool and the hot cell of MARIA reactor (Fig. 10.). The most important part of the MR type spent fuel encapsulation process, namely the drying out, leak-tight closure in capsules and checking their tightness, is carried out in the hot cell of MARIA reactor

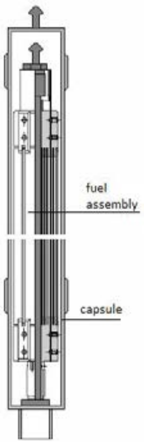

FIG. 9. View of encapsulated spent fuel [3].

After introducing the spent fuel assembly into the hot cell the process of encapsulation is to be performed.

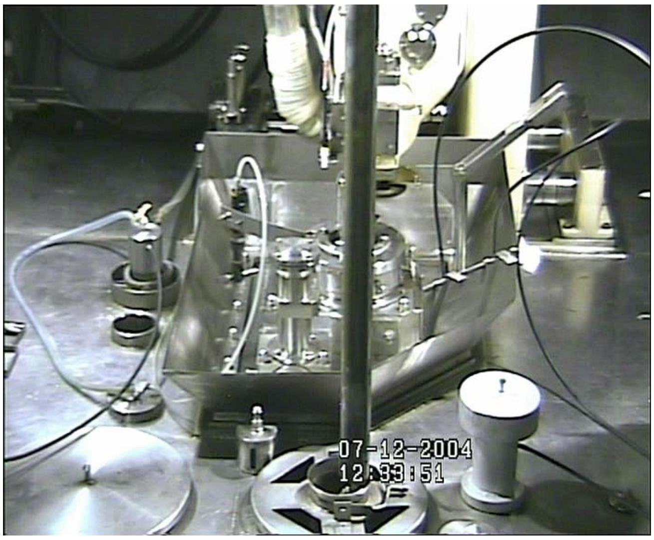

FIG. 10. Internal view of hot cell available in MARIA storage pool.

This encapsulation process involves the following actions:

(a) Fuel assembly drying up;
(b) Insertion of fuel into capsule and welding by TIG method, as shown in Fig. 11;
(c) Control of welding tightness by helium detector. The criterion for leakage tightness is less than 10^{-7} Pa/m^3/s;
(d) Filling up the capsule with helium through a special valve mounted in capsule cover. The helium pressure inside the capsule is 0.2 MPa, as shown in Fig. 12;
(e) The final tightness test of the capsule is to be carried out inside a vacuum cell. The criterion for leakage is the same as for welding control.

FIG. 11. View of welding process [6].

FIG. 12. Checking the capsule tightness [6].

Once it is encapsulated, the fuel is kept in reactor spent fuel storage pool of MARIA reactor or it can be transferred to 19A storage pond.

6.3. Proceeding with damaged spent fuel

Damaged spent fuel is also encapsulated using the same method as described above, but special procedure is applied for fuel cutting off and transport to the reactor spent fuel storage. This procedure assures limitation of spent fuel storage water contamination.

7. QUALITY ASSURANCE

Management of spent fuel is performed under Quality Assurance Programme for Reactor Operation. All activities related to spent fuel (cutting off, the tightness control, encapsulation, etc.) are performed on the basis of special written instructions and they are in accordance with Q.A.P.

8. SAFEGUARDS

The spent fuel storage at reactor MARIA and spent fuel storage 19A have physical protection system containing continuous monitoring system in the case of MARIA spent fuel storage. They are also under IAEA safeguard system.

9. PLANS RELATED TO STORAGE OF SPENT NUCLEAR FUEL FOR NEAR FUTURE

On 9 August 2006, the IAEA helped Polish authorities to remove close to 40kg of fresh highly enriched uranium (HEU) from Swierk. The HEU was safely airlifted back to the Russian Federation, which had originally supplied it to fuel Poland’s research reactors. The mission was a joint effort between the United States, Poland, the Russian Federation, and the IAEA, as part of the Global Threat Reduction Initiative (GTRI) to identify, secure and recover nuclear and radiological materials around the world [7]. Actually, there is no national plan for long term storage of nuclear spent fuel in Poland. All spent fuel from reactor EVA and all highly enriched fuel from reactor MARIA will be exported to the Russian Federation until the end of 2016, under GTRI. The next shipment is being planned for the end of this year, 2009. Also, there is no plan for long term storage of the low enriched fuel which is planned to be used after 2015.

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KINR EXPERIENCE ON STORAGE OF WWR-M RESEARCH REACTOR SPENT FUEL

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Abstract

WWR-M is a 10 MW, light water research reactor operational since 1960. The original reactor fuels were 90% enriched WWR-M5 and WWR-M7 fuel type assemblies. In the last decade a programme was developed to use 36% enriched WWR-M2 fuel type assemblies, instead of the 90% enriched fuel. Two types of WWR-M2 fuel assemblies are used in the reactor – single and triple type fuel assemblies. The main difference between them is that triple type fuel assembly is a mechanical combination of three single type fuel assemblies. The fuel assemblies stay in the reactor core for an average period of two years; afterwards it is unloaded and transferred to the spent fuel storage pool. At present, time the storage pool has irradiated assemblies that have been unloaded from the core since 1964. In total 175 triple and 103 single assemblies are stored in the pool. The oldest assemblies have been stored in the pool for more than 45 years. In the past several shipments were made directly to the Russian recycling plant, when the Ukraine was part of the former Soviet Union. The last shipment of spent fuel was carried out in 1988. Since 1988, the recycling plant has sent to Kiev 16 spent fuel transport casks that were properly identified and used for temporary storage of spent fuel. Current plans consider resuming shipping back the spent fuel to the Russian Federation, by the middle of 2010, using Skoda transportation casks. In order to meet the requirements for utilizing the Skoda casks, a new spent fuel storage facility has been built, and all spent fuels were transferred into it. This has enabled an inspection of the old spent fuel storage tank possible. In parallel, WWR-M reactor is undergoing a fuel conversion programme, to limit fuel enrichment to 20%.

1. INTRODUCTION

Nuclear research reactor WWR-M is a light-water moderated and cooled tank reactor with forced cooling. The maximum reactor power is 10 MW. Maximum thermal neutron flux is 10^{14} neutron/cm^{2}sec within the core region and $6*10^{13}$ neutron/cm^{2}sec in the reflector region.

The research reactor is situated in Kiev. It belongs to the Division of Nuclear Research Institute, of the National Academy of Science of Ukraine, and has been in operation since 1960. The reactor is used as a radiation source for radioisotope production; irradiation of silicon mono crystals; material testing experiments; experiments in radiobiology; neutron physics, nuclear physics, and applied physics experiments.

The reactor operational schedule is 5 days per week – about 100 hours per week. It has 9 horizontal experimental channels, a thermal column, 13 vertical isotope channels in the beryllium reflector, and availability for 10–12 vertical channels in core.

The vessel of the reactor is made of a 16 mm thickness aluminium alloy. The vessel has an outer diameter equal to 2300 mm, total height 5705 mm, and it is surrounded by concrete shielding. The vessel is filled with demineralized water to the level 5000 mm.

The reactor has a cover consisting of two rotating cast iron lids that are used during reactor reloading works. Figure 1 is an elevation drawing of the reactor building. It shows the reactor core within the vessel and the spent fuel storage area.

The coolant flows through the core downward with the maximum water temperature at the reactor output equal to 50° C. The nominal flow is 1300 cubic meters per hour.

The reactor core is composed of 36% enriched WWR-M2 fuel assemblies, supplied by the Russian Federation. Previously 90% enriched WWR-M5 and WWR-M7 fuel types, also supplied by the

Russian Federation, were used. Now the reactor core, which is surrounded by beryllium, is undergoing a fuel conversion programme, to limit the fuel enrichment to 20% [1, 2].

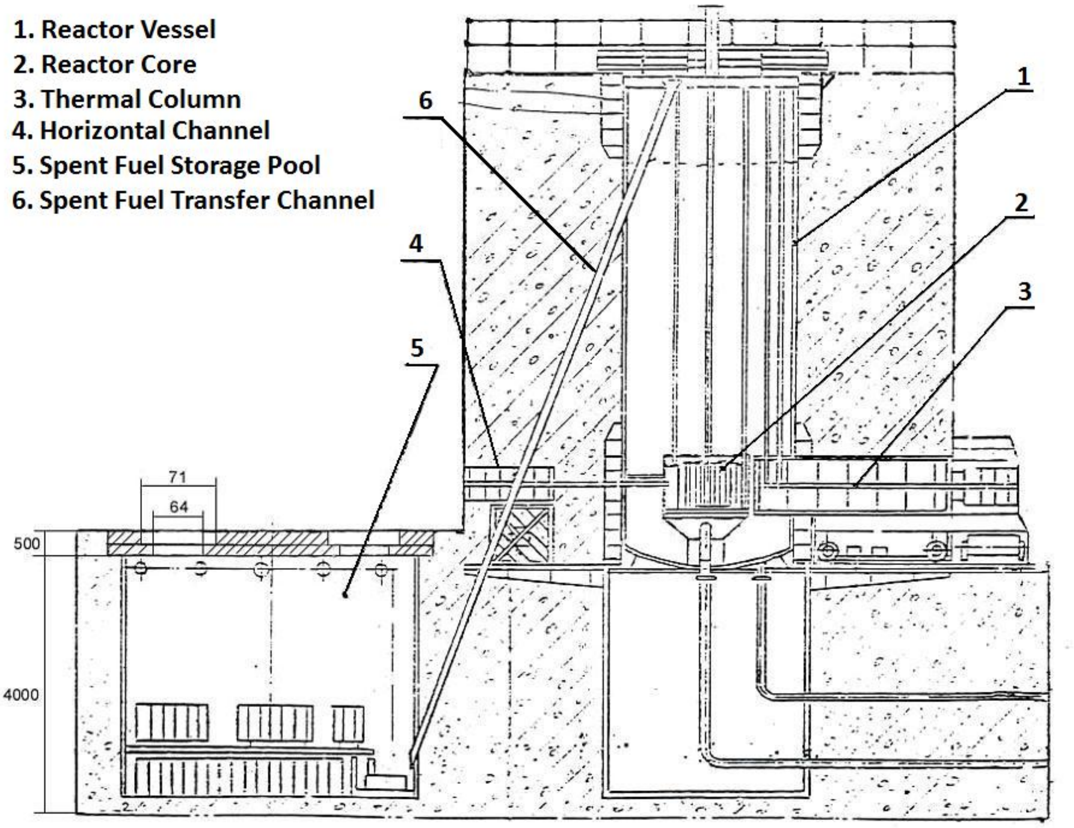

FIG. 1. Elevation drawing of WWR-M research reactor.

2. FUEL ASSEMBLIES

Two types of WWR-M2 fuel assemblies are used in the reactor — single and triple type fuel assemblies. Single type fuel assembly, shown in Fig. 2, consists of three concentric tubing-type fuel elements. The external tube has a hexagonal section and the two internal tubes have circular sections. The external fuel element is connected to the others at the top (head) and at the bottom (tail). The head has a special feature to enable capture and holding of the assembly by reloading tools, during refuelling operations. The tail serves for plugging the assembly into the grid of the reactor or within the grid installed in the storage facility.

Triple type fuel assembly is a mechanical combination of three three single type fuel assemblies, arranged as shown in Fig. 3.

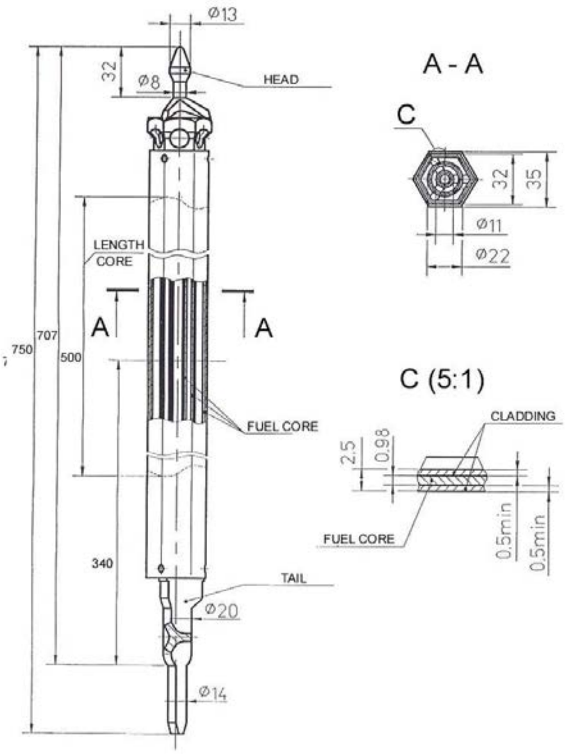

FIG. 2. WWR-M2 single type fuel assembly.

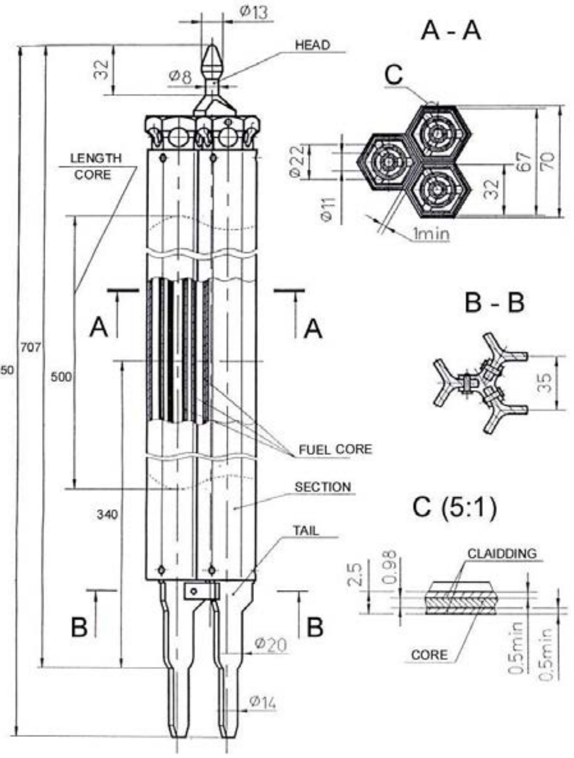

FIG. 3. WWR-M2 triple type fuel assembly.

2.1. Fuel elements

Each fuel element, that is a part of the fuel assembly, is a three-layer tube. The internal layer constitutes the fuel meat, with a thickness of 0.98 mm. The two outer layers are the fuel claddings, which are made of aluminium alloy with a thickness equal to 0.5 mm. These claddings protect the fuel meat from the environmental impact and prevent the release of fission products into the reactor coolant. The meat material is uranium dioxide, dispersed into an aluminium matrix. Total weight of a single type fuel assembly is 0.9 kg. Mass of uranium-235 in it is about 37 grams. As explained before, the triple type fuel assembly consists of three units (sections), each of them representing one single type fuel assembly, with the modified head details. The units (sections) are connected together in a form of a star, as shown in Figure 3. The total mass of a triple type fuel assembly is 2.6 kg. Mass of uranium-235 in it is 112 grams. The basic parameters of the fuel used in WWR-M research reactor are given in Table I. The table also provides parameters for the WWR-M5 (M7) fuel types, which were used in the reactor before the WWR-M2. Design of these assemblies differs from WWR-M2 by enrichment, number of fuel elements and thickness. The fuel matrix of the WWR-M5 was uranium-aluminium alloy.

3. SPENT FUEL STORAGE POOL

The spent fuel storage pool is located adjacent to the reactor, from level zero to -4.5 meters. Pool-storage, shown in Fig. 1, is filled with demineralized water that provides the necessary biological shielding and removes residual heat from the assembly. A sloping channel connects the spent fuel storage pool to the reactor pool, to allow the spent fuel assemblies to be transferred. The operation is carried out under the reactor lid, without exposing operating personnel to radiation.

In addition to the WWR-M2 fuel types, the spent fuel storage pool also has WWR-M5 (M7) fuel types in storage.

The storage tank is made of welded aluminium alloy sheets. Heavy concrete shielding surrounds the perimeter of the storage tank. The storage pool is covered by a cast iron lid. Thickness of the lid is 500 mm. The top of the lid is situated at the level of reactor hall floor. Access to the storage pool is made through two openings in the lid, with the diameter 640 mm, as shown in Fig. 1. One of the lids is kept shut because there is some scientific equipment arranged over it.

TABLE I. BASIC INFORMATION ON SPENT FUEL FROM THE WWR-M RESEARCH REACTOR

Assembly Type	WWR-M2	WWR-M5 (M7)
Total length, mm	750	750
Fuel core length, mm	500	500
Tubes quantity	3	5
Tube thickness, mm	2.5	1.25
Cladding thickness, mm	0.76	0.36
Fuel composition	UO_2+Al	U+Al (UO_2+Al)
Fuel core thickness, mm	0.98	0.53
Average burnup, %	50	40 (30)
Original enrichment, %	36	90
Total mass, kg		
Single assembly	0.9	0.8
Triple assembly	2.6	2.4
^{235}U mass, kg		
Single assembly	37	64 (68)
Triple assembly	112	192

Figure 4 shows an internal view of the spent fuel storage pool. There are two floors in the spent fuel storage pool. At the first floor the triple assemblies are arranged in a stationary square grid and in one removable rack (section) with hexagonal grid, as shown in Fig. 5. Pitch of the stationary square grid is 110 mm.

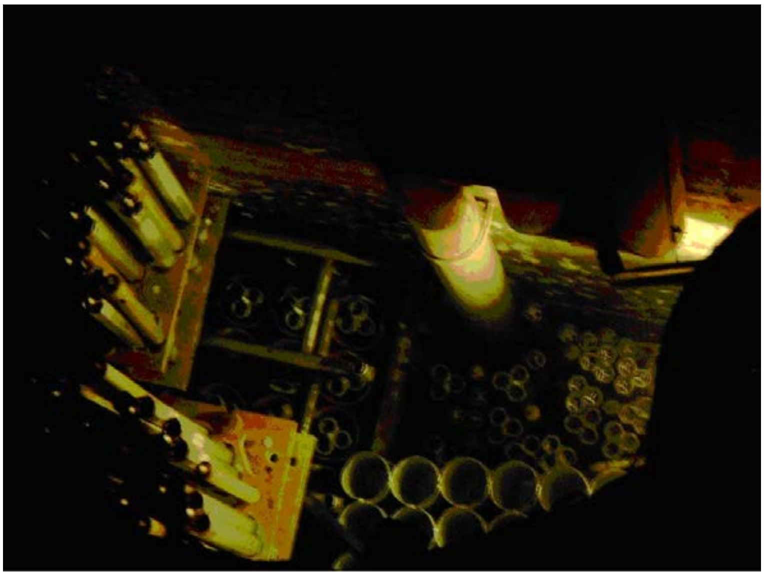

FIG. 4. Internal view of spent fuel storage pool.

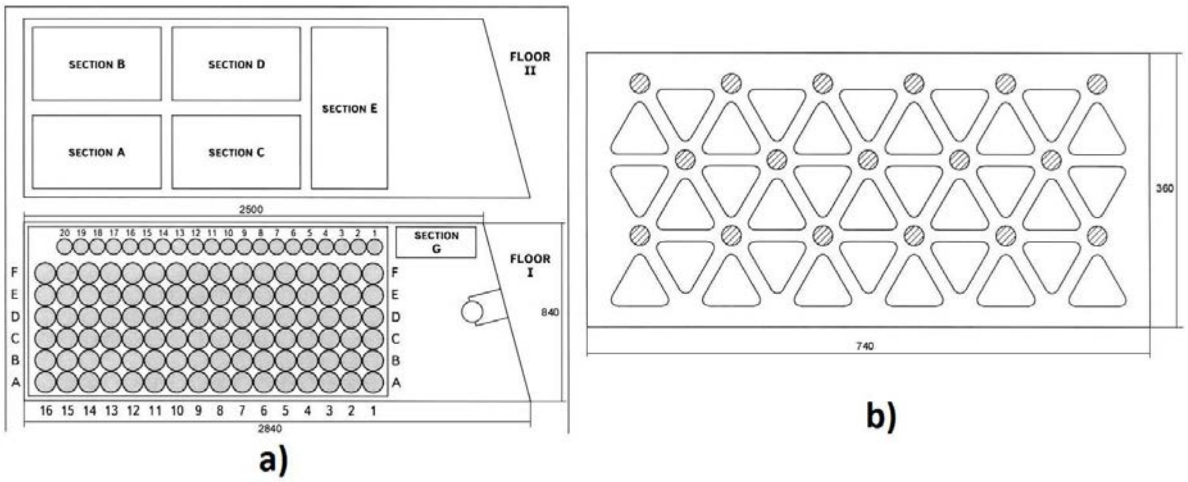

FIG. 5(a) Spent fuel storage scheme; (b) cell arrangement.

Removable racks (sections), with hexagonal grid for assemblies' arrangement, as shown in Fig. 5(b), are mounted on the second floor. There are special cells in each section for installation of boron carbide absorbers. The absorbers are installed so that, each absorber has a negative reactivity to safely compensate about 6 WWR-M2 single type fuel assemblies (original enrichment 36%), or 3 WWR-M5 single type fuel assemblies. In total 8 absorbers are distributed between 47 WWR-M2 fuel assemblies and 1 absorber is surrounded by 3 WWR-M5 fuel assemblies (original enrichment 90%).

The second floor has 5 sections, with 33 cells each, for triple assemblies. In total, considering first and second floors, the storage tank has 292 cells available for triple assemblies placing and 29 cells available for single assemblies. Into each cell designed for triple type assemblies, it is possible to place three single type assemblies. Until recently the spent fuel storage pool was full, and if no action was taken, for example returning the spent fuel to the country where it was originally enriched,, the reactor operation would no longer be possible.

The average burnup of spent fuel assemblies is 50%, and Table II shows the parameters that are monitored in the spent fuel storage pool.

TABLE II. MONITORED PARAMETERS IN SPENT FUEL STORAGE POOL

water temperature (20–25 C)
water level (maximum 380 cm)
water chemistry (as shown in Table III)
water activity (as Shown in Table III)
Facts that could demonstrate appearance of chain reaction

Measurements of gamma-radiation dose rate for the assemblies were not included in the reactor normal operational monitoring programme. Recently it was decided to make a specific measurement of the dose rate at a distance of 10 cm from one triple type fuel assembly, which was unloaded from the core in 1993. The measured value was 24 Gr/h.

TABLE III. STORAGE POOL WATER CHEMISTRY AND ACTIVITY PARAMETERS MONITORING

Monitored parameter	*Limit or average meaning*	*Frequency*
Cl-ion (µg/kg)	Not more than 10	Weekly
Hardness (µg-eq/kg)	Not more than 0.5	Weekly
Corrosion products (µg/kg)	Not more than	Weekly
Al	20	
Fe	20	
pH	5.5–6.5	Daily
Activity of (Ci/l)	Average	Weekly
^{60}Co	$7.1*10^{-10}$	
^{137}Cs	$3.6*10^{-8}$	

The ^{137}Cs measured activity has been very stable, which means that has been no release of fission products from the fuel assemblies into the spent fuel pool water. Two days a week the pool water is filtered using movable ion-exchange filters.

The material of the spent fuel storage pool is aluminium alloy SAV-1, and during the many years of operation of the reactor, it was only possible to make visual inspections of parts of the storage tank. The results of the inspections were positive, demonstrating its very good condition, as shown in Fig. 6. Measurements of the corrosive stains were also performed. The depth of the corrosion is not more than 2 mm.

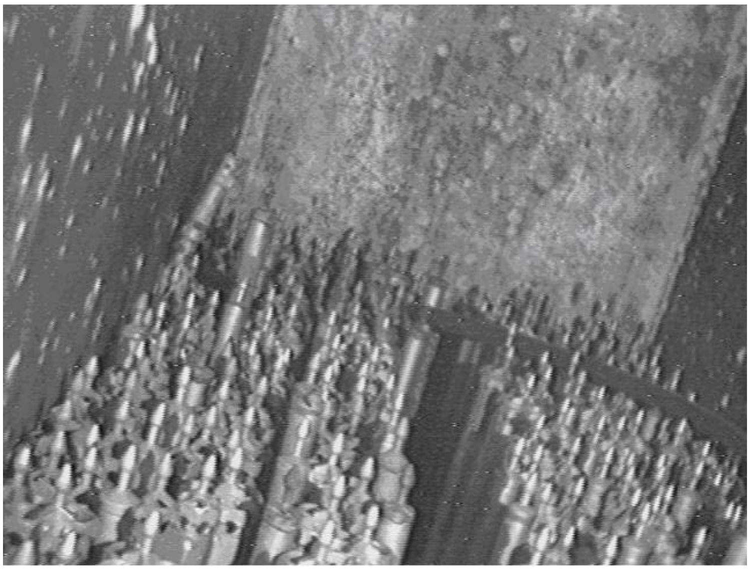

FIG. 6 Picture showing the wall of the spent fuel storage pool (top of fuel assemblies on lower part).

4. REACTOR FACILITIES

To handle loads within the reactor hall the facility has a 10 tonne capacity overhead crane. This capacity is enough to lift a fuel cask that is available to transfer the spent fuel. Four crane control panels are available in the reactor hall. One of them is situated in the crane cabin and is shielded.

The trolley on the overhead crane bridge is provided with an additional 150 Kg capacity winch. This winch has its own control panel on the reactor bridge.

The reactor hall is adjacent to a special compartment for car entrance. This compartment is equipped with another crane, whose capacity is 15 tones. A vehicle on a rail track allows the transference of heavy loads from the reactor hall to this compartment.

5. FUEL HISTORY

The fuel assemblies stay in the core during an average period of two years. After this period, the spent fuel assemblies are unloaded from the reactor core and loaded into the spent fuel storage pool. At present time, the storage pool has spent fuel assemblies that have been unloaded from the core between 1964 and 2007. The oldest assemblies have been stored in the pool for 45 years. Considering that the minimum decay time, required before shipment, is three years, at the present time there are 175 triple and 103 single assemblies stored in the pool that meet this requirement, i.e. they have been removed from the core more than three years ago.

5.1. Temporary storage of spent fuel in transportation casks

Several shipment operations were made in the past, when Ukraine was part of the former Soviet Union. During that time, the spent fuel was returned directly to the Russian recycling plant. The last shipment of spent fuel was carried out in 1988. Since then the recycling plant sent to Kiev two specialized carriages with eight casks on each one. These casks were brought one by one into the reactor hall using the rail track. Since the reactor has none of its own casks that could be used for spent fuel storage, the casks received from the recycling plant were properly identified and then loaded with spent fuel. The mass of each cask is five tonnes.

6. PRESENT SITUATION

In recent years, a decision was made to resume shipping operations of spent fuel to the Russian Federation, but now using Skoda transport casks, instead of the original Russian casks. Given the limitations of the original spent fuel storage pool, and in order to meet the requirements for utilization of the Skoda casks, a new spent fuel storage facility had to be built [1], and all spent fuels transferred into it. The new spent fuel storage facility is shown in Fig. 7.

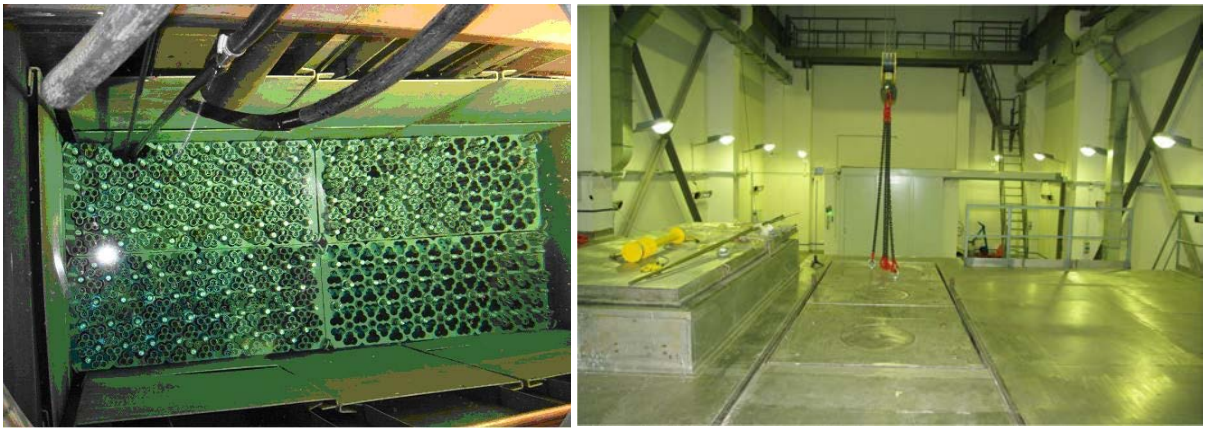

FIG. 7. Inside (left) and outside (right) view of new spent fuel storage facility.

Once the spent fuels were transferred to the new storage pool, it became possible to make a detailed inspection of the old spent fuel storage tank. The pool was cleaned, checked for signs of corrosion, and then filled with water, to continue being used as main storage. Figure 8 shows the inside view of the old spent fuel storage tank.

FIG. 8. Inside view of the old spent fuel storage pool.

Regarding the new spent fuel storage pool, its main purpose is to allow the necessary activities to load the Skoda transport containers. According to our present plans, we expect to resume the shipment operations, of the spent fuel to the Russian Federation by the middle of 2010, and the shipment of the new fuel is planned to start in 2012.

In addition to the efforts to resume the shipping operations of spent fuel to the Russian Federation, the WWR-M reactor is now under a fuel conversion programme to limit its fuel enrichment to 19.75%. It is estimated that the fuel conversion programme, combined with a permanent upgrading programme [3], will allow the reactor to be safely operated for the next decade.

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EXPERIENCE ON SPENT NUCLEAR FUEL HANDLING AND STORAGE AT RESEARCH REACTORS OF TROMBAY

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Abstract

Research reactors Cirus and Dhruva have been in operation at Bhabha Atomic Research Centre, Trombay, Mumbai since the years 1960 and 1985 respectively. Experience gained over the decades in handling and storage of spent fuel of Cirus was utilized for making several design improvements in Dhruva. Well established procedures are followed for all under water operations and equipment maintenance jobs to ensure radiation exposure control and safety. Details of some of the important spent fuel storage and handling experiences in Cirus and Dhruva research reactors are presented in this paper.

1. INTRODUCTION

CIRUS, a 40 MWt research reactor located at the Bhabha Atomic Research Centre (BARC), Mumbai, is in operation since 1960. The reactor uses natural uranium as fuel, heavy water as moderator and de-mineralized light water as coolant. DHRUVA is a 100 MWt research reactor located at BARC and is in operation since 1985. Dhruva uses natural uranium as fuel and heavy water as coolant, moderator and reflector.

The fuel elements in Cirus are in the form of single solid metallic uranium rod with aluminium cladding. Dimensions are about 3 meters in length and 3.5 cm in diameter. Top and bottom extensions are attached to the fuel elements placed in aluminium flow-tubes, to provide cooling water inlet and outlet passages as well as shielding in the vertical direction. The total length of a fuel assembly is about 10 meters and the fuel assemblies are loaded vertically in the reactor. The fuel assemblies used in Dhruva are in the form of 7-pin clusters of metallic uranium with one central pin and six pins placed around the central pin. These fuel pins have aluminium clad and are about 3 meters long and 1.25 cm diameter. The cluster is placed inside an aluminium flow-tube with a top section attached to provide cooling water outlet passage and shielding in the vertical direction.

2. SPENT FUEL HANDLING AND STORAGE SCHEME

In Cirus, the fuel assemblies are removed from the reactor with the help of a fuelling machine and are either stored in a water filled storage pool (Wet Storage Block – WSB) located inside the reactor building or transferred to the Spent Fuel Storage Building (SFSB) through a water filled canal using an underwater fuel transfer buggy. After receiving the fuel assembly in the vertical position, the buggy during its travel from reactor building to SFSB changes its position from vertical to horizontal. The fuel assembly is transferred from the buggy to a cutting station where it is trisected using underwater cutting equipment. The central section containing fuel is stored underwater and the top and bottom sections are removed for retrieval of reusable components after decontamination.

Handling scheme for irradiated fuel in Dhruva is essentially similar to that in Cirus except that the fuelling machine caters to the requirement of removing irradiated fuel from a heavy water cooled channel in the reactor and is therefore provided with a heavy water coolant circuit of its own. Irradiated fuel from the machine is discharged either into a water filled storage pool inside the reactor building or into a buggy for transfer to the SFSB. Before the fuel is lowered from the machine into the light water filled storage pool or SFSB, the heavy water coolant around the fuel in the machine is drained off and the fuel remains dry for a short time. It is ensured that the dry time of the fuel assembly is kept within prescribed limits. Provisions also exist for dousing the fuel with light water in case the dry time approaches the limit due to delay caused by unforeseen reasons. In Dhruva, the fuel assembly is handled vertically in the SFSB. After transportation to SFSB, the assembly is bisected to disconnect the top section, comprising of re-usable components like seal and shield sub-assemblies; from the fuel portion.

3. UNDER WATER STORAGE FACILITIES IN CIRUS

3.1. Wet Storage Block (WSB)

Wet Storage Block is a concrete tank lined with epoxy painted carbon steel plates and filled with demineralized water. Adequate shielding of high density concrete and lead has been provided all around the tank. The water in the storage block is re-circulated through a shell & tube type heat exchanger to remove the decay heat. Considering the long term integrity of the concrete structure of the pools, the temperature of water is maintained below 45 ^{0}C. Water level of WSB is continuously monitored and an alarm on low level has been provided to alert the operator in case of large leak. The chemistry of water is maintained by polishing it through a disposable mixed bed resin cartridge. Specific activity of the water is monitored to detect clad failure of any of the stored spent fuel assemblies. Air exhaust ducts are provided above the water level to ventilate the facility. Spent fuel assemblies removed from the reactor are stored in WSB for about one year before they are transferred to interim storage or processing locations.

3.2. Spent Fuel Storage Bay (SFSB)

The SFSB consists of underground storage bays having concrete walls of thickness of about 1.2 metres. Being an old design, no lining is provided on the walls and floors of the bays. The bays are filled with demineralized water and divided into five zones known as carriage bay, cutting bay, inspection bay, loading bay and cut rod storage bay. The depth of water varies from 4 m to 5.5 m in different bays. Isolation doors with inflatable seals have been provided to facilitate isolation of a bay for cleaning or any other purpose. A canal is provided to facilitate transfer fuel rods from WSB to SFSB using a motorized buggy moving on rails. The integrity of the bay is ensured through a very well established surveillance procedure.

The water is re-circulated to maintain proper chemistry and clarity. Water is skimmed from top of the bays through adjustable weirs and collected in a tank installed at the pump suction. The water is pumped back to the bays through filters and mixed bed ion exchange resin columns. Since the evaporative cooling (large surface of water open to ventilation air) is enough to dissipate the very small, almost insignificant heat load, no arrangement for forced cooling has been made. Underwater lights provide the required illumination for carrying out various operations like cutting of assembly, canning, crimping, transfer, storage, loading, etc.

The SFSB ventilation system maintains about four air changes per hour in the building and a slightly lower pressure as compared to atmospheric pressure. The air enters the SFSB through pre-filters. SFSB exhaust fans sucks air from the storage bays through duct openings provided just above the normal water level. The exhaust air, from SFSB through HEPA filters, joins the main reactor building ventilation exhaust duct and gets discharged through stack. The SFSB exhaust fans trip automatically on sensing failure of reactor building ventilation system, to prevent blow back of active air. The air activity in SFSB is monitored at regular intervals.

4. SPENT FUEL STORAGE EXPERIENCE IN CIRUS

During the initial days of Cirus, the middle cut section of irradiated fuel assemblies, comprising of fuel element, were being stored vertically in underwater storage racks of SFSB, with the bottom end of the elements resting at the floor. When stored in this fashion for prolonged duration, some of the fuel elements experienced bowing due to self weight. Based on this experience, the storage methodology was changed. The spent fuel from the reactor is now stored in WSB in the fully assembled condition. In this arrangement, the fuel elements remain suspended from the top shielding section and thus the problem of bowing due to self weight is solved. The assemblies are transferred to SFSB only when required for further processing thus avoiding the need for long term storage in SFSB.

The fuel elements which suffer clad failure during service are put in aluminium cans after removal from the reactor and the cans are then plugged. Such cans used to get internally pressurized due to uranium water reaction. A suitable pressure relieving arrangement incorporated in the plug design overcomes this problem.

Recirculation of storage pool water through filters and ion-exchange resin columns is maintained to achieve the required water clarity and optimum chemistry, and also to keep the bay water radioactivity level to a minimum. Earlier the ion-exchange resin columns of SFSB water purification system were being periodically regenerated on exhaustion. To obviate the problems associated with regeneration of the resins viz. disposal of radioactive liquid waste generated, maintenance of associated equipment and significant radiation exposure to operations and maintenance staff; cartridge type ion-exchange units are now used for bay water purification. This methodology is feasible on account of low ionic impurity load in the bay water. The resin cartridges are located inside the bays so that the radiation field in the working areas doesn't increase. On exhaustion, the cartridges are disposed of as solid waste. The connectors for the flexible hoses of these cartridges are of cam-lock type which can be disconnected remotely when required after exhaustion of the cartridge, thereby minimizing the radiation exposure during the job.

Maintenance of underwater equipment is another area which calls for large efforts. This is so, especially in view of the long years of service seen by these equipment in Cirus, giving rise to ageing related degradations. For example, there was an incident wherein the wire ropes constituting the drive mechanism for the fuel transfer buggy got disconnected from the buggy due to failure of a socket joint on account of corrosion. The resulting slackness of the rope led to its displacement from the underwater guide pulleys. For repairs, the section of the bay housing these pulleys had to be dewatered after isolation from other bay sections using gates with inflatable seals. This required significant effort towards preparatory work as also for the actual repairs due to difficult site conditions. This opportunity was utilized also for carrying out visual inspection of the concrete structure of the pool floor and side walls. The condition was found to be very good.

The layout of recirculation line from WSB to recirculation pump is such that a breach in the line can lead to loss of water from the pool due to siphoning. To overcome this deficiency, the relevant portion of the line was encased in a metallic box and concrete was poured into the box for embedding the line to guard against any breach. During an outage of Cirus for refurbishment, the lining of WSB was inspected visually and confirmed to be free of defects using a vacuum box technique. Minor superficial cracks noticed at a few locations were repaired.

Originally portable underwater lights (150 W, 24 V DC) were being used to carry out various operations like cutting of fuel assembly, canning, crimping, transfer, storage, loading, etc. The lights were moved and focused manually. Based on experience, fixed underwater lights (400 W, 230 V AC) with adequate electrical switchgear for protection have been installed at designated locations. This has helped in reducing the effort on maintenance of lights and man-Rem consumption.

5. UNDER WATER STORAGE FACILITIES IN DHRUVA

5.1. Wet Storage Block (WSB)

In Dhruva, the wet storage block is a storage pit made of high density concrete of thickness such that the radiation field outside the block is 0.2 mR/hr. The facility consists of a tank lined with SS 304L plates on the sides and the bottom to prevent leakage and for ease of decontamination. The pit is covered with a shielding block with lattice sleeves for inserting the fuel assemblies. The facility provides a temporary storage of spent fuel in de-mineralized (DM) water. The pit water is circulated through a plate type heat exchanger to maintain water temperature at 35°C. The water is recirculated with a pump which takes suction from top of the storage block and discharges to the bottom of the pit through an inverted U-loop to avoid accidental draining of water. A siphon break valve has been provided on top of the U-loop for this purpose. Piping has been arranged in such a manner that the facility can be drained either by gravity or by operation of a pump. WSB water is circulated through an ion exchange bed, a strainer and a micro-filter to maintain water chemistry and clarity. Water level in WSB is monitored continuously with provision for annunciation of ‘High level’ and ‘Low level’ alarms. The exposed piping and valves of the facility are made of carbon steel whereas all the embedded piping is made of stainless steel. Air exhaust ducts are provided above the water level to ventilate the facility.

5.2. Wet Storage Spent Fuel Storage Bay (SFSB)

SFSB offers various facilities for underwater vertical storage and handling of irradiated fuel assemblies, such as bisection of fuel assemblies to separate the fuel clusters from the re-usable components, like seal and shield subassemblies; underwater storage of irradiated fuel clusters; canning of clad failed fuel clusters, etc. Water filled SFSB is divided into three zones called general bay, loading bay and fuel sub-assembly storage bay. All of these bays, which are located above ground level, are made of 1.2 m thick concrete walls. Heavy density concrete is used at locations where radiation shielding is a requirement. A water filled fuel transfer trench (FTT) connects the SFSB to the reactor building, enabling the transfer of the spent fuel from the reactor building to the SFSB. The internal surface of the bays and FTT are lined with SS 304L plates to provide leak tightness and ease of decontamination. Clarity and chemistry of the de-mineralized water in the bays are kept under control using water recirculation and purification system comprising of strainer, recirculation pumps, micro filters, heat exchangers, ion exchange resin columns and associated piping.

The SFSB building is ventilated using supply and exhaust fans of appropriate flow ratings to have about six air changes per hour, keeping a slightly negative pressure within the building. The water surface of the bays is continuously swept with a stream of ventilation air to check the airborne activity due to evaporation. Air exhaust from SFSB building joins the Reactor Building ventilation exhaust stack through a bank of HEPA filters.

Online area radiation and air activity monitors are provided at both Cirus and Dhruva for radiological safety. The bays are also provided with diverse systems for making up of bay water level.

Experience gained from Cirus was utilized in the design of spent fuel storage facility of Dhruva. Some of the design improvements incorporated in Dhruva and experience in spent fuel storage are described below.

Adequate storage capacity for irradiated fuel has been provided towards avoiding the need for costly away-from-reactor storage facilities in future.

The spent fuel storage bays in Dhruva have been located above the ground level. This obviates the problem of ground water table exerting pressure on the stainless steel liner with the consequent possibility of its buckling when the bay is dewatered for repairs etc. Stress corrosion cracking of the SS liner due to chlorides in the sub-soil water is also avoided. This arrangement also facilitates the layout of bay water purification system and ventilation equipment’s without congestion.

The design of the fuel transfer buggy and its drive mechanism has also been significantly modified in Dhruva such that the drive wire ropes and guide pulleys are laid close to the water surface in the bay. For repairs to these components, the bay water level has to be lowered only by about 1 meter without the need for complete dewatering.

Improvements have been made in the design of underwater cutting equipment by use of submersible electrical motor for driving the cutting equipment. With this arrangement, there is no need for long shafts to connect the underwater cutting saws with the motors located above the bay. Consequently, failures due to vibrations have been minimized.

Removable trays have been provided below the underwater fuel storage racks for collecting any debris/uranium powder. With this the bay floors remain clean and the trays can be removed for cleaning when required.

A gantry crane has been provided over the bays, which travels over the entire length of the bays. This has facilitated underwater fuel handling to a large extent.

A pool site inspection facility is provided to carry out post-irradiation examinations at the reactor site itself which can provide vital information about the primary defects causing fuel failure. Thus, the need for transportation of irradiated fuel to hot cells can be reduced. However, generation of loose uranium particles during the preparatory operations on the irradiated fuel assembly for inspection is a cause for concern and is being addressed.

For canning of fuel elements with clad failure, the flow tube itself is used as a can. For achieving this, significant developmental work had to be done towards working out the design of plugs and the plugging tools.

The material of the hand rails around the storage pool was changed from carbon steel to stainless steel to minimize decontamination efforts.

Disposal of exhausted ion exchange resins from the water purification columns of WSB and SFSB is a repetitive job involving radiation exposure to the handling personnel. To circumvent this problem, the system is being modified with reusable type shielded cartridges, which can be disconnected from system on exhaustion and transported to the disposal site for emptying and polymer fixation of the spent resin.

6. CONCLUSION

Around five decades of experience in the field of underwater storage of irradiated fuel from research reactors exists in BARC. Based on this experience several improvements have been made in the design of spent fuel storage bays, in associated equipment and in the underwater storage and handling methodology of irradiated fuel.

Vertical storage of spent fuel with the fuel elements suspended from the top has been found to be better than storage with fuel resting at its bottom to avoid the problem of bowing. Provision of collection trays below the storage racks has been found to be helpful for improving bay floor cleanliness. For canning of fuel with clad-failure, the flow-tubes themselves are used as cans with provisions to prevent occurrence of internal pressure buildup.

Use of disposable type ion-exchanger cartridges for bay water purification was found to be beneficial in terms of reduction of radiation exposures for operating personnel during maintenance activities. Location of spent fuel handling bays above ground level has been considered superior from the point of view for the pool liner integrity.

Provision of adequate underwater storage capacity, improved design for ease of maintenance of underwater fuel transfer and handling equipment and availability of pool site inspection facilities are some additional improvements made to facilitate management and storage of irradiated fuel from research reactors in BARC.

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MANAGEMENT PROGRAMME FOR RESEARCH REACTOR SPENT NUCLEAR FUEL STORAGE AND INTERIM STORAGE FACILITIES AT NUCLEAR RESEARCH INSTITUTE ŘEŽ PLC, CZECH REPUBLIC

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Abstract

LVR-15 is a research reactor that operates in NRI Řež plc, Czech Republic since 1957. From the first criticality of the reactor until 2007, more than 600 fuel assemblies have been used in the reactor core, resulting in a large quantity of spent fuel. This paper describes the management of Research Reactor Spent Nuclear Fuel (RRSNF) storage at NRI Řež plc. Characteristics of fuel types used (EK-10, IRT-2M 80% enriched and IRT-2M 36% enriched), and characteristics of interim storage facilities and infrastructure available are described. The paper emphasizes the experience gained during the preparatory works for the SNF shipment (facility and equipment modification, cask licenses) and the preparation of the SNF for transport, in particular its checking, repackaging in a hot cell, loading into the VPVR/M casks, drying, manipulation, completion of the transport documentation, etc. including its transport to the High Level Waste Storage facility before its transportation to the Russian Federation. Future Russian Federation options for the spent fuel management at NRI Řež plc, including possibility of additional shipments to the Russian Federation, or interim storage in SKODA VPVR/M cask systems at NRI Řež is also discussed.

1. INTRODUCTION

In the Nuclear Research Institute Řež plc (NRI) of the Czech Republic, there are two research reactors, LR-O, a 5 kW critical assembly that reached criticality for the first time in December 1982, and LVR-15, a 10 MW beryllium reflected, light water moderated tank type reactor, which has been used for radioisotope production, silicon doping, material testing, neutron activation analysis, boron neutron capture therapy and basic research.

LR-O utilizes fuels designed for the WWER type power reactor, with 5% enrichment, and because of the reactor low power, spent fuel is not an important issue for this reactor. However, for LVR-15, a 10 MW reactor the situation is different. The reactor is a WWR Russian type model, which reached criticality for the first time in September of 1957, and since its first operation, more than 600 fuel elements were used in the reactor core. All fuel elements used in LVR-15 reactor, from the start of the reactor operation until 2007 were stored in storage facilities located in the NRI, most of it waiting for the conclusion of negotiations toward their transport to the Russian Federation, after a quadripartite agreement signed in 2004, when the NRI Řež plc jointed the Russian Research Reactor Fuel Return (RRRFR), programme under the US–Russian Global Threat Reduction Initiative (GTRI) initiative [1].

2. SPENT FUEL CHARACTERISTICS

Since the reactor started operation in 1957 three types of fuel were used. From 1957–1974 the reactor was fuelled with EK-10 Russian type fuel elements, enriched to 10%. EK-10, shown in Figure 1, is an assembly consisting of 16 rods of uranium dioxide-magnesium alloy in aluminium casing.

In 1974, after an upgrading process of the reactor, a new fuel element type, 80% enriched IRT-2M, was introduced in the reactor core. IRT-2M, shown in Figure 2, consisted of four or three square tubes of uranium/aluminium alloy fuel metal, covered on both sides with aluminium. This fuel was used to fuel the reactor until December 1998.

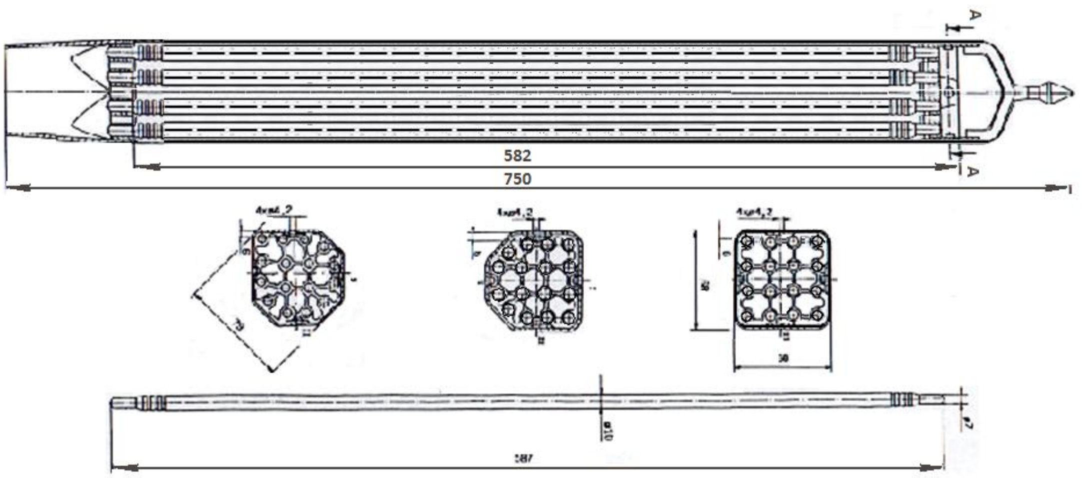

FIG. 1. EK 10 Fuel Assembly.

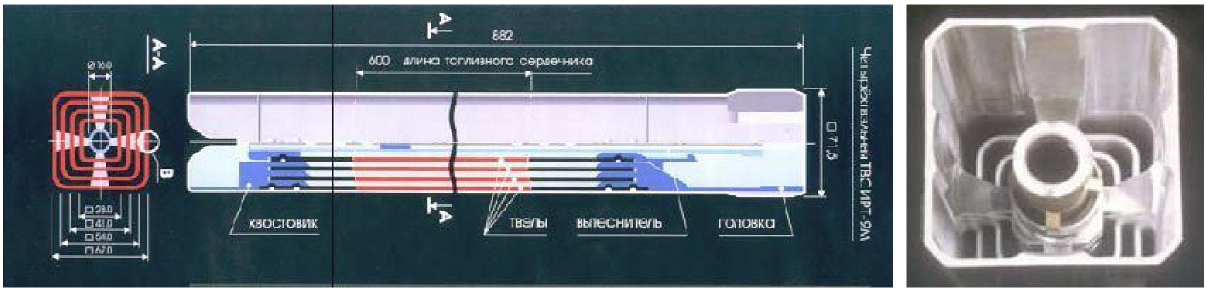

FIG. 2. The IRT-2M Fuel Assembly, showing details of the square tubes on the right.

Conversion of the reactor, to use similar IRT-2M fuel with enrichment of 36% started in 1995. In this year the first 36% enriched IRT-2M type fuel elements,, were introduced into the reactor core. The conversion process continued by replacing the used 80% with fresh 36% enriched fuel, and since January 1999 only 36% enriched fuel has been used in the core of LVR-15 research reactor. Instead of using a uranium/aluminium alloy, like the IRT-2M 80% enriched, the fuel material in the IRT-2M 36% enriched is UO_2-Al dispersion.

3. FUEL INVENTORY AT NRI

Up to March of 2007, a total of 572 fuel elements were used in the core of LVR-15 research reactor, being two hundred twenty two (222) 10% enriched EK-10; two hundred fifty two (252) 80% enriched IRT-2M and ninety eight (98) 36% enriched IRT-2M. By that time thirty two (32) IRT-2M 36% enriched fuels were in the reactor core. Table I shows the location of the total fuel inventory of LVR-15 research reactor in March of 2007.

4. SPENT FUEL STORAGE FACILITIES

In the NRI the spent nuclear fuel (SNF) can be stored in the at-reactor pond (AR pond), away from the reactor, in a building annex to the reactor (AFR1) and away from the reactor, in a facility called high level waste storage (HLWS) facility.

The at-reactor pond (AR pond) is located in the reactor hall and serves as an interim spent fuel storage as well for handling fuel from reactor core during the reactor outage or under off-load refuelling. The AR pond has an aluminium rack with capacity to accommodate 60 fuel assemblies, but 30 are always left empty for emergency fuel movements from the reactor core. Figure 3 shows a top view of the AR pond.

TABLE I: TOTAL INVENTORY OF LVR-15 RESEARCH REACTOR FUEL, PER LOCATION OF NRI REZ IN MARCH 2007

Location	Type of fuel assemblies	Number of assemblies
Fresh Fuel Storage	IRT-2M, 36%	94
Active core of the LVR-15 reactor	IRT-2M, 36%	32
AR pond	IRT-2M, 36%	28
Reactor annex pond	IRT-2M, 36%	70
HLWS pond	IRT-2M, 80%	252
HLWS pond	EK10	16
HLWS dry box in drums	EK10	206

The AR pond was inspected in 1996 and in 2007, when corrosion of the liner was detected, in the form of corrosion pits with depth between 3 and 4 mm. This pitting depth is not a problem and the next inspection is planned for 2013.

FIG. 3. General view of AR Pond.

The away from reactor annex pond (AFR1), located on the reactor site, near the reactor hall, is made of stainless steel, and has a total capacity for 100 fuel assemblies. Figure 4 shows some pictures of the AFR1 pond after it was empty. The pond was inspected in 2000 and in 2008, when minor discoloration on welding places was detected. The next inspection of AFR1 is planned for 2015. Table II shows the main parameters for AR and AFR1 ponds.

FIG. 4. General view into AFR1 Reactor Annex Pond.

TABLE II. CHARACTERISTICS OF AR AND AFR1 PONDS

Pond	AR	AFR1
Liner thickness and material	13 mm Al alloy	Stainless steel
Width	0.84 m	1.20 m
Length	2.84 m	2.30 m
Depth	4.00 m	6.00 m
Capacity	60 spent fuel elements [a]	100 spent fuel elements
Rack material	Al alloy	Cd, Al sandwich (on 2 of 4 walls)
Grid distance	150 mm	130 mm

[a] 30 positions permanently left empty for emergency fuel movements from the core.

The High Level Waste Storage (HLWS) facility, built in the upper part of NRI, is a building external to the reactor building. The construction of HLWS started in 1981, and after some delays it was put in operating in 1995. In the lower part it is a reinforced concrete monolith to the level of +5.65 meter, and in the upper part it is a prefabricated hall, originally with 12 square boxes measuring 6 × 6 m^2 each, as shown in Figure 5. It has been used to store EK-10 and IRT-2M type spent fuels, conditioned radioactive waste and nonstandard radioactive waste. Eight of the square boxes are kept dry, and serve for radioactive waste storage, two contain spent fuel storage pools, one contains water technology and one is used as entrance area for trucks. Each one of the two spent fuel storage pools can accommodate 300 spent fuel assemblies. Figure 6 shows the external and internal views of the HLWS facility.

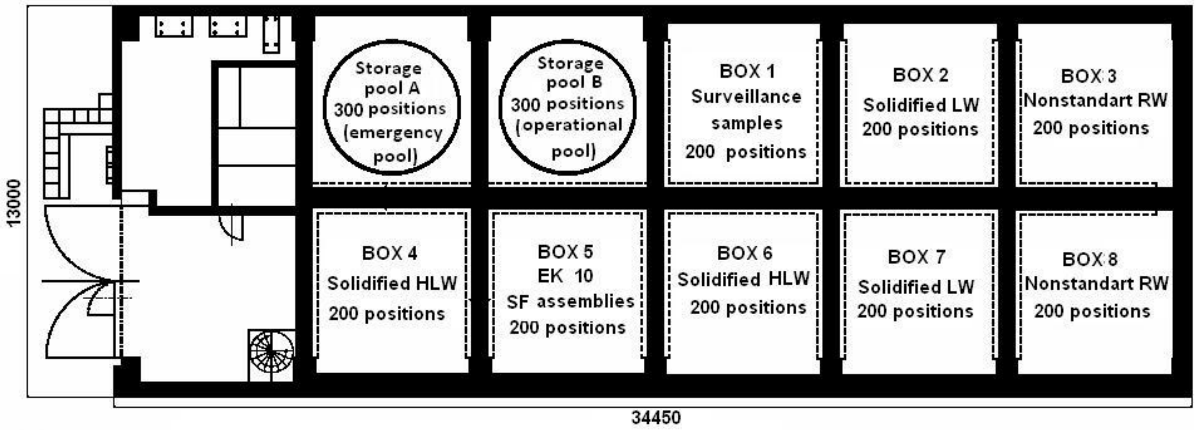

FIG. 5. Diagram of HLWS facility showing original position and purpose of each box.

FIG. 6. External (left) and internal (right) views of HLWS facility.

5. THE HISTORY OF SNF MANAGEMENT AT NRI

Up to 1969 all the EK-10 SNF was kept in the reactor site, and between the years 1969–1975, they were transferred to the Reloading Centre of NRI, a building originally designed for radioactive waste. During this period the EK-10 SNF was held in 200 litres dry storage drums filled with concrete. Between the years of 1996–7, after completion of the high level waste storage (HLWS) facility in 1995, all EK-10 SNF was transferred from the Reloading Centre to HLWS facility, and kept in the original 200 litres dry storage drums, as shown in Fig. 7. For the fuel assemblies IRT-2M, until 1996 they were kept in AR pond, for initial decay, and then transferred to AFR1. Between the years of 1996 and 2003, all existing IRT-2M spent fuel assemblies, including two leaky fuel assemblies, were transferred, one by one, to the HLWS facility, using a K-1xIRTM Škoda cask.

In 2003, a tripartite technical team, with representatives from the USA, Russian Federation and the IAEA visited NRI in Řež to discuss the possibility of returning the spent fuel to the Russian Federation. During the negotiations, special attention was given to the condition of the spent fuel. Taking into account the materials of drum, construction used to store the EK-10 fuel assemblies (carbon steel drum filled with concrete, carbon steel liner), and considering that the drums were not sealed, the possible interaction of the drum material with aluminium cladding, resulting in corrosion of the EK-10 fuel cladding had to be taken into consideration. A decision was taken to repackage all EK-10 SNF into canisters. For the repackaging process of the EK-10 type fuel assemblies, which had been placed in Pool B and Box 5 of the facility, the HLWS facility underwent a major reconstruction process, divided into two phases. Phase one involved the construction of a repackaging unit, composed by a hot cell, control room, storage installation (safe) and the handling hall for handling the drums in boxes 6, 7 and 8 of the facility.

FIG 7. Concreted drums with EK-10 SNF in the HLWS facility.

In phase two an annex was built for the storage of ŠKODA VPVR/M casks used for transport and interim storage of the research reactor spent fuel; to set up workplaces for loading the ŠKODA VPVR/M casks, for preparation of containers before transferring the high level radioactive material to the hot cell, and for management of damaged SF.

The modifications in HLWS facility also included an upgrade of the overhead crane (adding a nomadic load cell and speed controls); setting up the ŠKODA VPVR/M cask manipulation frame; adding special shielding around the cask pool support/fuel loading stand; setting up numerous specialized tools and equipment for handling the fuel and transport cask; and setting up cask drying and leak tightness testing equipment.

Figures 8 to 10 illustrate the result of the reconstruction process, and Fig. 11 shows the design of the EK-10 fuel repackaging unit [2].

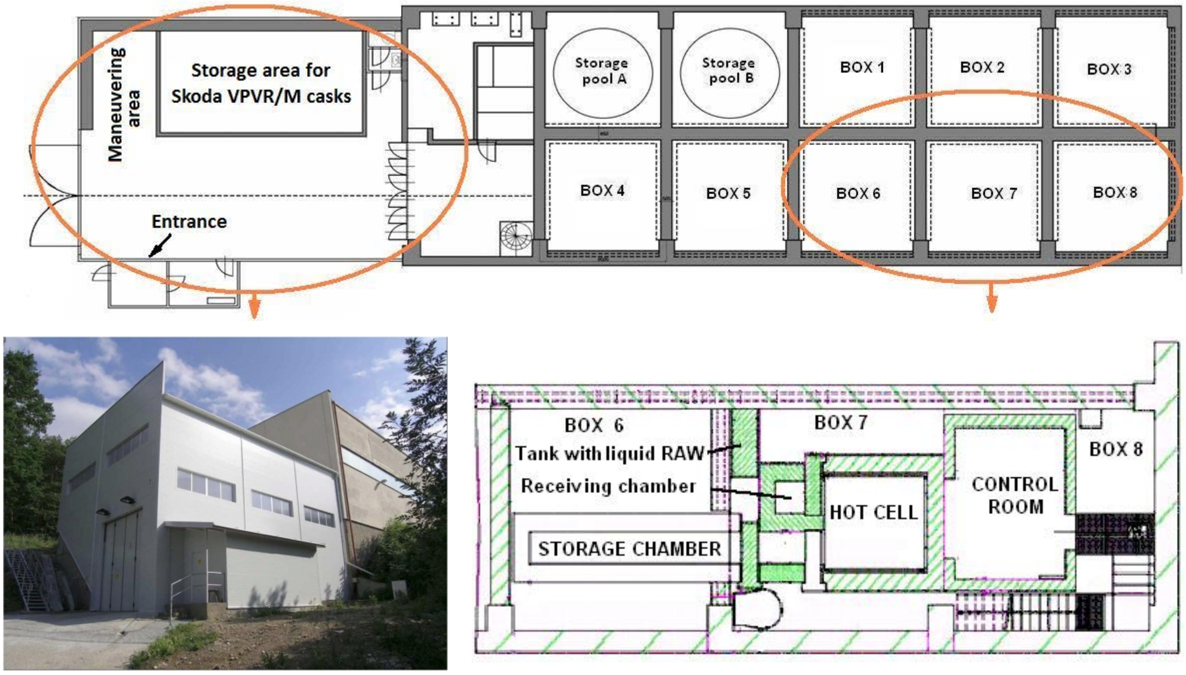

FIG. 8: Modifications made on HLWS facility.

FIG. 9: Internal view of HLWS facility before modification (left) and details of the handling hall building (right).

FIG. 10. External view of the handling hall (left) and its interior (right).

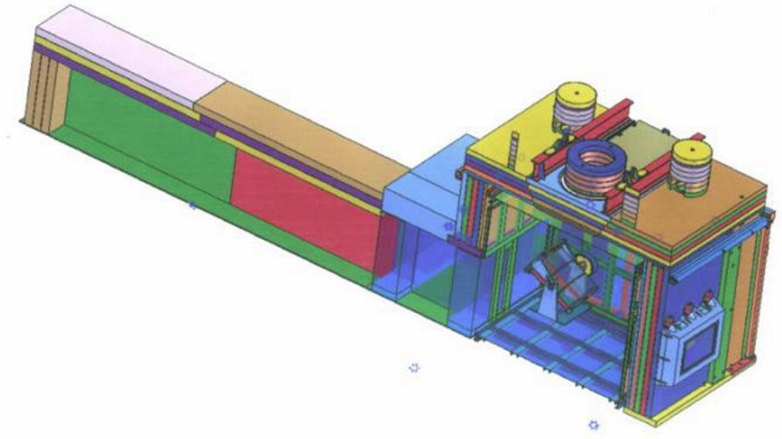

FIG.11. Design of the EK – 10 Fuel repackaging unit.

6. REPACKAGING EK-10 AND NON-SEALED IRT-2M IN EK-10 HOT CELL

When the new repackaging unit was ready in the HLWS facility, all EK-10 SNF were repackaged into stainless steel canisters, hermetically sealed (welded), put into a cask basket and then stored in a special storage facility located close to the hot cell. The EK-10 SNF repackaging process started in 2006 and was completed in 2007. In addition to repackaging all EK-10 SNF, some leaking IRT-2M FAs were also repackaged in the hot cell.

6.1. Repackaging EK-10 fuel assemblies stored in concrete drums

As shown in Figure 5, Box No. 5 of the HLWS facility had been selected to store the concrete casks containing the EK-10 SNF assemblies. The casks were, stored under dry conditions on three levels. Most of the fuel assemblies were complete, but some had been disassembled into individual rods. A steel-frame handling hall with transparent walls was built above boxes Nos 5, 6 and 7. A 2-tonne crane was installed inside, as well as several pieces of special equipment for the lifting, preparation, and transport of casks to and from the EK-10 hot cell. Entrance into the hall was via a construction module, modified as a “hygienic loop” with an emergency shower and a chemical toilet.

After removing Box No. 5 shield panels, the concrete casks in Box No. 5 were consecutively lifted by a crane with a ZMAN device and transferred to Box 6 (i.e. the handling hall). The cask was placed under a semiautomatic pin splitter, where the cask suspension trunnion pins were cut with a band saw. After the removal of the suspension, the cask was lifted again by the ZMAN device, transferred to the base slab of the tilting unity, and loaded together with it into the hot cell access port. The external opening port was then closed and the internal one opened. The loading mechanism pulled the cask into the hot cell, tilted it, and extracted the fuel assembly. After checking that it had been completely emptied, the cask was returned to the access port, from which it was later extracted and transferred to the handling hall by the ZMAN device, set aside, and checked with a dosimeter. The fuel assembly in the hot cell was placed into a stainless-steel canister, closed with a lid, and welded shut with semiautomatic equipment. The canister was exhausted with a vacuum pump and filled with helium to an absolute pressure of 0.12±0.07 Mpa.

After being filled with helium, a lid was placed over the cartridge’s fill valve and welded shut. The cartridge was then tested for leaks. The leak test was performed with a helium leak tester located in the hot cell control room. It was connected with a vacuum line to a test vacuum vessel in the hot cell. The welded shut canister was placed into this vessel, and after the vessel was evacuated, helium leak detection took place. The test normally took 8 to 12 minutes, until the desired air tightness criterion was reached (i.e. 1E-08Pa*m3/s). Once the test was completed, the loading mechanism pulled the storage basket from the access port in front of the vault into the hot cell, and the canister was placed inside and its position noted in the basket's cartogram. The storage basket was then returned to the access port, where it was closed. After all 36 cells of the basket were full, it was transferred into the vault, and another wagon with an empty storage basket was sent from the handling hall into the access port.

Figures 12 to 14 show the sequence followed in the repackaging process.

FIG. 12. Handling the concrete drum.

FIG. 13. Unloading the EK-10 fuel assembly from the concrete drum.

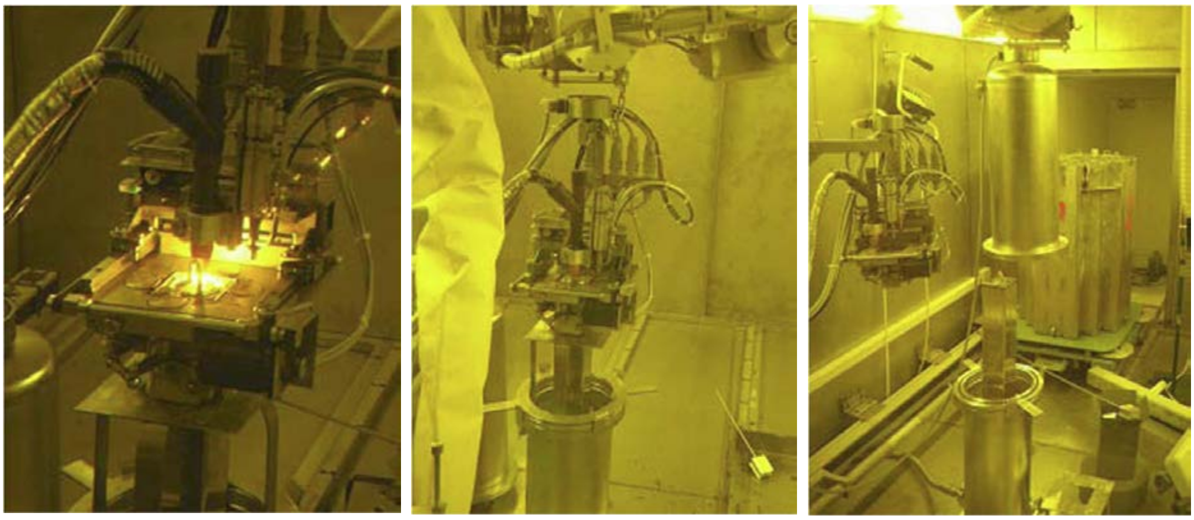

FIG. 14. Repackaging a fuel assembly into a SS canister in the hot cell of HLWS facility.

Following the fuel assembly storage records from Box No. 5, the repackaging process started with entire fuel assemblies, being followed by atypical containers with individual rods. In the hot cell, the rods were placed into the canisters one at a time (normally 36 rods were placed in the same canister). At the very end of the repackaging process, a double barrel (a double-height concrete cask) remained. The cartridge found inside was transferred into the hot cell directly via the ceiling opening, where it was cut apart and the rods from two fuel assemblies were placed into the two last canisters.

During the repackaging process of concrete casks from the first level of Box No. 5, the separation of cask suspensions proved to be a critical operation from the standpoint of reliability and staff radiation dose levels. An alternative solution was quickly tested and put into operation (i.e. separation of the handles using two remotely controlled plasma burners, mounted on a simple mechanical fixture directly in Box No. 5). Casks in the remaining two levels were then prepared for placement into the hot cell in this manner. Towards the end, for several atypical casks with rods (of smaller diameter), the burning apparatus was modified for their dimensions. When free space in the handling hall was completely filled with emptied casks, repackaging work was interrupted, the ceiling of the handling hall was opened, and the large warehouse crane transferred the empty barrels out of the storage facility, where after a dosimeter check, they were designated for disposal. Then the repackaging process was resumed [2].

6.2. Repackaging EK-10 fuel assemblies stored in pool B of the HLWS facility

After repackaging the EK-10 fuel assemblies from the concrete drums, some modifications were made in the HLWS facility. To allow handling and placement of the cask in the hot cell, it was necessary to make the hot cell accessible from above. The entire handling hall was disassembled, including the hygiene loop, and the shield plug was removed from the opening in the centre of the ceiling. A special shielding and transfer adaptor for the cask was installed (hot cell entry apparatus). Additional modifications included: closure of Box No. 5 containing a double barrel; disassembling handling hall; disconnection of the hygiene loop container from existing distribution; removal of the hygiene loop container from the radioactive waste storage facility; making the handling opening in the ceiling of the hot cell accessible.

To repackage the EK-10 fuel assemblies stored in pool B of the HLWS facility, after removal of the cover panels of Pool B, the cask and transport vehicle framework were relocated into the hall, and the framework with the cask was placed at a designated location. The cask was lifted into a vertical position and removed from the framework. The lower crush zone and lower secondary lid of the cask were removed, and the tapered bucket guide was installed. The cask was placed in a horizontal position, supports were mounted, and the cask was then tilted upright and relocated to the acceptance facility of Pool B. The upper crush zone and upper secondary lid were removed, and the bucket hoist was installed.

The shield plug was removed from the hot cell ceiling opening and a special shielded adapter for the cask (hot cell entry apparatus) was installed. The bucket was hung on a sling and lowered to the bottom of the pool. The fuel assembly was inserted from the storage grille into the bucket using a handling rod. The bucket was pulled into the body of the cask. The cask was relocated onto the hot cell entry apparatus. The bucket with the fuel assembly was lowered with the winch into the hot cell and opened. The fuel assembly was taken out of the bucket. Immediately after removal of the fuel assembly, the bucket was pulled back into the cask. While the fuel assembly was in the hot cell, the cask was left on the ceiling of the hot cell. The fuel assembly was dried out, the bottom and upper portion of the fuel assembly were removed, and the dried out and shortened fuel assembly was placed in a stainless steel canister, which was welded shut. After a leak test, the fuel assembly was placed in a Škoda VPVR/M cask basket and into the hot cell vault. The cask, with the bucket inside, was relocated from the hot cell entry apparatus above the pool for another fuel assembly. After handling of the K-1xIRTM cask was completed, a surface contamination check was performed, and, if needed, decontamination of the special shielded adapter for the cask, which was subsequently removed from the access opening of the hot cell, was performed. The shield plug was installed in the hot cell ceiling opening and pool B was covered by the shielding panels [2].

6.3. Repackaging IRT-2M fuel assemblies

Once all EK-10 SNF were hermetically sealed into the canisters (including those stored in wet conditions), damaged IRT-2M fuel assemblies stored in the LVR-15 pool and damaged IRT-2M fuel assemblies in the HLWS facility were also repackaged.

To repackage the IRT-2M fuel assemblies, additional modifications were needed, like installing specially designed cask basket support in the AFR1 facility; modifying the AFR pool cask transport device; and modifying the original ŠKODA K-1 × IRTM transfer cask, needed to transfer some damaged SNF assemblies from the reactor building to the HLWS facility. To transfer the damaged IRT-2M fuel assemblies from the AFR1 pond in the reactor building to the HLWS facility, the fuel assembly was loaded from the pond into a K-1xIRTM cask, which was then transported by truck to the HLWS facility. The cask, along with the framework from the transport vehicle, was relocated into the HLWS facility hall. The framework, with the cask, was placed at a designated location; the cask lifted into a vertical position and removed from the framework. The upper impact limiter and upper lid of the cask were removed, and the tapered bucket guide was installed. The cask was placed in a horizontal position, supports were mounted, and the cask was then tilted upright and relocated to the acceptance facility of Pool B. The lower impact limiter and lower lid were removed. The cask was relocated onto the hot cell entry apparatus. The bucket with the fuel assembly was lowered with the winch into the hot cell and opened, and the fuel assembly was taken out of the bucket. Immediately after removal of the fuel assembly, the bucket was pulled back into the cask. While the fuel assembly was in the hot cell, the cask was left on the ceiling of the hot cell. Figure 15 illustrates details of the transfer operation.

FIG. 15. Transferring IRT-2M spent fuel from storage pool B (e) to the repackaging hot cell; (f) using the Škoda K-1xIRTM cask.

Once in the hot cell, the fuel assembly was dried out, the bottom and catch portion of the fuel assembly were removed, as shown in Fig. 16, and the dried out and shortened fuel assembly was placed in a stainless steel canister, which was welded shut. After a leak test, the fuel assembly was placed in a Škoda VPVR/M cask basket and into the hot cell vault. The cask, with the bucket inside, was relocated from the hot cell entry apparatus above the pool for another fuel assembly. After handling of the K-1xIRTM cask was completed, a surface contamination check was performed, and, if needed, decontamination of the special shielded adapter for the cask, which was subsequently removed from the access opening of the hot cell, was performed, then the shield plug was installed in the hot cell ceiling opening.

Cask handling for transport from HLWS facility to reactor building took place in the opposite order of the above procedure, with the only difference being that both impact limiters were transported to the reactor building separately on a vehicle. The empty cask, without impact limiters, was transported in a transport framework [2].

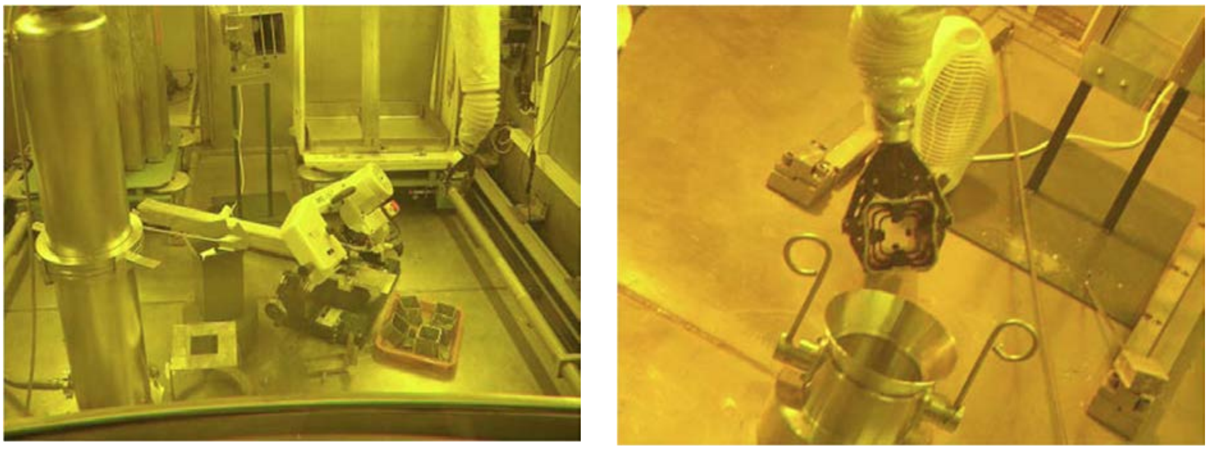

FIG. 16. Cutting off a leaking IRT-2M fuel assembly.

7. THE TRANSPORT CASK SYSTEM ŠKODA VPVR/M

Following the meeting with the tripartite technical team in 2003, in 2005 a formal contract was signed, to start the necessary activities to prepare the transport of the spent fuel to the Russian Federation. Once the contract was signed, and with the significant technical and financial support of the USA Administration and the US DOE (total of approximate CZK 450 mil.), it was decided that the Czech Republic should become a pilot country to carry out such shipment to the Russian Federation. Use was made of a specially developed and tested cask, compatible with the technology of research reactors of Russian design, and the reprocessing plant in the Russian Federation.

To comply with the agreement to transport the fuel to the Russian Federation, plans were established to repackage all EK-10 and IRT-2M leaking fuel assemblies, as described in the previous sections, and a tender process to obtain the necessary casks took place under the auspices of the IAEA. Six manufacturers from the USA, Russian Federation, Germany, France and the Czech Republic participated of the process, and ŠKODA JS a.s. under the auspices of the Ministry of Industry and Trade of the Czech Republic, was chosen as the supplier. Six ŠKODA VPVR/M casks were purchased by the NRI for shipment of LEU SNF; and ten casks, for shipment of HEU SNF, were purchased by the US Administration (approximately USD 4 M.) and given to the NRI with the condition that NRI would use them, as well as its own casks, for the Russian Research Reactor Fuel Return (RRRFR) programme. Also, by agreement between the NRI and the US DOE, and taking into account the experience of the NRI team from preparation and implementation of the transport from the Czech Republic to the Russian Federation, it was established that once the shipment of SNF from the NRI was completed, all 16 casks should be made available for return shipments of SNF from other countries to the Russian Federation.

The VPVR/M ŠKODA cask, shown in Fig. 17, consists of a cylindrical cask body made of cast steel with smooth outer surface. A double-lid system made of stainless steel (prim. lids) and carbon steel (sec. lids) and equipped with metallic Helicoflex sealing rings is tightly bolted to the top and bottom ends of the cask body, guaranteeing the leak-tight containment of the radioactive contents during transport as well as the safe long term containment during storage. Table III presents some general information for the VPVR/M ŠKODA cask, which has a unique top or bottom loading/unloading design [3].

Two different fuel baskets can be inserted into the cavity of the VPVR/M ŠKODA cask to accommodate of the following types of research reactor spent fuel: IRT-2M, 3M, EK-10, S-36, VVR-M, C, and TVR-S. It can also accommodate hermetic stainless steel canisters used for damage fuel elements. For transport the VPVR/M ŠKODA cask is equipped with shock absorbers at the top and bottom ends of the cask [4].

FIG. 17. Transport Cask SKODA VPVR/M.

The VPVR/M ŠKODA cask complies with the international regulations of the IAEA (International Atomic Energy Agency) for package designs of type B(U) and nuclear safety class I. It has Czech Republic certificate of CZ/048/B(U)F-96 (REV.1), and Russian certificate for package design in compliance with the Russian Federation regulations. It is also certified for transport through Slovakia and Ukraine.

TABEL III. GENERAL INFORMATION OF VPVR/M ŠKODA CASK.

Mass - Total 12 450 kg	**Internal arrangement**
- Body 9 000 kg	36 fuel assemblies
- Lids 1 650 kg	Type IRTM, EK-10, WWR-M or SS
- Shock absorbers 1600 kg	Hermetic canisters with damage
- During loading 9 900 kg	Fuel elements
Dimensions	**Main materials**
- External: without and with shock absorbers	Body cast steel Prim. lids stainless steel
- Height 1505 (2145) mm	Sec. lids carbon steel
- Diameter 1210 (1500) mm	Shock absorbers carb. steel, wood
Cavity dimension: - height 885 mm	- diameter 600 mm

In addition to the VPVR/M cask, ŠKODA also provided the ancillary equipment and agreed to maintain and store the casks for ten years (the life of the RRRFR Programme). The ancillary equipment for the VPVR/M cask system included a basket alignment tool, cask manipulating frame, cask tie down system, drying/leak-testing equipment, a cask lift fixture, fuel-handling tools, and specially designed and certified International Shipping Organization (ISO) containers, capable to accommodate 2 units of the VPVR/M cask during transportation, as shown in Fig. 18. The ISO container has the removable lid and the backside. Fixation tools are component parts of the ISO container [3].

FIG. 18. Transport Casks VPVR/M in the ISO Container.

8. PREPARATION FOR SHIPMENT

After the signature of the contract in 2005, NRI started working the negotiations to have the VPVR/M cask certified in four countries, i.e. Czech Republic, Russian Federation, Slovakia, and Ukraine. Finally the cask was licensed for transport by road, railway, river, and sea. It was also licensed to transport 10 fuel types and store three types of Russian origin research reactors fuels.

Additional preparations prior to the first cask loading included:

- Calculation and assemble data files for each FA that had to be transmit to Mayak for acceptance (252 + 91 IRT-2M FAs, 206 canisters with EK-10 FAs / fuel rods);
- Preparation of all documentation and procedures needed for loading, drying, testing, and storage of the VPVR/M casks in the NRI;
- Negotiations with the State Office for Nuclear Safety, IAEA and EURATOM, about verification of the loading FAs and cask sealing by their authorized inspectors;
- Preparation of fuel and cask handling equipment and facilities for operations;
- Sipping test and visual inspection of all FAs;
- Installation of ancillary equipment;
- Cask manipulations training.

When the casks were received, NRI conducted two tests to demonstrate that the newly designed, fabricated, and installed cask and fuel handling equipment worked properly at the reactor facility and at the HLWS facility; and that the detailed operating procedures were complete and operational. The testes were also performed to train NRI technicians for handling the ISO container, VPVR/M cask, loading fuel into the basket remotely, and drying and sealing of the cask. One demonstration used dummy fuel assemblies and the second demonstration used real HEU SNF assemblies. Both demonstrations confirmed NRI's preparedness, provided training for the operators, and were used to obtain approval from the State Office for Nuclear Safety (SONS) of Czech Republic for actual fuel and cask handling and loading operations.

The first cask was loaded with SNF in March 2007, to confirm adequacy for the VPVR/M shielding, and placed in the annex of HLWS facility. Considering that most of the EK-10 and some IRT-2M had already been encapsulated and accommodated in baskets compatible with the VPVR/M cask, following the first cask loading, operations went smoothly with no significant problems. It took about three days to load, seal, and prepare each cask for shipment. By the end of August 2007 all 16 casks had been loaded and placed in the HLWS facility extension cask storage vault.

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

- Transport cask in the ISO container to the respective SNF loading site;
- Dismantling the cask in the manipulation frame (see Fig. 19);
- Transport of the cask to the SNF storage facility (pool or hot cell);
- Putting the basket inside the loading facility (pool or hot cell);
- Loading the basket with SNF, and retract basket into the cask;
- Flushing of the cask with hot air, for desiccation of the cask;
- Cask closure and helium leaking test;
- Cask sealing by IAEA and EURATOM seals.

FIG. 19. VPVR/M cask manipulation frame (left) and basket with lower lid in manipulation frame (right).

Once the loading procedure was approved, three casks were loaded at the reactor site, and the other thirteen on the HLWS facility.

In the AFR1 pond, in the reactor building, three casks were loaded with 101 IRT-2M fuel assemblies, 91 with 36 % enrichment and 10 with 80 % enrichment. The loading was made between March and May 2007.

Considering that the VPVR/M cask can be bottom loaded, to load the cask in the AFR1 storage pool, a specially designed basket handling tool was used for lowering the selected basket from the cask into the storage pool, together with the lower lid, as shown in Figures 20 and 21. The basket was then filled manually with the FAs, using a special manipulation rod, and once it was filled, the basket was pulled back inside of the cask [3].

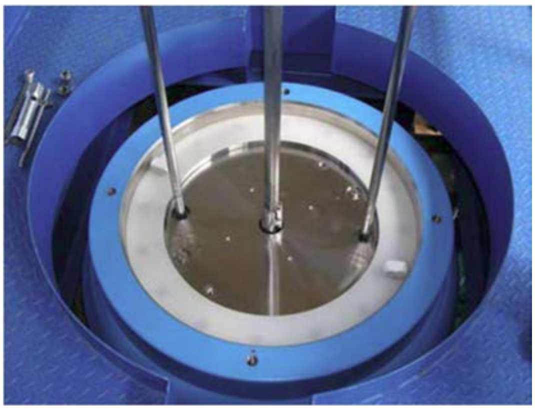

FIG. 20. The attached central suspension/ hanger and centring fixture pull rods.

FIG. 21. Basket with lower lid in the storage pool.

For the loading operation the crane and lift fixtures were equipped with a digital dynamometer, used to monitor the weight of the basket during reinstallation into the cask, with the purpose to prevent the disruption of the central suspension/hanger. A specially designed cask transport carriage, which moves by rail, shown in Figure 22, was used for the loading operations at the reactor site. It serves for transferring the cask from the reactor hall to the reactor annex pool for SNF storage. Also, the shielding above the pool was used during the loading of the SNF. The shielding was needed to protect the workers from radiation when the loaded basket was lifted out of the water and before it is completely inside the cask. After they were loaded in the reactor building, all the three casks were transported to the HLWS facility, as shown in Figure 23.

FIG. 22. VPVR/M cask being moved into the SNF storage area.

FIG. 23. Transport of VPVR/M cask with SNF to the HLWS facility.

In the HLWS facility, six casks were loaded with 206 canisters containing the EK-10 FAs/fuel rods previously encapsulated in the hot cell; and seven casks were loaded with 242 IRT-2M, 80 % enriched, fuel assemblies. From the total of 242 IRT-2M FAs, seven were leaking, and had been repackaged in the hot cell. The loading operation in the HLWS facility was performed between May and August 2007.

In the HLWS facility the EK-10 canisters were loaded from the hot cell by manual and robotic manipulators (see Fig. 24). To load the IRT-2M FAs, from the storage pool, the casks were positioned on top of the pool, into the special platform, and loading the IRT-2M FAs was performed using a manual manipulation rod (see Fig. 25). The loading in the HLWS facility was performed between May–August 2007 [2].

FIG. 24. Loading the SNF from the hot cell in the HLWS facility.

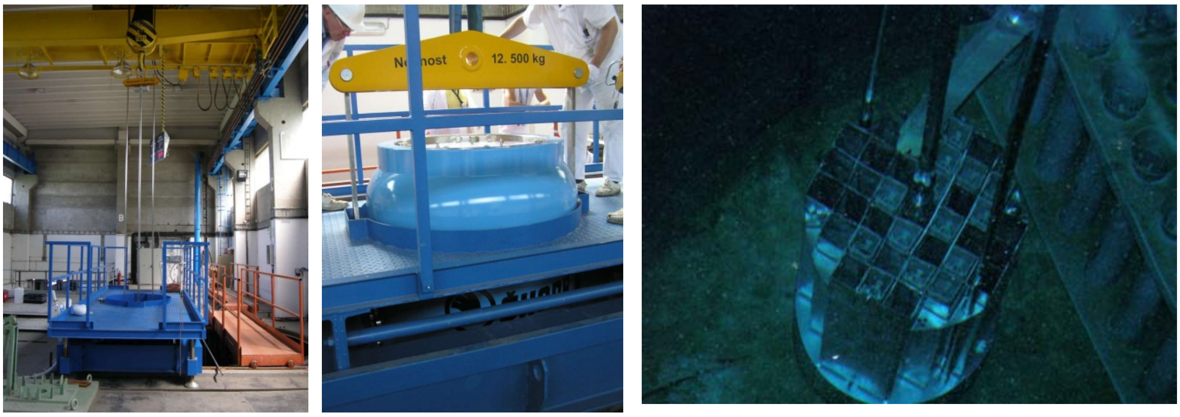

FIG. 25. Loading the SNF from the pool of the HLWS facility.

After each loading operation, representatives of IAEA and EURATOM installed seals on the cask, to confirm that it was not open without authorization, and the cask was transferred to the annex of the HLWS facility, to be kept in an especially designed storage vault, as shown in Figure 26. In October of 2007, the inventory of fuel in NRI was as shown in table IV.

FIG. 26. Storing the Skoda casks in the especially designed cask vault in the Annex of the HLWS facility.

TABLE IV. TOTAL INVENTORY OF LVR-15 RESEARCH REACTOR FUEL, PER LOCATION OF NUCLEAR RESEARCH INSTITUTE REŽ IN OCTOBER 2007

Location	**Type of fuel assemblies**	**Number of assemblies** [a)]
Fresh Fuel Storage		73
Active core of the LVR-15 reactor		28
AR pond	IRT-2M, 36%	32
HLWS annex – in VPVR/M casks	IRT-2M 80% / IRT-2M 36% / EK10	252 / 91 / (208)

[a)] On this column, 208 represents the number of canisters. Because in some drums there were only pins – the assemblies had been disassembled – and more than 16 pins could be placed inside of a single canister – up to 36 pins could be placed in one canister – so the number of canister with EK-10 fuel at the end of repackaging work were 208.

9. THE TRANSPORT OF THE SNF

In the winter of 2007, the 16 VPVR/M casks with EK-10 and IRT-2M FAs were ready to be loaded in the respective ISO containers. After the large concrete lids of the vault were removed, representatives of SONS, EURATOM and IAEA inspected the seals of each cask, to confirm that they had not been opened while in storage, and, as shown in Fig. 27. Then the casks were loaded into the ISO containers, as shown in Figure 28, and the containers were transported on trucks to the railroad station where they were transferred onto the railroad carriages. The physical protection and emergency preparedness secured the convoy during all times, until the full load of SNF arrived in Mayak, in the Russian Federation.

FIG. 27. Interior of the ISO container with one Skoda VPVR/M .cask.

FIG. 28. The ISO container being loaded on the truck.

10. CONCLUSION

The shipment of SNF is a very demanding operation and requires the cooperation of a number of organizations to assist in preparing, shipping and receiving the spent fuel, and respective operational facilities. In this special case, the involvement included many national institutions, regulatory agencies, and contractors, and the participation of institutions in four different countries: Czech Republic, Slovakia, Ukraine, and the Russian Federation.

Successful completion of the NRI SNF shipping and disposition project required a significant amount of planning and coordination, and since the beginning NRI was responsible for the entire project, including overall project management. NRI assigned a highly qualified senior technical person, with extensive management experience, to manage the project. DOE provided funding and oversight for NRI. The scope, cost, and schedule for activities were negotiated, and NRI issued task plans. DOE authorized the work by issuing a task assignment that established the scope of work, deliverables, schedule, and a fixed price.

On 8 December 2007, the shipment was completed. This shipment was the largest shipment of HEU SNF from a Russian designed nuclear research reactor.

The success of this project can be attributed to the excellent planning, scheduling, cost estimating, managing, and communications skills of the project manager. The project manager being fluent in Czech, Russian, Slovak, and English was a significant attribute.

11. FUTURE PLANS

In agreement with the contract signed in 2005, NRI is now working on the conversion of the LVR 15 research reactor to use Low Enriched Uranium (LEU) fuel. It is expected that the first LEU fuel assemblies be inserted in the core of the reactor during the first quarter of 2010, and the full core be converted by the end of January 2012.

The spent fuel inventory in October 2009 was 133 IRT-2M, 36% HEU fuel assemblies; 31 SNF in AR pond, 31 SNF in AFR1 pond, 28 in rector core and 43 fresh. It is our intention to ship some (21) fresh fuel to the Russian Federation in 2010, to have a second shipment of spent fuel to the Russian Federation in 2014, and after that date we plan to keep the spent fuel in AR and AFR1 ponds for a period of three to five years, before transferring them to Škoda VPVR/M casks for dry storage for up to 50 years, or if a final solution is decided before, dispose it accordingly, for example sending them for reprocessing or to a final repository.

As for the reactor LVR-15, normal operations are envisaged at least until 2028.

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SHIPMENT OF SPENT NUCLEAR FUEL FROM WWR-K RESEARCH REACTOR, KAZAKHSTAN

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Abstract

WWR-K, the largest nuclear research facility of the Institute of Nuclear Physics of the National Nuclear Centre of the Republic of Kazakhstan (INP NNC RK) is a multipurpose pool type research reactor in operation since 1967. The reactor was shut down from 1988 until 1997, for safety enhancement and modernization of the reactor core. Before shutting down the reactor, shipment of spent nuclear fuel (SNF) to the Russian Federation was a routine operation, however with the disintegration of the former Soviet Union in 1991, this procedure was suspended, and INP NNC RK was compelled to solve the problem of the SNF independently. Since there is no SNF reprocessing facility in Kazakhstan, and considering that by that time the priority was to implement the modifications to increase the safety of the WWR-K reactor, a decision was taken to simply collect the spent fuel assemblies and keep them in one of the 2 pools available for storage of the SNF. The capacity of the pools is 500 fuel assemblies, a number close to be reached after the reactor resumed operation in 1998. Considering the necessity to find an alternative solution for the SNF, contacts were made and in 2003 an IAEA special mission visited INP NNC RK, with the purpose to start negotiations in order to resume the shipping operations of the SNF to the Russian Federation. The negotiations involving The Russian Federation, United States of America, Republic of Kazakhstan and IAEA proceeded successfully, the final contract was signed in 2006, and on December 2008 the first train carrying the SNF left Kazakhstan in direction to The Russian Federation. In total 4 trips were needed to transport 278 SNF assemblies, representing about 74 Kg of Highly Enriched Uranium. The last trip was made in May 2009.

1. INTRODUCTION

WWR-K research reactor is the largest research facility in the Institute of Nuclear Physics of the National Nuclear Centre of the Republic of Kazakhstan (INP NNC RK). It is a multipurpose water-pool type research reactor with light water working as moderator, coolant and reflector. Its design and physical features are such that the reactor can be considered as belonging to the safest types of reactors, with a wide range of experimental and technological possibilities. It can also be classified as one of the best operational reactor, as it is the youngest of such type created in the former USSR.

2. WWR-K RESEARCH REACTOR

The reactor started operation in 1967 and during the next 20 years, it was intensively used for research and test of materials. It was also used for test of components of nuclear technology, especially for space nuclear power facilities developed in the USSR. In October 1988, the operation of the WWR-K reactor was suspended to allow the realization of activities to increase the reactor safety, especially in items related to high seismicity conditions, following a study carried out by experts from Kazakhstan. The work included a detailed study of seismic conditions of the reactor site; an evaluation of seismic resistance of the reactor building, facilities and equipment; the introduction of additional safety systems for earthquake situations. The safety enhancement programme, which also included the modernization of the reactor core finished in 1997, and it was followed by a re-commissioning programme. Physical tests and power start-up activities were successfully carried out, and since 1998, research reactor WWR-K resumed its regular operation programme.

Now the reactor is used for radiation tests of perspective materials for nuclear and thermonuclear reactors, production of radioisotopes for medical and industry applications, and neutron-activation analysis of various materials. The operational cycle of the reactor is such that after being refuelled, the core reactivity allows the reactor to work continuously during 20 days, at its nominal power and with a thermal flux over 10^{14} n/cm^{2}sec.

Along with the reactor technological systems, the reactor complex includes the following facilities:

- Critical assembly;
- "Hot" cells;
- Gas-vacuum loop;
- Neutron radiography facility;
- Hydro and pneumatic material transfer system.

The main characteristics of the reactor are shown in Table I.

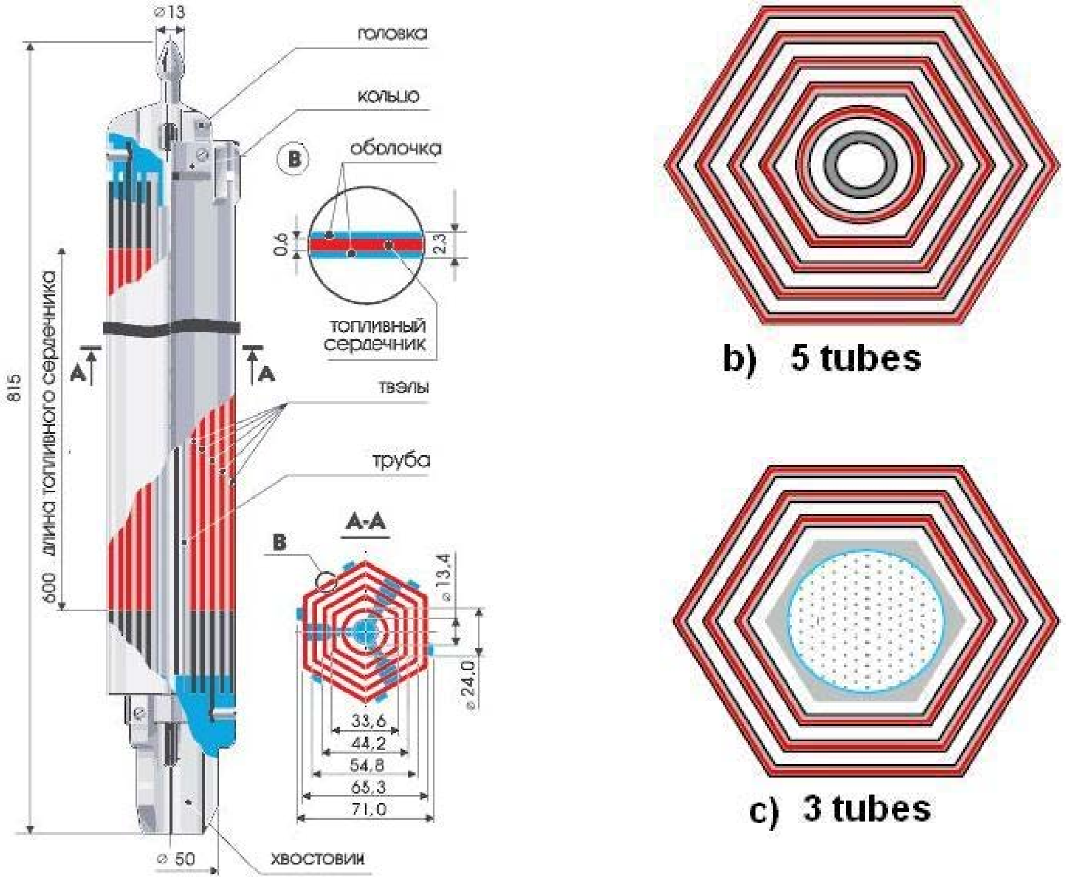
TABLE I. PHYSICAL CHARACTERISTICS OF THE WWR-K RESEARCH REACTOR

Parameter	Value
Rated power, MW	6
Fuel material	Uranium
Enrichment of ^{235}U, %	36
Type of fuel assembly (FA)	WWR-C
Moderator, coolant and reflector	Light water
Height of active core, mm	600
Diameter of active core, mm	720
Mass of ^{235}U in working loading at average burnup of 18 %, g	6300
Absorbing material of control rods	boron carbide
Quantity of control rods	8

3. WWR-K FUEL ELEMENTS

WWR-K reactor utilizes 36% enriched WWR-C Russian type fuel assemblies, shown in Fig. 1. Two types of fuel assemblies are used in the reactor core standard and partial. Standard fuel assemblies have 5 elements, as shown in Fig. 1(b), and partial fuel assemblies have three elements as shown in Fig. 1(c). The two types of fuel assemblies are similar; the only difference is that three element assemblies have an internal free cavity of hexahedral shape (the size on a turnkey basis 32.7 mm), intended to allow the insertion of the reactor control rods, and, whenever it is necessary the insertion of irradiation devices. Table II presents the main characteristics of the reactor fuel assemblies.

After reaching the maximum allowable uranium burnup, the fuel assemblies are unloaded from the reactor core and transferred to the spent nuclear fuel (SNF) storage facility, where they remain for at least a three years cooling period, to allow the removal of the heat generated by decay of the short lived uranium fission products. After the three years decay period, the fuel is considered ready to be shipped, as it was done traditionally until 1991.



b) 5 tubes

c) 3 tubes

FIG. 1. WWR-C Fuel type assembly.

TABLE II. CHARACTERISTICS OF WWR-C FUEL ASSEMBLIES

Quantity of fuel elements in Fuel assembly (FA)	5 (Standard) / 3 (Partial)
Fuel element type	three-layer, dispersive
Thickness of fuel element, mm	2.3
Fuel enrichment on ^{235}U, %	36
Total ^{235}U weight, g	109 / 83
^{235}U density in active core, g/l	45.1 / 34.3
Heat transfer area per unit of volume in active zone, cm^{2}/cm^{3}	3.54 / 2.71
Material of fuel cladding and end fittings	aluminium alloy
Fuel type	UO_2 Cermet
Total weight, kg	3.9 / 2.9
Coolant	Demineralized light water
Inlet coolant temperature, °C	35
Maximum temperature of the fuel cladding, °C	up to 95 (boiling on the fuel element surface is not permitted)
Average ^{235}U burnup, %	Not less than 40

4. SPENT FUEL STORAGE POOLS

The SNF storage facility has two storage pools. Storage pool No. 1 is a 2.5 × 1.2 m^2 pool and storage pool No. 2 is a 2.7 × 2 m^2. Storage pool No. 1 is used to store spent fuel assemblies immediately after removal from the reactor core. It is located near the reactor pool wall and below the reactor hall floor. This pool forms the bottom part of the reactor building and is capable of withstanding an earthquake of reasonable magnitude, as per design features [1].

Both pools are 5 meters deep, and filled with demineralized water. Storage pool No.1 has a total volume of 10 m^3, and a capacity to store about 200 fuel assemblies, in two levels. Storage pool No.2 has a total volume of 17.8 m^3, and a capacity to store about 300 fuel assemblies, also in two levels. Figure 2 shows the top of storage pool No.1 and the fuel assembly storage racks on storage pool No.2.

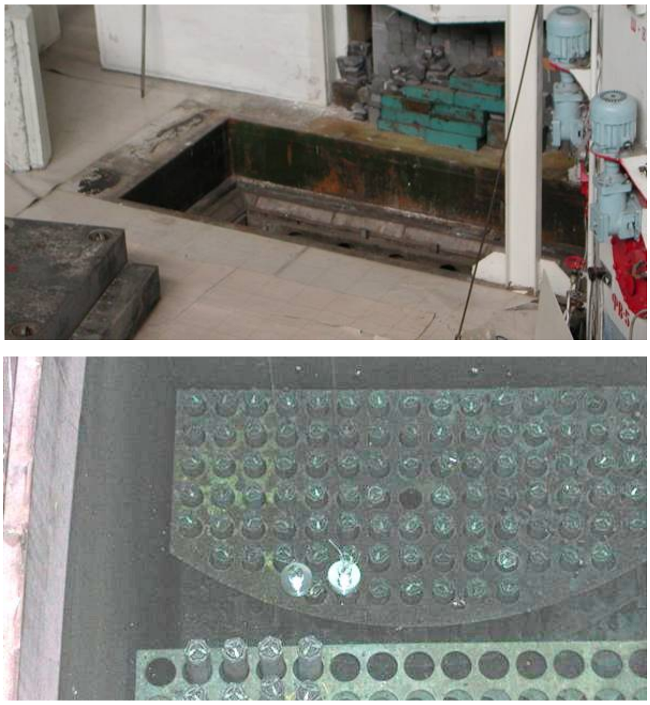


FIG. 2. Top of storage pool No.1 (top) and fuel assembly storage racks on storage pool No.2 (bottom).

To prevent any degradation of the fuel assemblies while they are in the SNF storage facility, it is necessary to provide appropriated fuel cooling ventilation for the facility, a constant radiation monitoring programme of the ventilated air, and also to keep the chemical parameters of the pool water within specified limits. Table III lists the storage pool parameters, with respective limits.

TABLE III. PARAMETERS OF WATER ON STORAGE POOL AND RESPECTIVE LIMITS

Parameter	Specified value
pH	between 5.5 and 6.5
Conductivity (mcS/cm)	< 4.0
Hardness (mcg-eq/kg)	<3,0
Cl (mcg/kg)	<50
Al (mcg/kg)	<50
Fe (mcg/kg)	<50
Cu (mcg/kg)	<10

Departure from the specified values listed in Table III can lead to corrosion of the spent fuel aluminium cladding, and if no action is taken, the corrosion process can progress through the cladding, causing an eventual exposition of the fuel cermet, with consequent release of radioactive fission products into the water of the SNF storage pool. Considering that the risk of fuel assembly degradation increases with the storage time in the water, therefore an action is necessary to remove the fuel from the pool as soon as it is possible. Two options must be considered when the decay heat production reaches insignificant values. First option is to send the aged assemblies for processing in the specialized companies that acquired such technology. The second option, considered as an alternative is to pack the "cooled" spent fuel assemblies into tight shielded stainless steel containers, and keep them in a "dry" storehouse.

5. PRIOR SHIPMENTS OF SPENT FUEL

During the 70s and the 80s, by the time of the former Soviet Union SNF of all Russian origin research reactors were taken for reprocessing at the "MAYAK" facility (in the Chelyabinsk area) each 5–7 years. The last shipment of SNF from WWR-K reactor to MAYAK was made in 1986. After the disintegration of the former USSR in 1991, this procedure was suspended, and the Institute of Nuclear Physics was compelled to solve the problem of the SNF treatment independently. Considering the absence of SNF reprocessing facilities in Kazakhstan, and considering that by that time the priority was to implement the necessary modifications to increase the safety of the WWR-K research reactor, that was being kept in shutdown condition, a decision was taken to simply collect the spent fuel assemblies and keep them in the pools of the "wet" storehouses.

5.1. Last shipment

In 2003, an IAEA special mission visited Kazakhstan with the objective to organize a new shipment of the stored research reactor SNF. The mission was composed by experts from the US Department of Energy and from Russian Federal Atomic Energy Agency (Rosatom, now the Nuclear Energy State Corporation). At the end of the mission the experts had defined a programme establishing the necessary steps to resume the SNF shipments to MAYAK.This mission was followed by a long period of negotiations and coordination involving Kazakhstan, USA, the Russian Federation and the IAEA. In particular, IAEA promoted all negotiations to reach the necessary agreements for the acceptance, by the Russian Federation, of research reactor spent nuclear fuel made with uranium originally enriched in the Russian Federation. The agreement between the governments of the Russian Federation and the United States of America, about the necessary cooperation toward the realization of SNF shipment was signed in May 2004.

It is necessary to notice that, by that time two agreements were the basic legal bases for the participation of Kazakhstan in a joint project with the USA and the Russian Federation on SNF export. The first agreement was between the Government of the Republic of Kazakhstan and the Government of the Russian Federation on cooperation in the field of peaceful use of atomic energy, signed on September 1993. The second was an agreement between the United States of America and the Republic of Kazakhstan concerning the destruction of silo launchers of intercontinental ballistic missiles, emergency response, and the prevention of proliferation on nuclear weapons, signed in December 1993. These two documents became the first documents of the independent Kazakhstan in the field of atomic energy.

In 2005, INP NNC RK organized a meeting with the representatives of the US Department of Energy in which the project of the SNF shipment to the Russian Federation was discussed in detail. In 2006, in coordination with the Department of Energy and Mineral Resources of RK, the Institute of Nuclear Physics of NNC RK concluded the contract with the National Nuclear Security Administration of the US Department of Energy, related to the preparation work for the SNF shipment.

The framework the programme for SNF transportation was developed and coordinated by INP NNC RK with all corresponding state organizations. This programme included the programme of physical protection; the emergency reaction procedure, and the environmental impact assessment. The validity

15 of the Russian certificate for the TUK-19 transport package sets, approving its design and utilization in SNF transport, was extended to the territory of Kazakhstan. The certificate of "goods" origin and the license for transportation of nuclear materials and SNF export were properly prepared and approved. Repairs were made of premises and equipment required for use during SNF packing loading and transportation. Trucks for transportation of containers with SNF from the Institute of Nuclear Physics to railway station were purchased and additionally equipped.

Calculations of the following parameters were made for each spent fuel assembly:

- Burnup;
- Remaining decay heat emission;
- Radioactivity;
- Radionuclide composition.

In September 2008 the permission to proceed with the SNF shipment, from Kazakhstan to the Russian Federation for reprocessing, was obtained from the Department of Energy and Mineral Resources of RK. In November the contract between INP NNC RK and the Federal Centre on Nuclear and Radiation Safety of Rosatom (FCNRS) – the organization, authorized by the Government of the Russian Federation to conclude the foreign transactions on SNF import, on shipment services and reprocessing of the spent nuclear fuel of the WWR-K research reactor was signed. In the framework of this contract, the SNF shipment from WWR-K to MAYAK was made systematically between December 2008 and May 2009.

The transportation of the SNF from WWR-K RR research reactor was made using the TUK-19 transport package sets manufactured in the Russian Federation and shown in Fig. 3. The general characteristics of the TUK-19 transport package set are shown in Table IV.

FIG. 3. TUK-19 transport package.

After being loaded, the cargo was transported to the railway station using two specially equipped KAMAZ trucks, each one loaded with two complete sets of transport packages TUK-19, as shown in Fig. 4. At the railway station the TUK-19 packages were transferred to the special railway TK-5 cars-containers, shown in Fig. 5, and the special train departed to the final destination station in the Russian Federation.

TABLE IV. CHARACTERISTICS OF TUK-19 TRANSPORT PACKAGE

Characteristic	Value
External dimensions (mm)	D 910 × H 2170
Cavity dimensions (mm)	D 220 × H 1400
Total weight (loaded) (kg)	4750
Capacity for SNF of WWR-C type (item)	4
The maximum enrichment of uranium on ^{235}U (%)	90
The maximum heat emission (W)	112
Time of FA cooling (year)	3
Quantity of packages available (item)	16 + 4
Way of transportation	railway (TK-5); car

FIG. 4. KAMAZ trucks, loaded with transport packages TUK-19.

FIG. 5. Special railway TK-5 cars containers.

The special train made four trips from Kazakhstan to the Russian Federation, transporting a total of 278 spent nuclear fuel assemblies. Altogether, it represented about 74 kg of highly enriched uranium plus a considerable quantity of radioactive products of its fission. The first trip was made in December 2008, followed by trips made on March, April and May of 2009. The operation was completed, when the last shipment was safely delivered to the destination.

As a conclusion, we can affirm that with the joint efforts of experts from Kazakhstan, the Russian Federation, the USA and IAEA, the shipment of the WWR-K reactor spent nuclear fuel to the country of origin was a complete success. It was done safely, and according to the planned time and terms.

6. PRESENT SITUATION

WWR-K research reactor presently utilizes 36% enriched WWR-C Russian type fuel assemblies. Studies have shown the feasibility to convert the reactor to utilize Low Enriched Uranium (LEU) fuels [2], and plans are been made to concretize the full conversion of the reactor to use LEU fuel. According to the planning, three lead test assemblies will be irradiated during the period of 2011–2013, the necessary modifications of the facility will take place in 2012–2013, and the new LEU fuel will be used starting in 2013. After cooling of the remaining HEU spent fuel it will be also shipped to the Russian Federation.

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THE MANAGEMENT OF TRIGA SPENT FUEL AT ENEA RC-1 RESEARCH REACTOR

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Abstract

TRIGA Mark II reactor of ENEA's Casaccia research Center (in Italy named RC-1) reached first criticality in 1960. Reactor core was realized with 61 standard TRIGA fuel elements, aluminium clad. In this condition, the reactor was operated until August 1965 at a steady state power level of 100 kW. In the summer of 1965, a programme was established to increase the reactor power to 1 MW. After significant plant modifications (in order both to adapt the reactor to the new operative circumstances, including safety regulations, and to extend reactor flexibility in the widest research areas), the new criticality was reached in July 1967. The 1 MW reactor operative configuration was initially obtained with 76 standard TRIGA fuel elements, but stainless steel clad. The RC-1 Reactor is still operational and during these years, many fuel elements were used. In this paper we describe the facility, the infrastructure available for spent fuel storage, and the operative experience accumulated during these years in the management of RC-1 Spent Nuclear Fuel (SNF). The activities and the incumbencies during SNF shipment that was carried out in 1999, in the frame of the USA Return of Foreign Research Reactors Spent Fuel Programme, are also described.

1. TRIGA RC-1 RESEARCH REACTOR, FUEL AND SPENT FUEL FACILITIES [1]

RC-1 is a pool thermal research reactor having a core contained in an aluminium vessel and placed inside of a cylindrical graphite reflector, bounded with lead shielding. The biological shield is provided by a concrete structure having an average thickness of 2.2 m. Demineralized water, filling the vessel, ensures the functions of neutron moderator, cooling mean and first biological shield. Figure 1 shows the horizontal and vertical sections of the reactor.

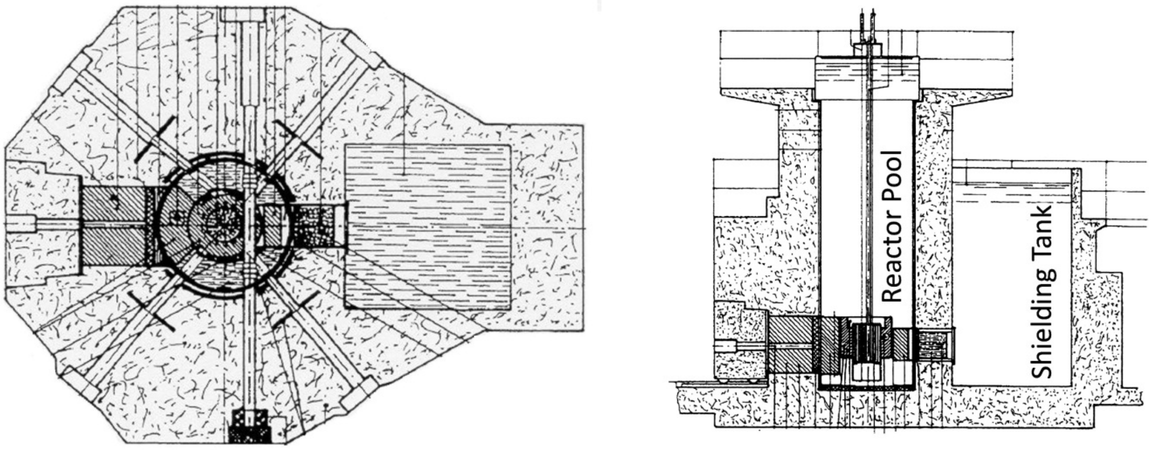

FIG.1. Horizontal and vertical reactor sections of RC-1 research reactor.

The reactor is mainly utilized for training, radiochemistry and NDT activities. With neutron activation and radiochemical treatments, several short lived radiopharmaceutical radioisotopes are produced. A small homemade neutron tomography system is operating at the thermal column. A larger system is under construction, to be installed in correspondence of the tangential channel.

Reactor control is ensured by four boron carbide absorber rods: two fuel follower shim rods, one fuel-follower safety rod, and one power regulation rod.

The produced thermal power is removed by natural water circulation through a suitable thermo-hydraulic loop comprehending heat exchangers and cooling towers. Table I shows the value of the neutron flux on some of the irradiation facilities and positions available in the facility.

TABLE I. NEUTRON FLUX AVAILABLE FOR IRRADIATION ON RC-1 RESEARCH REACTOR

Description	Neutron flux [n·cm-2·s-1]
Lazy Suzan (rotating rack with 40 positions)	2.00·1012
Pneumatic transfer system	1.25·1013
Central channel	2.68·1013
Thermal column collimator	~1.0·106
Tangential piercing channel (w/o) collimator	~1.0·108

The reactor and the experimental arrangements are surrounded by a concrete shield structure. The core and the reflector assemblies are located at the bottom of an aluminium tank (190.5 cm diameter). The overall height of the tank is about 7 m, therefore the core is shielded by a 6 m layer of water. The core, surrounded by the graphite reflector, consists of a lattice of TRIGA stainless steel standard fuel elements, graphite dummy elements, control and regulating Rods. There are 127 channels divided in seven concentric rings (from 1 to 36 channels per ring). The channels are loaded with fuel rods, graphite dummies plus regulation and control rods, depending on the power level required. One channel houses the startup Am-Be source, while two fixed channels (the central one and a peripheral one) are available for irradiation or experiments. A pneumatic transfer system allows fast transfer of sample from the radiochemistry end station to the peripheral irradiation channel, and vice-versa.

The diameter of the core is about 56.5 cm while the height is 72 cm. Neutron reflection is provided by graphite contained in an aluminium container, which is surrounded by 5 cm of lead acting as a thermal shield. The core structure has aluminium grid plates (top and bottom) that hold the core components in place: the top grid plate has 126 holes for insertion of the fuel elements, control rods and a central thimble for high flux irradiations. The reactor core is cooled by natural circulation of the water in the reactor pool. Figure 2 shows a vertical section of the reactor core and a diagram sketch of the core configuration.

The TRIGA fuel elements consist of a stainless steel clad (AISI-304, 0.5 mm thick, 7.5 g/cm^3 density) characterized by an external diameter of 37.3 mm and a total height of 720 mm, with end cap included. The fuel is a cylinder (381.0 mm high, 36.3 mm in diameter, 5.8 g/cm^3 of density) of a ternary alloy uranium-zirconium-hydrogen with a metallic zirconium rod inside (381.0 mm high, 5.0 mm in diameter, 6.49 g/cm^3 of density). In the fuel H-to-Zr atom ratio is 1.7 to 1; and the uranium, enriched to 20% in ^{235}U, makes up 8.5% of the mixture by weight. The total uranium content of each rod is 190.4 g, of which 37.7 g is fissile. There are two graphite cylinders (87.0 mm high, 36.3 mm in diameter, 2.25 g/cm^3 of density) at the top and bottom of the fuel rod. Externally two end fittings are present in order to allow the remote movements and the correct placement of the fuel into the grid plates. Figure 3 shows a TRIGA standard fuel element. The graphite dummies are similar to a fuel element but the cladding is filled only with graphite.

The regulation rod has the same morphological aspect as the fuel rod. The only difference is that instead of the mixture of the uranium-zirconium-hydrogen ternary alloy, the cylinder has the neutron absorber that is made with graphite and powdered boron carbide. The control and safety rods are "fuel-follower type", which means that the geometry is similar to that of the regulation rod but in its bottom there is a fuel element. In the vertical section of the reactor core, shown in Figure 2, it is possible to see a schematic view of a fuel-follower control rod.

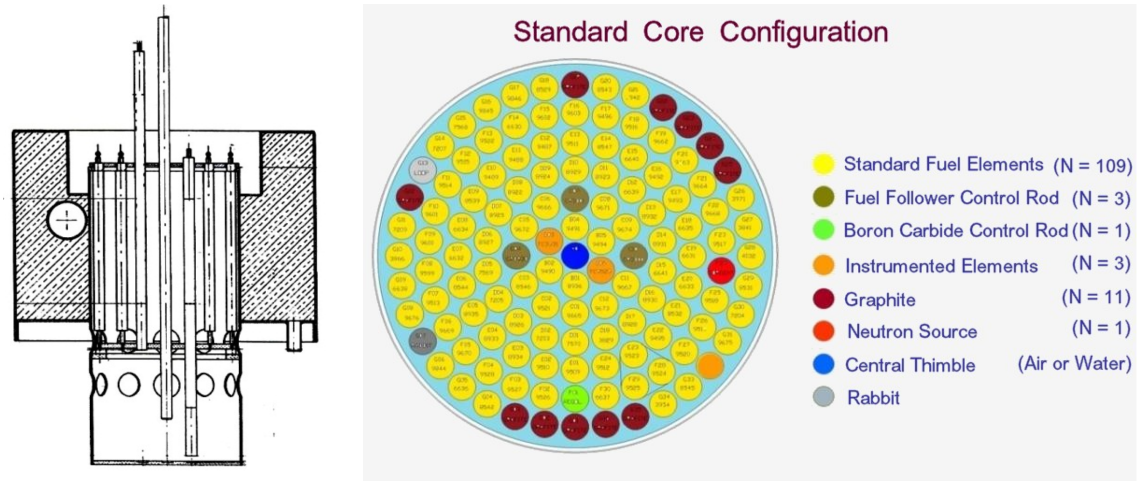

FIG. 2. Vertical section of RC-1 TRIGA reactor core (left) and reactor core configuration (right).

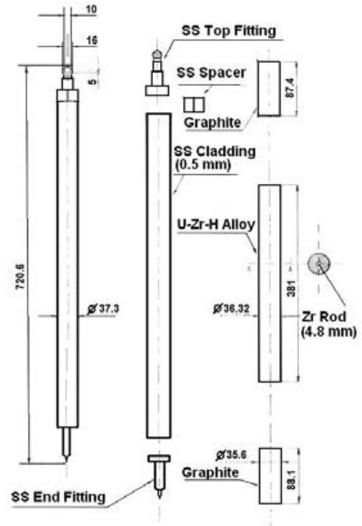

FIG. 3. Standard TRIGA type fuel.

A characteristic of the TRIGA fuel alloy is the used percentage of uranium that allows a very peculiar metallurgical stability. Even if all the ^{235}U is burned, only 0.7% of the alloy atoms will change.

According to the technical conditions established by Italian Nuclear Regulatory Body, the fuel has to be removed from the reactor core when it reaches a burnup equal to 35%, of the original amount of ^{235}U. This means that when the fuel is removed from the core, the stability of the alloy is still classified as “very good” and the irradiated fuel is quite “cold”.

TRIGA RC-1 has several racks and facilities where the reactor spent fuel can be stored. All the racks and facilities are “wet” and never, during the plant life, it was proposed, or even hypothesized, the possibility to use a dry storage method.

The first set of devices used to store the spent fuel consists of 3 racks, each one with 12 positions, situated on the side of the reactor pool tank, as shown in Figure 4. The racks are kept about 3.00 m below the water surface, a distance large enough for the water to shield the radiation emitted by the fuel. These racks are mainly utilized for temporary storage of the fuel during core loading/unloading/reconfiguration operations. Also when the fuel is definitively unloaded from the core, it is kept in these racks for several months in order to allow the decay of the fission products.

FIG. 4. Reactor pool storage rack.

For the long term fuel storage there are five stainless steel pits into the reactor building and a fuel storage pool outside of the reactor building. The pits are situated in a corner of the reactor building. Each pit has a special circular rack located in the bottom, about three meters below floor level, which can accommodate up to 19 fuel elements. The total storage capacity of the pits, shown in Fig. 5, is 95 fuel elements.

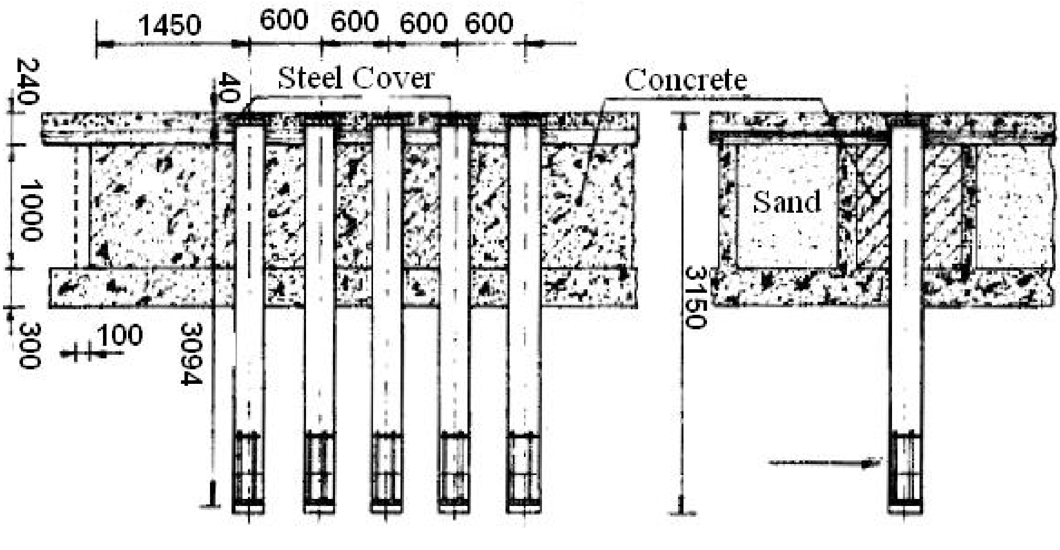

FIG. 5. Spent fuel storage pits.

The external spent fuel storage pool is located into a special building, illustrated in Figure 6, near the reactor building. It can host 279 spent fuel elements and recently, the walls of the pool were covered with a stainless steel liner. This storage pool has never been used so, the building is totally “clean”.

The reactor has also a shielding tank, an experimental facility that was already used, and can be used again in the future, for temporary storage of the spent fuel. This shielding tank, shown in Fig. 1, was licensed for temporary storage of RC-1 spent fuel in 1999, when its use was essential to allow the shipment of the TRIGA RC-1 spent fuel to the USA within the framework of DoE´s Foreign Research Reactor Spent Nuclear Fuel acceptance programme. Inside of this tank there are 12 racks, each one with a capacity to host 12 fuel elements, so the total storage capacity of the shielding tank is 144 elements.

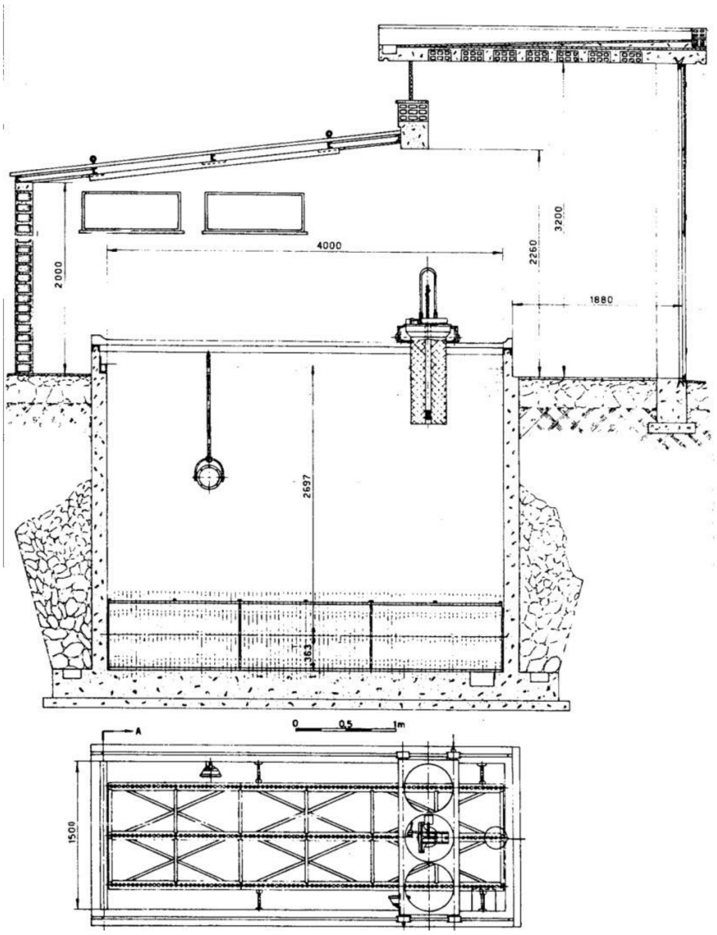

FIG. 6. External storage pool.

All the storage facilities are filled with demineralized water and provided with circulating pumps, filters and water quality control devices.

The fuel handling is made with a special electromechanical grapple tool and the transport around the plant site is made with the help of a "coffin" that can contain one element at a time.

2. FUEL MANAGEMENT AND CHARACTERIZATION [2]

2.1. Aluminium clad fuel element

During the first five years of operation, from 1960 to 1965, the nominal power of RC-1 reactor was 100 kW and the core was loaded with 61 (initially) to 63 (finally) aluminium clad fuel elements. At the end of the 100 kW operation cycle, the whole core was discarded into the reactor building pits where a total of 64 irradiated fuel elements were stored for 34 years. During a core loading operation, a fuel element felt into the reactor pool. After the recovery and a visual inspection, no cracks were found, but conservatively the element was definitely removed from the core, despite its very low burnup (8 days into the core at 100 kW). The element was canned and stored into one of the available pits.

The total amount of energy produced in the reactor during the 100 kW operating cycle was about 850 MWh, and the average fuel burnup was less than 2%. Table II shows a resume of the main average parameters of the fuel used during the 100 kW operating cycle. These data were calculated using computer code Origen 2 to demonstrate the accomplishment of the requirements established for shipping the fuel to USA.

TABLE II. PARAMETERS OF TRIGA RC-1 ALUMINIUM CLAD SPENT FUEL ELEMENTS

Total number of fuel elements		64
Average time in reactor (days)		1964
Average power level (kW)		1.6
Average cooling time (days)		11754
Average burnup (%)		1.8
Decay heat (watts)	0.244 (total)	0.004 (average)

U and Pu content (gr)

	Pre- Irradiation		Post Irradiation (on December 31, 1997)							
	U–total	^{235}U	U–total	^{235}U	^{236}U	^{233}U	Pu–total	^{239}Pu	^{240}Pu	^{241}Pu
Total	11566	2285	11530	2239	8.943	6.65E-06	2.53	2.506	2.34E-02	5.25E-05
Average per element	186.37	36.82	185.77	36.08	0.14	1.07E-07	0.041	0.040	3.78E-04	8.47E-07

Even considering that RC-1 procedures prescribe very restrictive values for the chemical characteristics of the water in the storage pits (conductivity is usually kept in a range of 0.5÷2.0 μmho/cm), during the pre-shipment visual inspection two aluminium clad elements were found to have some serious corrosion effects on the surface: two holes (~10÷15 mm in diameter). The surface of the clad was completely corroded and the fuel pellet was visible.

After that, an extraordinary radiological analysis of the water on the pits was carried out and the results are reported in Table III.

The results of this analysis showed that there was no significant contamination of the water: Only in pit n.1 there was a very small excess of cesium-137, with respect of "very pure water" (reference values in the 6th column of the Tab.III). The cleanness of the water is probably due to the low value of fuel burnup and the great capacity of TRIGA fuel to trap the fission products.

2.2. Stainless steel clad fuel elements

After upgrading the reactor power to 1 MW, the core was initially loaded with 76 stainless steel standard TRIGA fuel elements. From 1967 (first criticality, 28 July 1967) until 1999, a total of 82 fuel elements were unloaded from the core and 76 of them were selected to be shipped to the USA, together with the 64 aluminium clad ones (the total capacity of the cask was 140 TRIGA fuel elements). Also for these fuel elements, the main average characteristics were calculated by means of Origen 2 computer code in order to demonstrate the accomplishment of the requirements for shipping them to the USA. The results of the calculations are shown in Table IV.

The visual inspection of the SS clad elements showed that for these elements there was no problem of corrosion, bending or any other kind of damage. The surface of the cladding was just a bit reddened in correspondence to the length of the fuel pellets.

Since the shipment of the spent fuel to the USA, 123 additional TRIGA fuel elements have been used at RC-1. Twelve have been moved to the storage racks in the reactor pool, and 111 are still in the core. Table V shows the parameters calculated for these additional elements.

TABLE III. RADIOLOGICAL ANALYSIS OF WATER ON STORAGE PITS OF TRIGA RC-1 (Bq/l).

Radionuclide	PIT1	PIT2	PIT3	PIT4	PIT5	BIANCO
^{137}Cs	0.27	<0.097	<0.11	<0.090	<0.084	<0.092
^{234}Th	<5.41	<4.36	<5.82	<7.79	<8.09	8.60
^{234}Pa	<0.21	<0.21	<0.24	<0.19	<0.25	<0.27
^{230}Th	<25	<25	<47	<43	<33	<32
^{226}Ra	<1.57	<1.44	<1.60	<q.50	<1.67	<1.41
^{214}Pb	0.37	0.23	<0.17	<0.18	0.24	<0.18
^{214}Bi	<0.20	0.91	0.36	1.10	0.32	0.53
^{210}Pb	<12.4	<9.62	<120	<8.60	<8.93	<10.8
^{235}U	<0.11	0.20	0.16	<0.09	<0.11	0.18
^{231}Th	1.74	1.71	2.16	2.06	1.75	2.13
^{231}Pa	<16.9	<8.75	<10.4	<8.55	<12.0	<12.9
^{227}Th	<0.76	<0.70	<0.77	<0.72	<0.79	<0.69
^{223}Ra	<0.70	<0.52	<0.46	<0.37	<0.58	<0.48
^{219}Rn	<0.88	<0.64	<0.60	<0.55	<0.62	<0.56
^{228}Ac	0.73	0.63	0.71	<0.34	0.66	0.88
^{228}Th	<8.80	12.8	<10.4	11.8	13.1	<8.74
^{224}Ra	<1.51	<1.30	<1.51	<1.38	<1.51	<1.26
^{212}Pb	0.33	0.19	<0.18	0.20	0.26	0.27
^{212}Bi	<0.95	<1.06	<0.73	<1.14	<1.17	1.49
^{208}Ti	0.21	0.11	0.23	1.08	0.09	0.18
^{232}U	<2.52	<1.86	<1.62	<1.35	<2.05	<1.78
^{234}U	Rows outside the useful energy field					
^{238}U	Rows outside the useful energy field					
^{235}Np	<0.23	<0.24	<0.40	<0.09	<0.20	<0.23
^{236}Np	<0.06	<0.05	<0.04	<0.04	<0.06	<0.07
Att. total	3.65	15.1	3.61	22.0	22.7	14.3

As evidenced in Table V, the actual average burnup for the fuel elements in the reactor core is about 15%>. Only 30 of the fuel elements in the core have a burnup value between 20% and 30%. This means that there is no imminent problem of spent fuel storage for the RC-1 research reactor.

According to our previous experience, the wet storage in the pits can be adapted for the next 4060 years without problems of capacity or fuel integrity.

3. FUEL SHIPMENT TO THE USA

In 1999, in the frame of the US Foreign Research Reactors Spent Fuel Return policy (RRSFR), 140 TRIGA spent fuel elements (64 Al clad and 76 SS clad) were shipped to the Irradiated Fuel Storage Facility (IFSF) of Idaho National Engineering and Environmental Laboratory (INEEL) of Idaho Falls in the USA. This shipment had been scheduled in 1990, when TRANSNUCLEAIRE's PEGASO cask was selected as fuel container and TRIGA RC-1 plant structures were adapted for these circumstances. In particular, a rail mobile carriage was prepared to support and to move the PEGASO cask in the reactor room and outside. From hot storages in reactor room (racks in the reactor pool and pits underground reactor floor level), and using the reactor room travelling bridge-crane, a suitable coffin and a fuel handling tool, irradiated fuel elements could be transferred to the PEGASO cask, already arranged on the carriage placed in the reactor room. The shipment operation had already been approved by the Regulatory Body, but it had to be suspended because of the expiration of the previous USA "Off-Site Fuels Policy" programme in 1992. When the USA announced the new US Foreign Research Reactor Spent Nuclear Fuel Acceptance Program", in May 1996, the opportunity to ship the RC-1 spent fuel was considered again.

TABLE IV. PARAMETERS OF TRIGA RC-1 STAINLESS STEEL CLAD SPENT FUEL ELEMENTS

Total number of fuel elements	82
Average time in reactor (days)	7676
Average power level (kW)	9.1
Average cooling time (days)	3190
Average burnup (%)	32.8
Decay heat (watts)	31.47 (total) / 0.384 (average)

U and Pu content (gr)

	Pre- Irradiation		Post Irradiation (on December 31, 1997)							
	U–total	^{235}U	U–total	^{235}U	^{236}U	^{233}U	Pu–total	^{239}P	^{240}Pu	^{241}Pu
Total	15641	3130	14709	2070	202	1.32E-04	49.086	40.33	7.37	0.87
Average per element	190.74	38.17	179.38	25.25	2.46	1.61E-06	0.599	0.492	8.98E-02	1.06E-02

TABLE V. PARAMETERS OF TRIGA RC-1 STAINLESS STEEL CLAD FUEL ELEMENTS STILL ON THE FACILITY

		Initial Isotope Composition (gr)		Burn up(%)	Produced Energy (MWD)	Actual Isotope Composition (gr))		
		^{235}U	^{238}U			^{235}U	^{238}U	Pu–total
Fuel still in the core (111)	Total	4170	16858	N.A.	503	3541	16797	61
	Average per element	37.57	151.87	15,1	4.53	31.90	151.32	0.55
Spent fuel removed from the core (12)	Total	449	1811	N.A.	121	297	1796	15
	Average per element	37.42	150.92	33.7	10.10	24.79	149.69	1.22

3.1. Activity planning

During the period in which the repatriation of US origin fuel was suspended, the DOE license to use the PEGASO cask as irradiated fuel transport container on US territory declined, and TRANSNUCLEAIRE (the French company responsible for the shipment) was not able to renew it, even considering the design modifications made to improve the cask performance. As a consequence, PEGASO cask loading procedures and devices (already available) had to be abandoned, and in 1998, TRANSNUCLEAIRE itself proposed to use the NAC-LWT shipping cask, specifically configured for TRIGA fuel, to ship RC-1 spent fuel elements to IFSF in Idaho Falls.

During a visit to RC-1, TRANSNUCLEAIRE and NAC technicians suggested to use the reactor shielding tank (see Fig. 1), which had been opportunely predisposed and licensed, as temporary storage of all irradiated fuel elements stored in the different reactor sites (reactor pool racks and storage pits) (see Figs. 2 and 3). The reasoning was that the loading operation could be a lot simplified if all fuel elements were available in the shielding tank, and the two tonnes NAC Intermediate Transfer System could be immersed in the tank to be loaded.

In January 1999, after a meeting with representatives from the Regulatory Body, RC-1 staff agreed to use RC-1 shielding tank as temporary storage of all irradiated fuel elements, to facilitate subsequent activities toward the shipment of the fuel to the USA.

3.2. Safety and radiological aspects

During the preparation phase of the spent fuel shipment operation a detailed analysis was carried out in which it was evident the necessity to make mechanical adaptations to RC-1 facility, in order to make it compatible for reception of NAC cask, utilization of TRIGA coupling devices, and radiological operative aspects.

The authorization issued by the Regulatory Body was subordinate to requests resumed mainly in the following points:

- Development of a quality assurance plan for construction and installation of fuel element storage racks, to be installed in the shielding tank;
- Seismic evaluations considering the site reference earthquake;
- Checking reactivity de-coupling between TRIGA reactor core and the 144 TRIGA fuel elements in the shielding tank racks (even considering that only 140 would be shipped, the racks installed in the shielding tank had 144 positions, so the calculations had to consider the possibility to have 144 fuel elements in the tank);
- Checking about radiological fitness of shielding tank water and concrete thickness;
- Cleaning and repainting (impermeabilization) of the internal surface of the shielding tank;
- Full revision of the reactor room bridge-crane system.

The RC-1 actions, in order to satisfy these requests were:

- Twelve racks in AISI 304L were built, each rack with 12 fuel element positions.
- Structural/seismic analysis was performed using SUPERSAP code, module SSAP0 according to ASME and ACI rules;
- Two independent calculations (by Oak Ridge SCALE 4.4 code and by MCNP.4B routine KCODE according with USNRC-SRP-9.1.2 and ANSI/ANS 8.1) were produced, in order to verify the subcriticality of 144 TRIGA fuel elements in the racks and verify reactor core and racks de-coupling;
- Concrete thickness of shielding tank is not the same in all radial positions. Therefore, to reduce γ exposure in the reactor room a concrete block wall was assembled all around the shielding tank, to assure, at least, in all radial directions, 90 cm of concrete thickness. This was sufficient to reduce γ exposure to negligible values (~1.5 μSv/h). About vertical water shielding thickness, irradiated fuel elements in the racks would be deeper than 2 m in the water (for a γ exposure of ~1.5 μSv/h) and no operations foreseen a water minimum layer lower than 1 m (for a γ exposure of ~0.5 mSv/h);
- Considering that the shielding tank internal surfaces presented diffused swellings on more external coat of epoxy paint. Therefore the reactor was stopped for two weeks, the demineralized water was pumped out from the shielding tank and the coat swellings were removed. After that, two epoxy paint coats were newly distributed in appropriate manners and times, floor included. Before repainting start, to reduce core exposure from the thermalizing

column, a lead bricks shielding (10 cm thick) was built inside of shielding tank, so the dose absorbed by TRIGA staff during the painting was negligible. At last, new demineralized water (~28 m3) was used to fill the tank.

- Regardless of the periodical maintenance programme established for the reactor room crane, an extraordinary maintenance programme was performed too. Braking system was checked and renewed.

As for the accident analysis, during shielding tank fuel elements storage and fuel handling, it was assumed, as conservative reference, the handling accident reported in the TRIGA RC-1 Safety Analysis Report.

Additional technical prescription was imposed by the Regulatory Body about racks, racks distance, shielding tank minimum water level, handling procedures and other minor issues.

The handling procedures and the working programme were completed with a detailed analysis of workers doses.

3.3. Preliminary operations

Spent fuel transfer from reactor storage facilities to the shielding tank required about 6 days and it was concluded by reactor staff in June the 30th. A loading plan was elaborated according to NAC cask loading procedures and the positioning of the fuel into the racks of the shielding tank was made accordingly. The cask loading plan was submitted and approved by Regulatory Body.

The preliminary road transport plan and NAC cask handling plan outside of the reactor room, was developed together with technicians from TRANSNUCLEAIRE and MIT Nucleare (Italian authorized shipper). Also these plans were submitted and approved by Regulatory Body.

3.4. Shipment cask and ancillary systems

- The NAC-LWT shipping and containment equipment for TRIGA fuel (showed in Figs. 7 and 8) is constituted by:
- Main CASK (Fig. 7-left): It is the real road container. In its cavity, five modular stainless steel baskets are stacked axially, each one of them able to accommodate up to 28 TRIGA fuel elements, for a total load of 140 TRIGA elements;
- BASKET (Fig. 7-right): It is a non-recovery unit because it is removed from the cask only for insertion into the final container, in USA, for permanent storage. The baskets can have three different lengths, to accommodate the 45 inches fuel follower control elements. Each basket has seven MTR fuel element positions and each one of them of them can accommodate 4 TRIGA type fuel elements, so the total capacity of the basket is 28 TRIGA type fuel elements.
- Intermediate Transfer System: I.T.S. (Fig. 8-left): It is a transfer assembly consisting of an inner and an outer shield. The inner shield has a maximum loaded weight of two tonnes, and it is provided with some holes in the bottom, to allow draining the water during the loading operation. The outer shield has a maximum weight of three tonnes (five tonnes when combined with the inner shield) and provides additional shielding for personnel during operations;
- Trolley (Fig. 8-centre): It is a transfer carriage for the I.T.S. to allow its transport to the cask loading location;
- Transfer cask (Fig. 8-right): It is a device designed to temporarily receive the basket from I.T.S. and to place it inside of the transport cask. It is provided with a lid and a pneumatic system for basket movement.

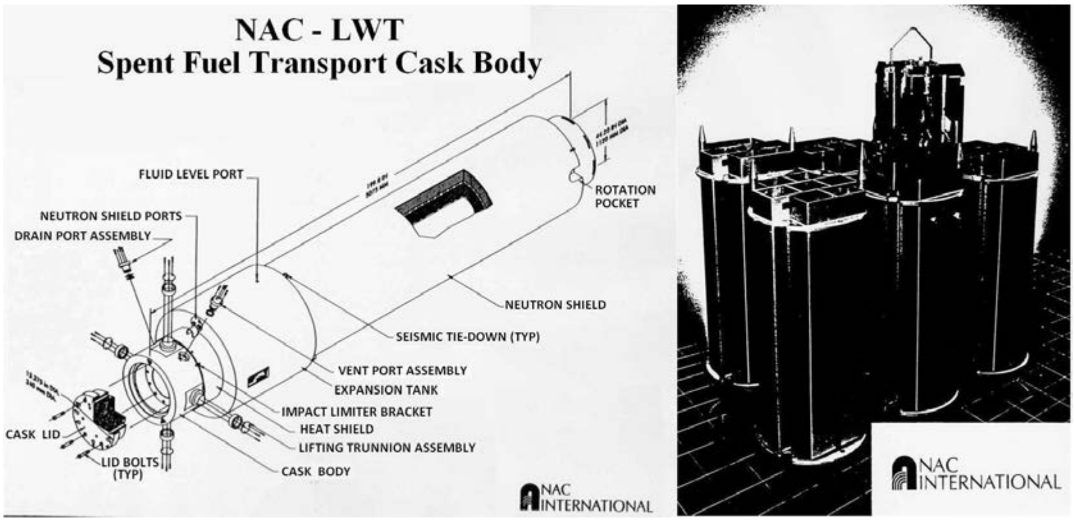

FIG. 7. NAC LWT transport cask (left) and fuel element basket (right).

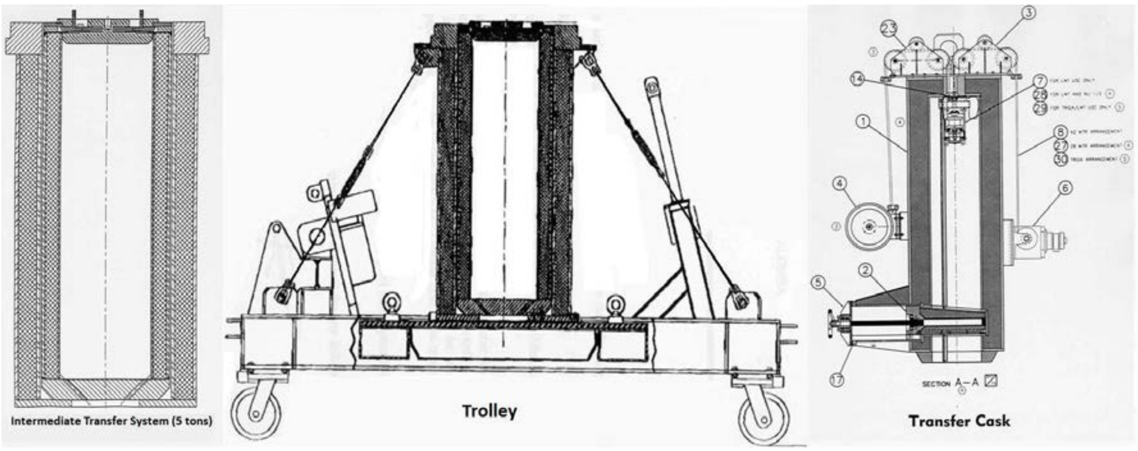

FIG. 8. NAC LWT cask loading system.

3.5. Synthetic description of loading operation

(1) The I.T.S., with an empty basket inside placed on the trolley, is introduced into the reactor room and placed as close as possible to the pool (shielding tank);

(2) The carriage door is closed to have reactor room in depression (ventilation plant is running);

(3) Using the reactor room crane, the I.T.S. inner shield, with empty basket inside, is immersed in the pool, and placed on its bottom;

(4) Under water and using the fuel handling tool, 28 fuel elements are transferred from the storage racks to the basket;

(5) After loading the I.T.S. a lid is placed on top of the inner shield, the I.T.S. is lifted from the pool and allowed to drain;

(6) Using the reactor room crane, the inner shield is placed inside of the outer shield on the trolley; the carriage door is opened, and, the trolley is removed from the reactor room to an area close to the transport cask;

(7) A coupling shielding plug is placed on top of the I.T.S. the lid is removed;

(8) With the help of a tractor crane the transfer cask is placed on top of the coupling shielding;

(9) The gate of the transfer cask and the gate of the coupling shielding plug are opened, the basket with the fuel elements is clasped using a pneumatic device on the transfer cask and dragged inside;

(10) The transfer cask's gate in closed;

(11) Using the tractor crane, the transfer cask is coupled with the cask (on which a second coupling shielding plug is present in closed position);

(12) The gate of the second coupling shielding plug and the gate of the transfer cask are opened;

(13) The basket with the fuel elements is lowered inside of the cask;

(14) The gate of the coupling shielding plug gate is closed again;

(15) Operations from 1 to 14 are repeated 4 additional times, until loading operation is completed.

(16) Cask is closed and after its radiological survey, surface contamination measurements, and leak test, it is placed on the truck using the tractor crane.

(17) All other devices are radiation surveyed and placed on the respective road transport means;

(18) Convoy leaves Casaccia Center directly to Koper, a port in Slovenia.

The TRIGA element loading operations, yard installation and removal, was done in about five days, finishing on 23 July 23 1999.

4. CONCLUSION

As described, the management of spent fuel in research reactor RC-1 has not created too many problems and the fuel storage facilities showed to be suitable and easy to operate.

The only remarkable incidents that occurred, without any consequences, were a fuel element fall occurred during the first years of operation and the corrosion of the aluminium cladding. The first one was resolved by modifying the handling procedure and the second one was not analysed in depth because RC-1 does not use anymore aluminium clad elements. Probably the corrosion occurred because some debris of carbon steel felt into the water of the pits during the openings of the lids.

The TRIGA RC-1 reactor is still operational and ENEA´s wishes to maintain it in operation for many years yet, especially considering the eventual return of Italy to use nuclear power plants in order to produce electrical energy: nowadays there are only three nuclear research reactors in operation in Italy and RC-1 is one of them. It could constitute an excellent starting point to contribute to the reconstruction of "nuclear culture" for technicians and engineers know-how.

Currently the reactor is not in operation and it needs some important maintenance interventions, which have already been scheduled. In this perspective it can be hypothesized a fuel need of 2÷4 standard fuel elements per year. This does not constitute a problem for interim storage of spent fuel into the RC-1 storage facilities for many years, but could be a problem if we think about the final disposal of them. Nowadays Italy does not have a waste repository and the return of the TRIGA fuel to the USA, if the return policy is extended, could again be the best solution.

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STUDY OF TIGHTNESS OF SPENT FUEL ASSEMBLIES IN STORAGE OF WWR-SM RESEARCH REACTOR[35]

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Abstract

WWR-SM is a Russian type research reactor located in Tashkent, at the Institute of Nuclear Physics of Uzbekistan Academy of Sciences. The reactor reached criticality for the first time in October of 1959, using 10% enriched uranium fuel assemblies (FA), and after a modernization process completed in 1979, the reactor started using 90% enriched FAs. The conversion of the reactor from Highly Enriched Uranium (HEU) to Low Enriched Uranium (LEU) fuel started in 1998, when the reactor was fuelled with 36% enriched FAs. In order to completely convert WWR-SM reactor to LEU fuel, in 2008 the reactor started being fuelled with 19.7% enriched fuel. To confirm the integrity of the new fuel assemblies, and to ensure that conditions of interim storage pool are not compromised, a programme was established to test the new fuel in two distinct phases: during its life in the core, and after final discharge. The report describes the tests performed and the respective results.

1. THE WWR-SM RESEARCH REACTOR

WWR-SM is a Russian type water cooled, beryllium reflected and water moderated research reactor located at Tashkent, in the Institute of Nuclear Physics of Uzbekistan Academy of Sciences. The reactor, originally rated at 2 MW, reached criticality for the first time in October of 1959, and after a modernization process, between 1971 and 1979, the reactor had its nominal power increased to 10MW. The reactor has a cylindrical core about 60 cm height and 64 cm in diameter, beryllium reflected. Three rods are used for safety purpose, six rods are used as shim rods, and one is used for automatic control of the reactor. All the ten rods are B_4C [1].

2. THE FUEL USED IN WWR-SM

Several types of fuel assemblies have been used in the core of the reactor. In the early years the reactor was fuelled with 10% enriched EK-10 fuel assemblies, an assembly consisting of 16 rods of uranium dioxide-magnesium alloy in aluminium casing, as shown in Fig. 1. In 1971 the reactor started using IRT-2M fuels, an assembly comprised of a set of concentric square rings of U-Al alloy, 90% enriched, as illustrated in Fig. 2. After the process of modernization and power increase, the reactor was loaded with new advanced IRT-3M fuel elements, an assembly similar to IRT-2M, with more concentric square rings. IRT-3M, also a 90% enriched fuel assembly, has two versions, six and eight concentric tubes, as illustrated in Fig. 3. After 1979 both versions of the 90% enriched IRT-3M fuel assemblies were used in the core of the WWR-SM reactor [1].

Between March 1987 and March 1989, three (3) 36 % enriched IRT-3M fuel assemblies (two 6-tube and one 8-tube) were successfully tested in the WWR-SM reactor, and in August 1998 four IRT-3M fuel assemblies were loaded into the core, initiating the full conversion of the WWR-SM reactor core to the new 36% enriched fuel. The main characteristics of the new version of the 6 tube IRT-3M fuel assemblies were: 36% enriched UO_2-Al alloy meat with Al cladding, meat thickness 0.5 mm; minimum cladding thickness equal to 0.3 mm, and uranium density 2.51 g/cm^3 [2]. The dimensions of the assembly were kept the same; 880 mm total length with an active length equal to 600 mm.

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[35] This report contains pos meeting data that was obtained during the work done to qualify the new LEU enriched fuel used in the WWR-SM research reactor. It was agreed to use the pos meeting data in order to make it available for research reactor managers who consider using the same type of test during conversion programs.

The full conversion of the reactor was completed in February 1999 [1, 3].

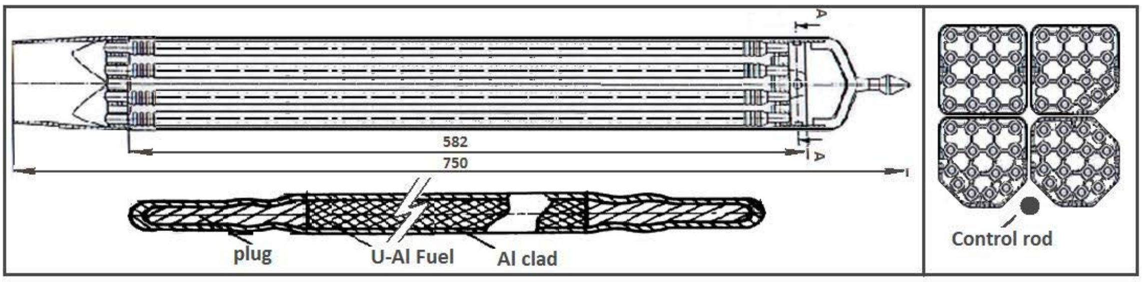

FIG. 1. EK-10 fuel assembly (left), and detail of the core configuration (right).

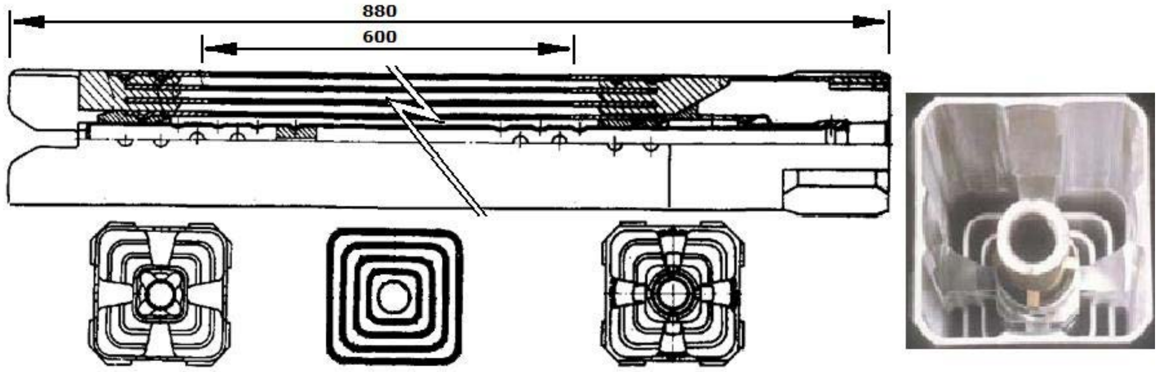

FIG. 2. IRT-2M fuel assembly.

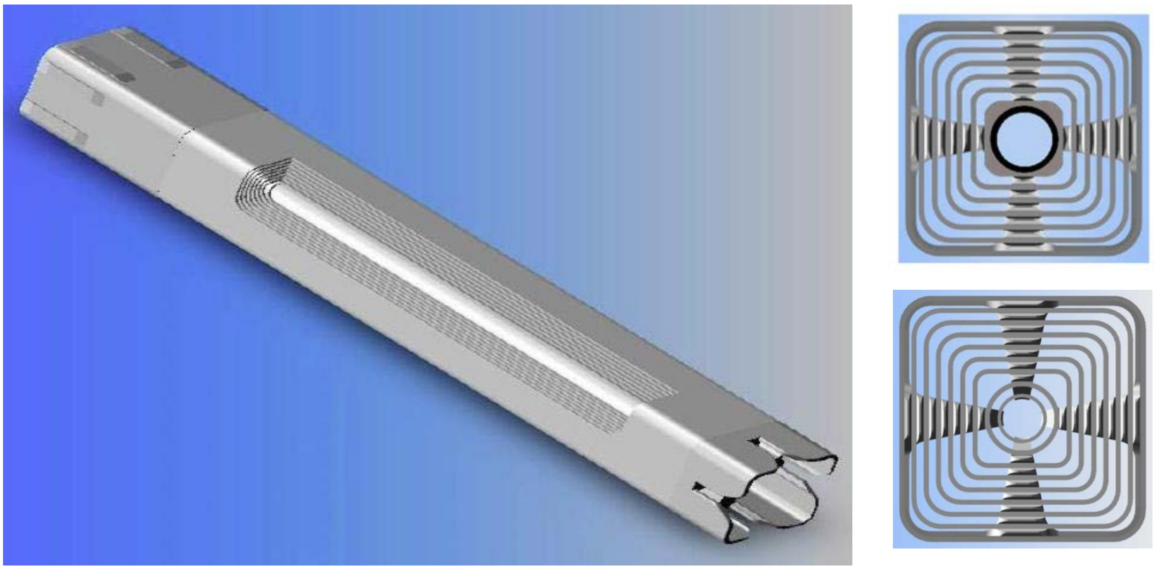

FIG. 3. IRT-3M fuel assembly with 6 (top right) and 8 (bottom right) concentric tubes.

According to the Russian RERTR programme[36], in 1995 the "Novosibirsk Chemical Concentrates Plant" (NZHK) started the development of a new family of fuel assemblies the 19.7 % enriched

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[36] In 1978 the former USSR Ministry of Medium Engineering (currently Minatom of Russia) passed a resolution prohibiting to supply uranium with enrichment over 21% after renovation and modernization of research reactors constructed abroad with technical assistance from the former USSR. In the case that the supply of fuel elements and assemblies with enrichment higher than 21 % was absolutely necessary for operation of a specific research reactor, a special permission of the Ministry should be required [7].

IRT-4M. After some preliminary tests in the IR-8 research reactor at Kurchatov Institute, in 1999 NZHK fabricated four experimental IRT-4M assemblies: two with eight tubes, and two with six tubes. In 2000 these fuel assemblies, made with UO_2-Al and uranium density equal to 3.0 gU/cm^3, were transferred to Tashkent to be tested in the core of the WWR-SM research reactor [4]. Testing of the IRT-4M type FAs started in November 2000 and ended in March 2002, after sixteen test cycles. The average fuel burnup reached 66.5%, and the maximum (localized) burnup was 92.6%. Results from continuously monitoring of gas under the reactor cover and daily monitoring of water activity in the primary circuit demonstrated that there were no failures of the IRT-4M FAs during testing [4].

After a detailed analysis of the test results, in March 2008 the WWR-SM reactor started gradually being converted to use the new LEU 19.7% enriched fuel. The full conversion process has been planned to last nine operating cycles, with expansion of the reactor core from the previous 18 IRT-3M fuel assemblies configuration to a new configuration with 20 IRT-4M fuel assemblies [5].

3. STORAGE OF WWR-SM NUCLEAR FUEL

At the WWR-SM, the storage of fresh fuel is in airtight containers in a dry locked room where the presence of water is excluded. The distribution of the fuel assemblies on the shelves ensures minimum subcritical condition of not less than 0.05 for all possibilities, even when the complete flooding of the room with water is taken into consideration.

The facility has two original spent fuel storage ponds (FSP), and one additional storage area with four storage tanks. The storage ponds and tanks are made of stainless steel, and covered with steel deck plates. FSP number one has 60 cells and FSP number two has 192 cells, two racks with 96 cells each that can be placed one on top of the other. The additional storage area was added to the facility in 2000, to comply with the necessity to increase the facility storage capacity, after the decision to stop shipping spent nuclear fuel to the Russian Federation in 1992 was made (see Section 4). This additional storage area has four storage tanks. Tanks one and four have 44 cells each, and tanks two and three have 37 cells each. In total the facility can store up to 414 spent fuel assemblies.

Figure 4 shows the configuration of the racks in the storage ponds, and the configuration of tank three in the additional storage area.

In 2005 there were 370 fuel assemblies at the reactor site, 28 fresh, 18 in core, 159 in the two Fuel Storage Ponds (FSP) {33 in FSP 1 (29 36% and 4 19.8%) and 126 in FSP 2 (48 90%, 67 36% and 11 10%)}, and 162 in the tanks of the additional storage area, all 90% enriched [6].

Additional information on the conditions of the reactor pool and spent fuel storage ponds is given in Section 5.

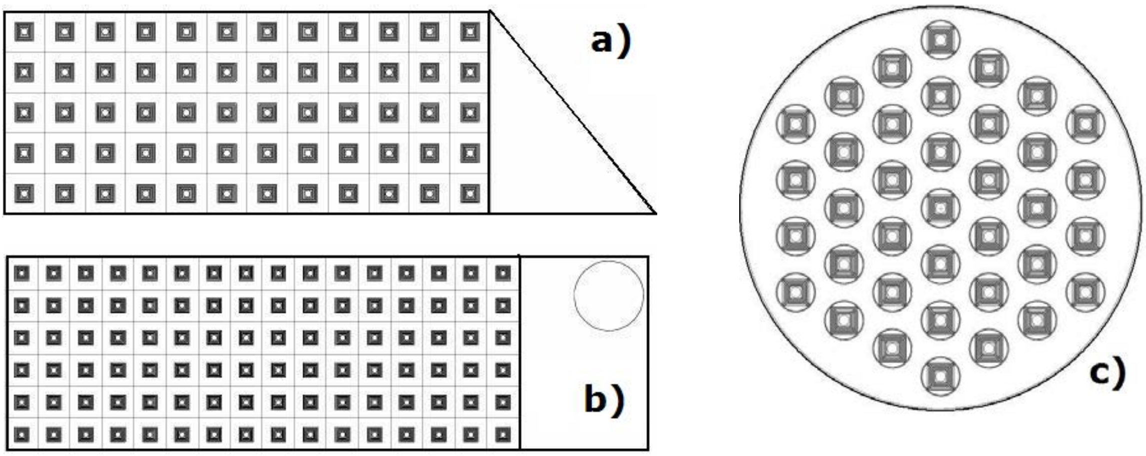

FIG. 4. Rack configuration of FSP 1(a); FSP 2 (b) and tank three in the additional storage area (c). FSP two can accommodate two racks (each one with 96 positions), one on top of the other.

4. RETURN OF WWR-SM SPENT NUCLEAR FUEL TO THE RUSSIAN FEDERATION

Between 1973 and 1992, after initial cooling in the spent fuel storage pools available at the facility, the spent fuel from WWR-SM research reactor was sent to Mayak reprocessing facility, in the Russian Federation. The spent fuel was shipped to the Russian Federation in tranches, in the 70's every four years (1973 and 1977), and during the 80's every two years (1981, 1983, 1985, 1987, 1989 and 1991). In total 440 fuel assemblies were shipped comprising 128 EK-10, 90 IRT-2M and 222 IRT-3M [6].

After the dissolution of the Soviet Union, the parliament of the Russian Federation passed a law forbidding the return to the Russian Federation of any radioactive waste. This resulted in the necessity to store more than 300 highly (90% and 36%) enriched spent fuel assemblies. The situation created serious threats to radiation safety as well as to nuclear security, and through the initiatives of the IAEA, USA, and the Russian Federation, a new Russian Research Reactor Fuel Return (RRRFR) programme was established in December 1999. This programme allows for the return of highly enriched spent fuel from 24 research reactors at 17 facilities in 15 countries, including Uzbekistan WWR-SM, to the Russian Federation [1].

Following the establishment of the RRRFR programme, an agreement was signed between the Government of Uzbekistan and the US Department of Energy (USDOE) in March 2002, and INP began the initial planning for the return of the SNF from its WWR-SM reactor to the Russian Federation. The progress of the project in the beginning was very slow. It gained speed in May 2004, after signature of an *Agreement between the Government of the United States of America and the Government of the Russian Federation concerning cooperation for the transfer of Russian produced Research Reactor Nuclear Fuel to the Russian Federation,* which gave the project the legal basis to proceed [8]. As part of the agreement, twenty three (23) fresh fuel assemblies, containing about 10 Kg of uranium, of which about 1.75 Kg were fissile ^{235}U, returned to the Russian Federation in August 2004, to be down blended [9].

In January 2006, Uzbekistan became the first country in the last fifteen years to return spent nuclear fuel (SNF) to the Russian Federation and the first under the new RRRFR Programme. A total of 252 IRT-3M spent fuel assemblies were shipped, 210 with 90% enrichment, and 42 with 36% enrichment. Several months prior to the shipment, the fuel assemblies were inspected by experts from Mayak. All the assemblies were intact, and met the acceptance criteria for shipment and receipt without requiring encapsulation. The TUK-19 cask was chosen because it was designed for Russian research reactor fuel and for use in the Russian Federation. The TUK-19 has a capacity to hold 4 IRT-3M assemblies and a total of 16 casks were available for each shipment. The casks were transported to Mayak by rail in two TK-5 railcars, each one with eight TUK-19 casks. With a maximum of 64 IRT-3M fuel assemblies transported in each shipment, four shipments were needed to return the 252 spent fuel assemblies to Mayak, as shown in Table I [10].

TABLE I. THE 2006 SHIPMENT SCHEDULE [10]

Date	No of FA	No of containers	Enrichment (%)	No of FA
10.01.06	64	16	90	64
14.02.06	64	16	90 / 36	57 / 7
07.03.06	64	16	90 / 36	57 / 7
15.04.06	60	15	90 / 36	32 / 28

5. WWR-SM REACTOR POOL WATER CONDITIONS

The core of WWR-SM research reactor is beryllium reflected and light water cooled and moderated. To comply with the requirements to protect the fuel integrity the pool uses high quality demineralized water.

Water-chemical conditions of the water in the reactor pool must provide:

- Conditions to avoid sediments on the cladding (heat exchange surface) of the fuel element;
- Enough fluid velocity to remove the produced heat, without causing erosion or corrosion of construction materials.

To comply with these requirements, a water quality control programme is necessary, to maintain the quality of the water in the reactor pool within established parameters. This programme involves:

- Establishment of quality parameters for the water in reactor pool and primary cooling system;
- Establishment of quality parameters for pool water supply (replenishment);
- Requirements for water-chemical maintenance;
- Conditions for pool water chemical control (purification).

There is permanent control of water quality in all pools and tanks, with periodic measurements of such parameters as pH (normally pH is in the interval 5.5–6.5), electric conductivity (less than 4 µSm/cm), specific activity and the content of Al^{+3} ions.

5.1. Pool water physical-chemical parameters

During reactor start up, the physical-chemical parameters of the cooling water must be controlled within the limits established in internal document OST 95 10134-91 “Regulatory document of water quality control” inv. № 1/03–01 2003 [11] which are summarized in Table II.

The following items are used to maintain the quality of the water in the reactor pool:

- Pool water purification system;
- Pool water supply system (replenishing);
- Circulation pumps;
- Autonomous pump, whenever it is necessary.

In addition to the components and system available, a permanent programme requires sampling the water from the primary cooling system, to confirm compliance of parameters with values listed on table I. Once it is detected that maximum tolerable values are being reached, a maintenance activity is planned for the pool water purification system. Coolant sampling for determination of micro-admixtures is made before and after ion-exchange filter. The exact sampling location is the CMD (control and measuring devices) safety box, premise 17^{A}. Chemical analysis is carried out by laboratory analysis methods.

TABLE II. PHYSICAL-CHEMICAL PARAMETERS FOR WATER IN REACTOR POOL AND IN PRIMARY COOLING SYSTEM

Parameter	Value at normal operation	Maximum tolerable deviation	Allowed discrepancy of measuring method %
pH at 25 ºC	5,5–6,5	$5,0 \le pH \le 6,8$	± 1
Specific electric conductivity at 25 ºC (µSm/cm)	≤ 4,0	__	± 1
Hardness (mg-eq/kg)	≤ 3,0	__	± 10
Chloride-ion concentration (ppm)	≤ 50,0	$50 < Cl^{-} < 150$	± 2
Aluminium concentration (ppm)	≤ 50,0	__	± 2
Iron concentration (ppm)	≤ 50,0	__	± 2
Copper concentration (ppm)	≤ 10,0	__	± 2
Total specific activity by ^{235}U decay products (Ci/l)	$< 1 \cdot 10^{-4}$	$< 1 \cdot 10^{-4}$	± 2

For the pool water supply system (for the first pool filling and for replenishment), the water must comply with the conditions shown in Table III.

TABLE III. PHYSICAL-CHEMICAL PARAMETERS FOR THE POOL WATER SUPPLY SYSTEM (*primary filling and replenishment*)

Parameter	Accepted value	Allowed discrepancy, %
pH at 25 ºC	5,5–6,5	± 1
Specific electric conductivity at 25 ºC (μSm/cm)	≤ 1,5	± 1
Hardness (mg-eq/kg)	≤ 1,0	± 2
Chlorides (ppm)	≤ 20,0	± 2
Iron corrosion products in Fe terms (mkg/kg)	≤ 20,0	± 2

6. THE FUEL CLADDING TIGHTNESS VERIFICATION SYSTEM (FCTVS)

Given the importance of establishing the quality of the new LEU fuel assemblies, a complementary activity developed by the staff of INP was the construction of a system to verify and confirm the leak tightness of the new fuel assemblies. With this purpose, a fuel cladding tightness verification system (FCTVS) was developed, and assembled in the spent fuel storage pool [12]. Figure 5 presents the schematic of the FCTVS. The system was used to test the fuel assembly during the operational cycle in the reactor core and after final removal from the reactor core, in order to avoid eventual degradation of the water in reactor and storage pools.

Using Fig. 5 as a reference, the working procedure is as follows: after placing the FA in the infusion box (1), the box is sealed using a gasket (2) and a screwed box cap (3), and clean demineralized water (from 4) is added for around 30 minutes at a flow rate equal to 3 litres/minute. Around 90 litres of water is used, with 70 litres directed to the storage pool, through valve 16, and 20 litres kept in the system. The system is then closed for a period of 6–24 hours**.** Afterwards, by opening valves (6), (11) and (13), water is ejected to the “pre-vacuumized” tank (14). When tank 14 is filled with water (tank 14 has a volume equal to the volume of tank 7 and tubes after tank 7), then by opening valve (10) a sample of water is collected in tank (9) for laboratory analysis. Spectrometric analysis of the sampled water is performed using the γ–sensor (8). Drawing water from the pool is prevented by checking water levels in the transparent glass tube gauges installed in tanks (12) and (4).

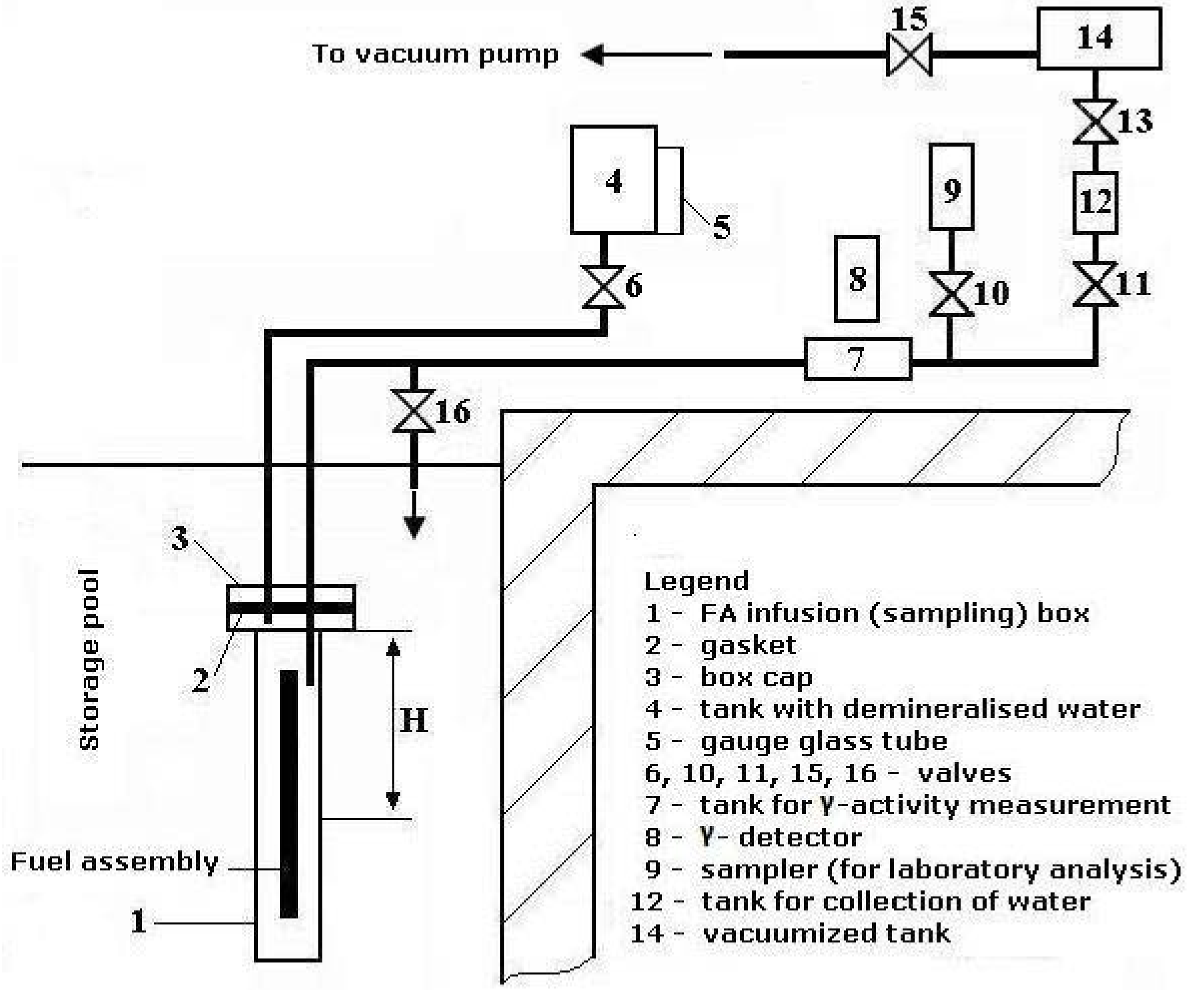

FIG. 5. Diagram of fuel cladding tightness verification system (FCTVS) installed in spent fuel storage pool.

One important activity performed using the FCTVS, is the determination of cladding tightness of the fuel elements. The test can be done on single fuel assemblies, during the intervals between operating cycles and when the fuel is unloaded from the reactor core. The test is carried out by simply placing the FA in the infusion (test) box followed by spectrometric measurements of samples of the "infused" water samples. The test allows for quality control of the new nuclear fuel assemblies. If any manufacturing flaws are revealed, they will be unconditionally returned to the manufacturer and replaced with new FAs at no extra cost. The test also allows early determination of leaking assemblies, so that they can be segregated, to avoid contamination of the water in the storage pool.

6.1. Measurement of activity in water from primary cooling system and FCTVS

During the first phase of the conversion process, when some of the new FAs were loaded into the reactor core, water samples from the reactor pool and primary cooling system were analysed, during reactor operation and after its shutdown. Water samples were also taken, for analysis, from the storage pool 2; from tanks in the additional storage area, and from the FCTVS stand box while performing leak tightness tests control of spent FAs.

The volume of each sample taken for laboratory analyses was 5 ml. Before gamma spectrum analysis, the samples were left to decay for one hour, one, three, five, seven and ten days to remove short lived nuclides, with the γ spectrum being measured after each period. In the reactor pool, a reference level was established to identify a possible "unusual" condition caused by the existence of a leaking fuel assembly. This reference level was established as the total activity resulting from the uranium surface contamination obtained with the 18 FAs loaded into the core of the reactor. The acceptable result was a mass of uranium equivalent to 10^{-8} g/cm^2 per fuel assembly. Based on the FA area of 1.37 m^2, it was established that the maximum mass of uranium allowable in the reactor coolant system could be 2.466×10^{-4} g of uranium, which corresponds to a concentration of 1.76×10^{-8} g/l [13]. Calculated activities A_i and specific activities A_s of the decay products, resulting from the irradiation of 2.466×10^{-4} g of uranium with thermal neutron flux equal to 2×10^{14} cm^{-2}s^{-1} during seven days are shown in Table IV.

Based on the analysis performed an upper limit of 3.7E+06 Bq/l was established as the measured total specific activity in water for the new fuel to be accepted as “fully integral and clad tightness approved”. If this limit is exceeded, then the new developed LEU fuel should not be accepted as the new standard fuel for use in the core of WWR-SM research reactor. If the value is between 3.7E+06 and 3.00E+05 Bq/l then a more detailed examination is necessary, including visual examinations using an underwater camera.

TABLE IV. CALCULATED TOTAL (A_I) AND SPECIFIC (A_S) ACTIVITIES OF URANIUM DECAY PRODUCTS IN THE COOLANT, AS A RESULT OF MAXIMUM FA SURFACE CONTAMINATION[a].

Nuclide	A_i (Bq)	A_s (Bq/l)	Nuclide	A_i (Bq)	A_s (Bq/l)
^{88}Kr	$3.1\cdot10^9$	$221\cdot10^3$	^{135}I	$13.5\cdot10^9$	$962\cdot10^3$
^{88}Rb	$3.1\cdot10^9$	$221\cdot10^3$	^{135}Xe	$13.9\cdot10^9$	$994\cdot10^3$
^{99}Mo	$8.5\cdot10^9$	$606\cdot10^3$	^{137}Cs	$12.1\cdot10^6$	862
^{103}Rh	$0.6\cdot10^9$	$42\cdot10^3$	^{138}Cs	$13.2\cdot10^9$	$947\cdot10^3$
^{129m}Te	$2.6\cdot10^7$	$2\cdot10^3$	^{141}Ba	$13.2\cdot10^9$	$947\cdot10^3$
^{130}I	$4.4\cdot10^9$	$315\cdot10^3$	^{141}Ce	$4.0\cdot10^9$	$288\cdot10^3$
^{131}I	$3.4\cdot10^9$	$243\cdot10^3$	^{143}Ce	$12.6\cdot10^9$	$899\cdot10^3$
^{132}I	$8.8\cdot10^9$	$631\cdot10^3$	^{147}Pm	$7.0\cdot10^7$	$5\cdot10^3$
^{133}I	$15.2\cdot10^9$	$1088\cdot10^3$	^{155}Eu	$1.5\cdot10^6$	$0.1\cdot10^3$
^{134}I	$17.2\cdot10^9$	$1231\cdot10^3$	^{156}Eu	$1.7\cdot10^7$	$1.2\cdot10^3$

[a] Activities considering 2.466×10^{-4} g of uranium irradiated with a thermal neutron flux of 2×10^{14} cm^{-2}s^{-1} during seven days.

Considering that most of the γ-lines of ^{235}U are in the soft (background) region of the spectrum, to decrease background count of Bremstrahlung lines (E<100 keV), which prevail in an unfiltered spectrum of any sample, and to account for errors caused by high detector loading and spectrum distortion, filters were used almost in all measurements. The thickness of the filters were chosen so as to decrease the amplitude of the peaks with energies less than 70 keV, with a tolerance to lower amplitude of peaks below 100 keV. In the great majority of samples taken, gamma radiation with energies up to 100 keV were quite considerable, and that is why a thin foil of cadmium was used as filter. The influence of the cadmium filter on the activity measurement of a 2g powdery sample of $UO_2(CH_3COO)_2\ 2H_2O$ is shown in Table V. Without filter gamma quants with energy of E<100 keV prevail in the measured spectrum, which can cause pile-up and saturate the detector. Cadmium filters with thickness of 1 to 2 mm essentially weaken those lines and only slightly weaken lines of higher energies, particularly 185.7 keV line ^{235}U. The spectrum of decay products in the energy range lower than 250 keV was measured using the 1 mm cadmium filter. Only when gammas with energies higher than 400 keV were considered, then lead was used as filter material, to attenuate gammas with energies in the range of 100 to 200 keV and to absorb most gammas with energy below 60 keV. The results obtained, are shown in Table VI.

TABLE V. INFLUENCE OF CADMIUM FILTER ON LOW ENERGY GAMMA COUNTING FROM A 2 g SAMPLE OF $UO_2(CH_3COO)_2\ 2H_2O$

Thickness of filter (mm)	Gamma energy			
	60 кeV	129 кeV	186 кeV	414 кeV
0.0	$3,57\times10^{6}$	$1,29\times10^{4}$	$8,50\times10^{4}$	$2,02\times10^{4}$
1.0	$2,40\times10^{4}$	$0,67\times10^{4}$	$6,76\times10^{4}$	$1,85\times10^{4}$
2.0	$1,86\times10^{2}$	$0,34\times10^{4}$	5.37×10^{4}	$1,69\times10^{4}$

TABLE VI. RADIONUCLIDE ACTIVITY IN WATER FROM THE PRIMARY COOLING SYSTEM

Nuclide	Activity (kBq)		Weight content (g)	
	10 MW	11 MW[b)]	10 MW	11 MW[b)]
^{24}Na	74.0	222.0	$0,3\times10^{-2}$	0.4×10^{-2}
^{88}Rb	59.2	236.8	0.3×10^{-3}	0.1×10^{-2}
^{99m}Tc	74.0	162.8	0.2×10^{-3}	0.4×10^{-3}
^{131}I	99.9	310.8	0.2×10^{-3}	0.1×10^{-2}
^{132}I	14.8	40.7	0.9×10^{-3}	0.1×10^{-2}
^{133}I	3.7	185.0	0.4×10^{-3}	0.1×10^{-2}
^{134}I	11.1	37.0	0.8×10^{-3}	0.5×10^{-2}
^{135}I	3.7	140.6	0.1×10^{-4}	0.4×10^{-3}
^{137}Cs	22.2	74.0	0.1×10^{-3}	0.1×10^{-2}
^{140}Ba	11.1	40.7	0.4×10^{-3}	0.2×10^{-2}
^{239}Np	70.3	148.0	0.6×10^{-3}	0.1×10^{-2}

[b)] Under special conditions the reactor can operate at 11 MW.

6.2. Results of fission products activity measurements

Specific leak tightness measurements were performed on the following 12 IRT-4M fuel assemblies (FA) with initial burnups shown in parentheses (in %): 419 (5.94), 417 (7.42), 415 (10.53), 414 (14.93), 411 (16.38), 413 (16.71), 410 (21.48), 408 (26.37), 405 (38.05), 403 (38.42), 401 (40.82) 399 (46.42). In order to consider all possible variables, air activity in FCTVS, air of the reactor upper space and primary system water activities were also measured.

A preliminary analysis of the results, shown in Table VII and in Fig. 6 indicates that the total activity and fission products specific activity increase with time, suggesting some correlation with the increase of FA burnup, however, a more detailed analysis is necessary before any final conclusion can be made.

For some fuel assemblies the leak tightness test was made a few times while their burnups increased. The results are shown in Fig. 7. As before they indicate a total activity increase with time, suggesting some correlation with the increase of FA burnup, but again, a more detailed analysis is necessary before any final conclusion can be made.

TABLE VII. SPECIFIC ACTIVITY MEASURED FOR VARIOUS RADIOISOTOPES[c] IN SOME PARTIALLY BURNED FUEL ASSEMBLIES (Bq/l)

FA No.	419	417	415	414	411	413	410	408	405	403	401	399
Burnup	5,94	7,42	10,53	14,93	16,38	16,709	21,48	26,37	38,05	38,42	40,82	46,42
FCTVS date	02/04 2010	21/01 2010	29/12 2009	21/01 2010	02/04 2010	04/02 2010	29/12 2009	21/01 2010	21/01 2010	29/12 2009	29/12 2009	04/02 2010
^{95}Nb	9,3E+01	5,5E+02	6,1E+02	9,2E+02	2,4E+03	1,2E+03	3,9E+03	4,3E+03	1,2E+04	1,1E+04	1,9E+04	2,1E+04
^{95}Zr	0,0E+00	0,0E+00	9,8E+01	1,2E+02	1,0E+02	3,2E+02	3,2E+02	4,0E+02	6,3E+03	5,3E+03	8,2E+03	1,3E+04
^{99}Mo	0,0E+00	0,0E+00	0,0E+00	0,0E+00	1,9E+01	0,0E+00	0,0E+00	0,0E+00	3,1E+03	1,8E+04	3,5E+04	4,6E+04
^{103}Ru	6,3E+02	9,2E+02	3,9E+02	5,0E+02	7,6E+02	6,1E+02	3,8E+02	4,3E+03	2,3E+04	1,6E+04	3,1E+04	3,3E+04
^{106}Ru	0,0E+00	0,0E+00	0,0E+00	0,0E+00	0,0E+00	0,0E+00	0,0E+00	0,0E+00	7,6E+03	9,0E+03	1,8E+04	3,0E+04
^{131}I	1,2E+03	1,8E+03	4,8E+03	6,3E+03	1,1E+04	1,1E+04	2,4E+04	2,2E+04	3,5E+04	6,3E+04	1,1E+05	1,6E+05
^{133}I	0,0E+00	4,2E+02	2,4E+03	5,5E+03	2,3E+03	1,1E+03	2,7E+04	1,8E+04	6,3E+04	5,4E+04	1,1E+05	1,8E+05
^{134}Cs	2,4E+02	2,4E+02	4,3E+02	6,2E+02	2,6E+03	2,6E+03	1,7E+03	2,4E+03	9,2E+03	4,9E+03	2,4E+04	2,9E+04
^{137}Cs	3,1E+02	3,1E+02	1,5E+02	3,1E+03	2,4E+03	3,9E+03	3,1E+03	5,5E+03	3,1E+04	1,7E+04	3,8E+04	4,7E+04
^{140}La	3,1E+02	4,6E+02	1,1E+03	1,2E+03	1,4E+03	2,2E+03	2,4E+04	2,6E+04	4,2E+04	6,1E+04	8,3E+04	1,6E+05
^{141}Ce	1,6E+02	1,9E+02	2,4E+03	3,3E+02	3,4E+02	6,1E+03	3,8E+02	4,6E+03	1,4E+04	1,7E+04	3,0E+04	3,5E+04
^{144}Ce	0,0E+00	0,0E+00	3,2E+02	8,5E+02	4,2E+03	1,6E+03	6,0E+03	6,7E+03	3,4E+04	3,8E+04	5,3E+04	6,0E+04
Total activity	2,9E+03	4,9E+03	1,3E+04	1,9E+04	2,7E+04	3,1E+04	9,1E+04	9,4E+04	2,8E+05	3,1E+05	5,7E+05	8,1E+05

[c] Measurements of activity for ^{109}Cd, ^{134}I, ^{135}Xe, ^{138}Xe, ^{139}Ce and ^{140}Ba, resulted 0.00E+00 for all FAs.

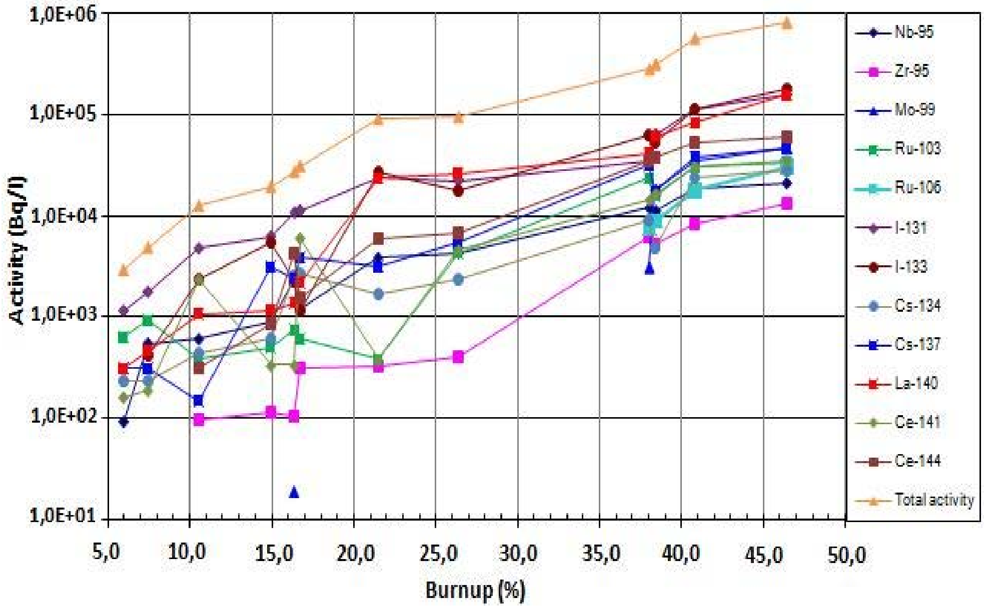

FIG. 6. Water activity measured as a function of FA burnup (built based on data of Table VII).

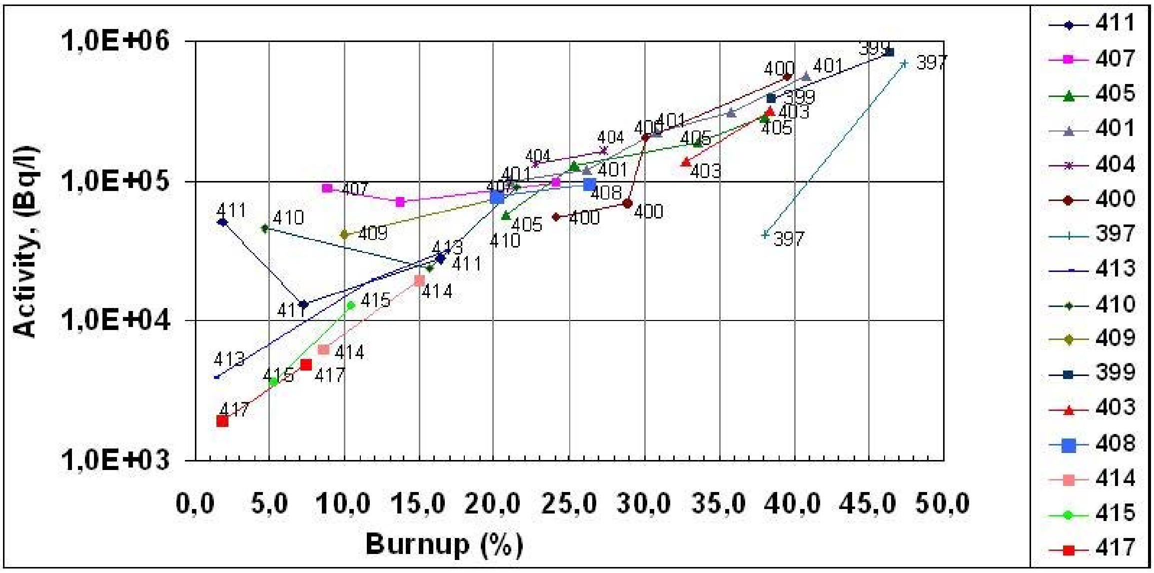

FIG. 7. Water activity for new LEU FAs as a function of burnup level.

To complement the analysis, the activity of the air collected from the top of the sampling box (total and specific for some radioisotopes) was also measured for several FAs with different burnups. The results are shown in Table VIII and in Figures 8 and 9.

The upper limit for the total air activity was established as 2.77E+04 Bq/l. If this limit is exceeded, then the new developed fuel is deemed not accepted for use in the WWR-SM reactor. If the value is between 2.77E+03 and 2.77E+04 Bq/l then the FA is submitted to a more detailed examination.

TABLE VIII. ACTIVITY OF AIR IN SAMPLING BOX FOR SEVERAL FUEL ASSEMBLIES WITH DIFFERENT BURNUP LEVELS (Bq/l)

FA No.	419	417	415	414	411	413	410	408	405	403	401	399
Burnup	5,94	7,42	10,53	14,93	16,38	16,709	21,48	26,37	38,05	38,42	40,82	46,42
FCTVS date	02/04 2010	21/01 2010	29/12 2009	21/01 2010	02/04 2010	04/02 2010	29/12 2009	21/01 2010	21/01 2010	29/12 2009	29/12 2009	04/02 2010
^{85M}Kr	1,3E+00	1,5E+00	2,0E+01	2,3E+01	2,9E+01	3,2E+01	3,4E+01	1,4E+01	2,4E+02	1,9E+02	2,7E+02	3,1E+02
^{89}Kr	5,0E-01	5,3E-01	2,8E+01	3,5E+01	6,8E+01	8,2E+01	1,2E+02	1,7E+02	2,4E+02	2,3E+02	2,9E+02	3,0E+02
^{135}Xe	7,0E+01	7,9E+01	2,3E+02	2,6E+02	3,5E+02	3,0E+02	3,4E+02	5,4E+02	1,7E+03	1,7E+03	2,3E+03	2,8E+03
^{137}Cs	1,9E-01	1,3E-01	5,4E-02	4,6E-01	7,9E-02	1,2E-01	1,6E+00	2,3E+01	3,1E+00	3,2E+00	3,7E+00	5,9E+00
^{137}Cs	1,9E-01	1,3E-01	5,4E-02	4,6E-01	7,9E-02	1,2E-01	1,6E+00	2,3E+01	3,1E+00	3,2E+00	3,7E+00	5,9E+00
Total activity	7,2E+01	8,2E+01	2,7E+02	3,2E+02	4,5E+02	4,1E+02	4,9E+02	7,4E+02	2,2E+03	2,1E+03	2,9E+03	3,4E+03

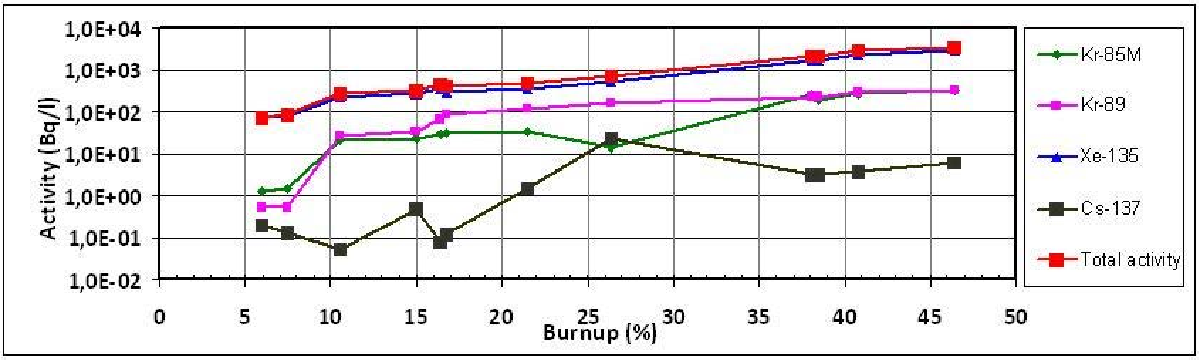

FIG. 8. Activity of air as a function of burnup (built based on data of table VIII).

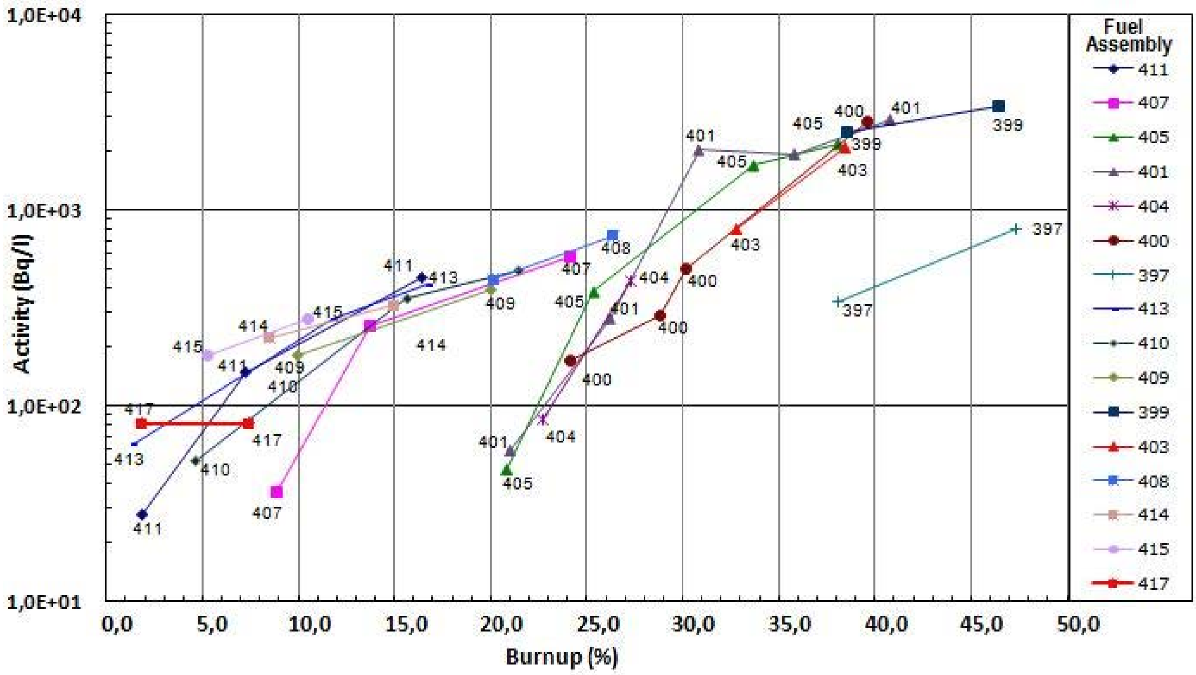

FIG. 9. Total activity of air for several FAs measured as a function of FA burnup.

Analysis of the measured activity of air in reactor pool (air between pool water surface and pool upper lid) resulted 2.42E+03 Bq/l, also well below the allowed limit of 2.77E-+04 Bq/l.

Finally, to verify the influence of the new LEU fuel on the physical-chemical parameters of water in the reactor pool core, the water in the primary cooling system was sampled before the input to the purification system. The results are shown in Table IX and in Fig. 10, and demonstrate that the new fuel did not cause any increase in the conditions of the system.

TABLE IX. MEASURED ACTIVITY OF WATER IN PRIMARY COOLING SYSTEM, BEFORE THE FILTER[d)] (Bq/l)

Nuclide	Half Life (hours)	06/01 2010	13/1 2010	27/1 2010	11/2 2010	25/2 2010	11/3 2010
^{95}Nb	841,439	1,8E+04	7,7E+03	3,1E+04	2,1E+04	1,6E+04	3,9E+04
^{95}Zr	1536,48	1,6E+04	1,7E+04	2,3E+04	2,3E+04	3,9E+04	3,1E+04
^{99}Mo	66,02	2,4E+03	3,8E+03	4,6E+03	4,1E+03	5,5E+03	2,7E+03
^{103}Ru	944,4	3,0E+03	4,6E+03	5,1E+03	6,1E+03	4,0E+03	4,0E+03
^{109}Cd	11136	1,6E+04	9,8E+03	8,3E+03	2,6E+04	3,8E+04	3,8E+04
^{131}I	192,96	3,5E+04	3,1E+04	3,8E+04	4,0E+04	5,0E+04	4,3E+04
^{133}I	20,8	1,9E+05	1,8E+05	3,3E+05	1,1E+05	3,1E+05	3,0E+05
^{135}Xe	9,11	4,6E+05	4,0E+05	5,4E+05	6,3E+05	4,7E+05	7,3E+04
^{137}Cs	264463	1,8E+04	2,6E+04	3,8E+04	5,0E+04	3,0E+04	5,5E+04
^{140}La	306,936	1,0E+04	2,1E+04	2,9E+04	5,5E+04	6,4E+04	6,4E+04
^{141}Ce	780	5,3E+03	5,0E+03	6,1E+03	1,1E+04	4,1E+03	3,4E+03
^{144}Ce	6823,2	1,6E+04	9,0E+03	2,7E+04	3,8E+04	1,7E+04	2,1E+04
Total activity		7,8E+05	7,1E+05	1,1E+06	1,0E+06	1,0E+06	6,7E+05

[d)] Measurements of activity for ^{106}Ru, ^{134}Cs, ^{134}I, ^{138}Xe, ^{139}Ce and ^{140}Ba resulted 0.00E+00.

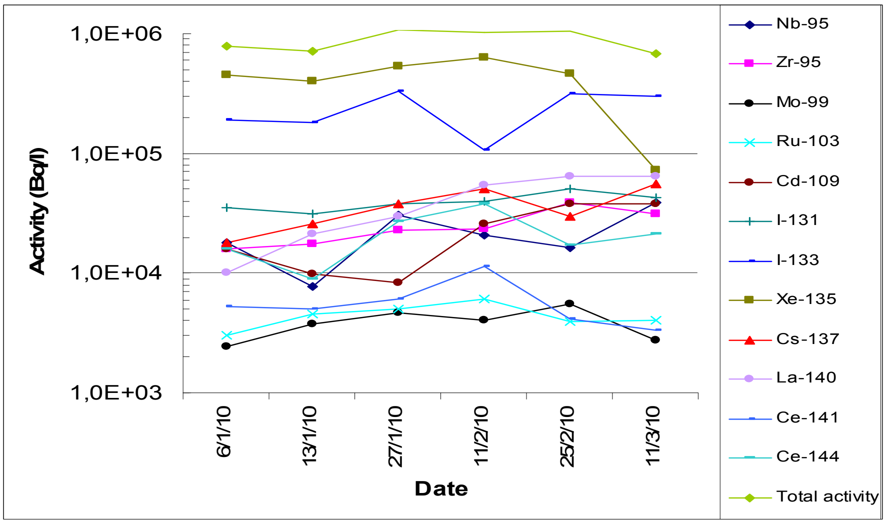

FIG. 10. Water activity in primary cooling system before filter (built based on data of table IX).

7. CONCLUSION

To convert the WWR-SM reactor from HEU to LEU fuel, a standard procedure was established to assess the integrity of the new fuel assemblies during its operating cycle and after being removed from the reactor core. The purpose of the procedure, based on the utilization of a Fuel assembly Cladding Tightness Verification System (FCTVS), was to confirm that the newly developed LEU fuel assemblies were leak tight, in accordance to the specifications, and their utilization would not increase radiation level in the reactor pool, neither would compromise the quality of water, in reactor and storage pools.

Measurements were made on individual fuel assemblies with different levels of burnup, in relation to the total activity and the activity of some specific radioisotopes that can result from the fission process. These measurements were complemented with measurements of activity of water in the reactor pool and in primary cooling system; and measurements of activity of air in the sampling box, and air in the upper reactor space region (space between pool water surface and pool upper lid). To complement the qualification process, the fuel assemblies were also subject to visual inspection.

All measured values were below the established upper limits, confirming the leak tightness of the new LEU fuel. Based on these results the conclusion was reached that the new fuel cladding tightness was in accordance with the specifications, and could now become the new standard fuel for the WWR-SM research reactor.

The new fuel will gradually replace the spent fuel in the core of WWR-SM research reactor until the full core is converted to LEU. After that, it will become the standard fuel for the reactor.

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DRY STORAGE PRACTICES AND EXPERIENCE

(Part 3)

WET AND SEMI-DRY STORAGE OF SPENT NUCLEAR FUEL AT THE BUDAPEST RESEARCH REACTOR

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Abstract

In the frame of the Russian Research Reactor Return Programme the spent nuclear fuel assemblies (SNFAs) of the Budapest Research Reactor (BRR) irradiated prior to 2005 were shipped back to the country of origin. Since the time of the reactor's first criticality in 1959 these SNFAs were stored in the At-reactor (AR) and Away from reactor (AFR) pools of the reactor. Fuel assemblies stored at the BRR site were noted as being in good condition following fuel characterization and observations performed prior to and during loading. The good condition of the SNFAs was due, amongst other things, to the well-controlled water quality in the storage pools, the periodic visual inspections, but mostly to the encapsulation of SNFAs that changed the temporary storage mode from wet to semi-dry. In accordance with the encapsulation technology and storage mode, the encapsulated fuel assemblies (packages) were kept in the same pool under water but the fuel was maintained in dry conditions, namely in an inert gas atmosphere. Therefore a safe, extended temporary storage was ensured until the assemblies were transported to their place of final disposal. The cladding of repacked fuel demonstrated that the applied encapsulation technology had slowed down or even stopped the corrosion effects of the cladding and thereby ensured safe extended storage conditions. This paper introduces the storage modes (practice of water chemistry and applied monitoring techniques) and gives a detailed description of the canning technology used. It also details experiences gained during the canning and the storage phases of the encapsulated fuel. The paper focuses less on the wet storage problems of SNF and more on the canning technology and the canning equipment itself, as well as on the repacking experiences gained during the transport container loading. Finally the paper draws some concluding remarks on lessons learned.

1. INTRODUCTION

The Budapest Research Reactor (BRR) is operated by the KFKI Atomic Energy Research Institute (AEKI) of the Hungarian Academy of Sciences and is located on the KFKI Campus site in Budapest, Hungary. The BRR is a Soviet origin tank type research reactor, and is moderated and cooled by light water. It achieved its first criticality in 1959 with a power of 2 MW. The first upgrade took place in 1967 when the power was increased to 5 MW using a new type of fuel and a beryllium reflector. The second, so called full-scale reactor reconstruction and upgrade project began in 1986 following 27 years of operation since initial criticality. Because of political obstacles, the outage was prolonged, and the reactor was not restarted until 1992. The upgraded 10 MW reactor received the operation license in November 1993. Since the time of startup the upgraded reactor has been operating ≈3500 hours/year on average without any significant problem.

In line with Hungarian safety regulations a Periodic Safety Review (PSR) was conducted in 2002–2003, as a result of which the operation license was renewed in November 2003. The PSR was a complex overall assessment concerned not only with the reactor operation but also with the SNF storage. At that time all SNFAs irradiated since 1959 were stored at the BRR's SNF storage pools.

The BRR, since its initial criticality, has been utilized as a neutron source for research and industrial applications on material science and education and for training proposes in the nuclear field. Presently the reactor is mainly utilized as a neutron source for irradiation and research.

2. BRR FUEL ASSEMBLIES

In the service life of the BRR two Soviet types of fuel elements have been used. They are:

- EK-10 from start up to 1967 (up until the first upgrade).
- VVR-SM and VVR-M2 from 1967[37].

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[37] For the time being the BRR is in operation with these types of fuel assemblies. By the middle of September 2009 the reactor started being converted from high enriched (HEU) to low enriched uranium (LEU) fuel. Prior to this,

2.1. EK-10 fuel assemblies

As with all VVR research reactors, the original fuel assembly was EK-10, which was made up of rods with 10% enriched uranium dioxide-magnesium alloy in aluminium clad. An assembly consists of 16 rods in an aluminium casing. The rods are held at both ends in aluminium grids fastened to the casing. A lifting lug is provided at the top of the assembly for handling. The geometry details of the fuel are shown in Fig. 1, while its main technical data is summarized in Table I. As seen in Fig. 1, two types of standard EK-10 assemblies exist: EK-10 assemblies with proportional pitch and with distorted pitch. The assemblies with distorted pinches could have one, two or three cut corners.

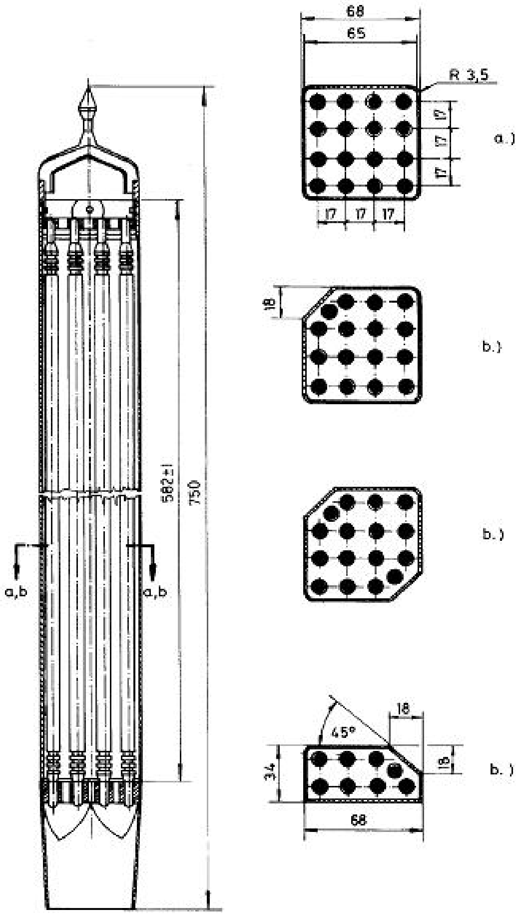

FIG 1. EK-10 nuclear fuel (NF) assembly.

HEU VVR-M2 fuel assemblies were used, and started being replaced by the same type of LEU assemblies. The conversion will be performed through four mixed cores configurations, containing HEU and LEU fuel. The conversion process is expected to be completed by the middle of 2012. The test operation with entirely LEU fuel is scheduled to be accomplished by the end of 2012.

TABLE I. MAIN DATA OF EK-10 FUEL ELEMENT

Fuel material:	U_3O_8 mixed with Al and Mg
Initial enrichment:	10 %
Aluminium fuel cladding thickness:	1,5 mm
Height of the fuel element:	750 mm
Horizontal dimensions of the fuel element:	68 × 68 mm
Total mass:	3500 grams
^{235}U Mass per fuel element:	128 g ^{235}U
Average burnup:	23 %
Estimated activity (calculated):	$<2,5 * 10^{13}$ Bq
Residual heat emission (calculated):	<0.34 W (after 40 years)
Dose rate (measured in 2005):	1-4 Gy/h
Last fuel element was unloaded:	1967

2.2. VVR-SM and VVR-M2 fuel assemblies

In 1967 (after first upgrade of the reactor), VVR-SM (Soviet origin) fuel elements with 36% enrichment were introduced in the core. The VVR-SM fuel consists of two concentric tubes inside of a hexagonal tube. The fuel meat is a 0.7 mm thick uranium dispersion in aluminium that is clad on both sides with 0.9 mm thick aluminium. This fuel element is used in single assemblies or as combined triple elements[38]. In 1996 a modified version of this type of fuel was introduced, the VVR-M2. The main change was to use uranium dioxide as the fuel with an increase of the mass of uranium per assembly from 108g to 125g. This fuel is also used in single or triple elements. The dimensions of both fuel types are the same. Their main technical data is given in Table II, and the overall dimensions can be seen in Fig. 2 (single and triple).

TABLE II. MAIN DATA OF VVR-SM AND VVR-M2 FUEL ELEMENTS

Denomination	VVR-SM	VVR-M2
Fuel material:	UAl_4 alloyed with Al	UO_2 dispersed in Al matrix
Initial enrichment:	36 %	
Fuel cladding:	Aluminium 0.9 mm	Aluminium 0.7 mm
Height of the fuel element:	865 mm	
Hexagonal pitch size:	35 mm	
Total mass:	single: ≈1 kg; triple: ≈3 kg	
^{235}U Mass per fuel element:	38.9 g ^{235}U	44 g ^{235}U
Average burnup:	50–55 %	55–60 %
Estimated activity (calculated):	$<3.5*10^{13}$ Bq SEQ	
Residual heat emission (calculated):	<1.5 W SEQ (after 3 years)	
Dose rate (measured in 2005):	10–60 Gy/h	
Last fuel element was unloaded:	These types of fuel elements are still being used	

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[38] In the fuel inventories fuel assembly numbers refer to single tube elements and the triple unit has three assemblies.

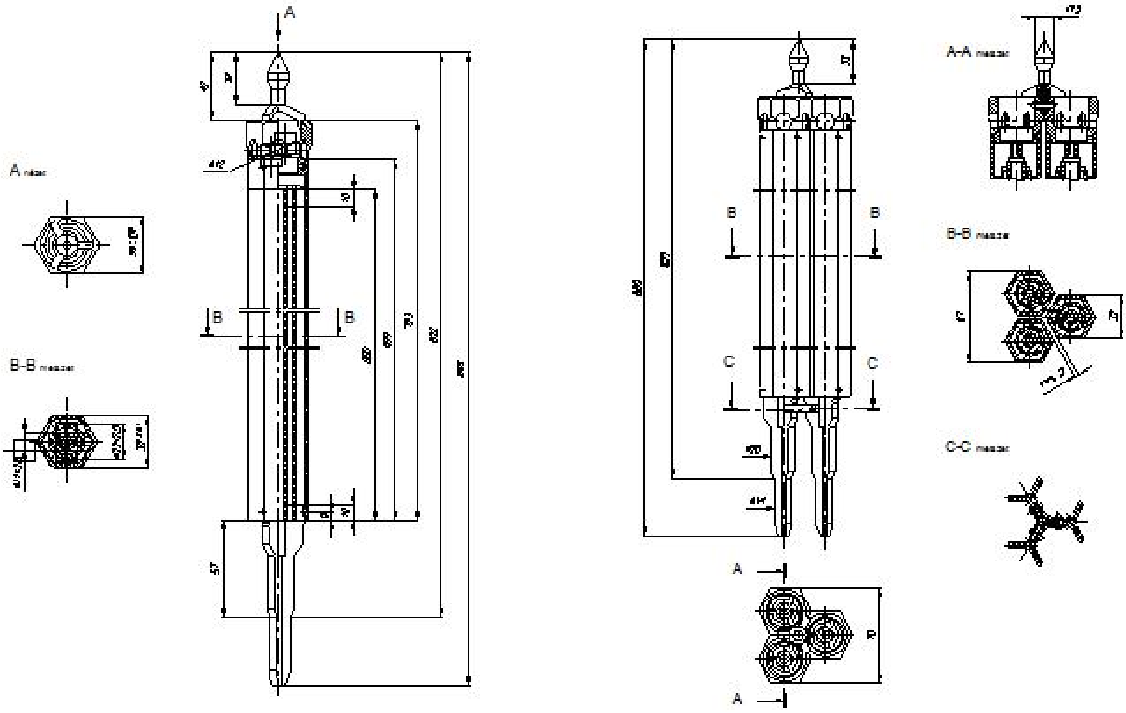

FIG. 2. VVR-SM and VVR-M2 single and triple fuel assembly.

2.2.1 Instrumented VVR-SM fuel assemblies

From the time of restarting the reactor in 1992 until the beginning of 2003 the cladding temperature was measured on three fuel assemblies by means of a thermocouple embedded under the cladding surface of the designated fuel assemblies[39]. The technology to put and fix the thermocouple was made by way of slotting the fuel meat (see Fig. 3). The last instrumented fuel assembly was removed from the core during the 22nd core modification (2005-09-12 to 2005-10-03), and the total number of instrumented SNF assemblies, altogether, was sixteen.

3. SPENT NUCLEAR FUEL STORAGE AT THE BRR

The original storage philosophy was to store the irradiated fuel elements under water pending their return to their country of origin: the Soviet Union. As the storage pool had a limited storage capacity and the negotiations on spent fuel back shipment did not start, another storage pool had to be constructed after ten years of operation. With the dissolution of Soviet Union the probability of any back shipment decreased dramatically, and the prospects for other alternative solutions extended far into the future. A decision was therefore made by the AEKI management to develop a canning technology to encapsulate the SNFAs (see Section 3.3 for details), establishing a so called semi-dry storage methodology. With the adoption of this semi-dry storage methodology, the safe interim spent fuel storage was extended for at least another 50 years.

In this section the spent fuel storage pools, storage modes, water chemistry and surveillance procedures are introduced with a detailed description of the semi-dry storage methodology for the SNFAs. The canning technology was an initiative to extend the safe interim storage of SNF. Therefore there is a great focus on the canning technology, the *canning* equipment itself and the experience gained with the storage and unpacking activities that were necessary for shipping back the SNF to the Russian Federation. Less attention is paid to the wet storage problems of SNF.

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[39] The cladding measurement obligation was released on 2003-01-07 by the LA's resolution of RE-3170.

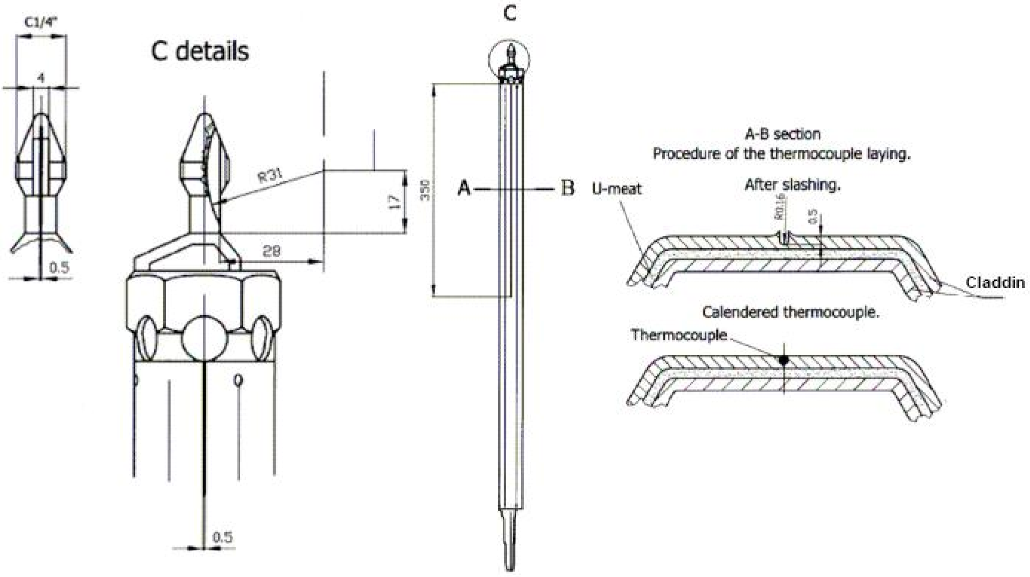

FIG 3. Draft of the instrumented fuel assembly.

3.1. BRR's storage pools

The BRR has two storage pools to ensure the temporary (interim) storage of the SNF assemblies irradiated in the reactor core. One of them is the at AR pool in the reactor hall; the other is the AFR pool that is located outside the reactor hall.

3.1.1. The at reactor (AR) Pool

The AR pool is located in the reactor hall, which is a semi-hermetically sealed, individually ventilated, rectangular structure of approximately 600 m^2. The central object of the hall is the heavy concrete reactor block. The AR pool is located nearby the reactor block. It consists of a steel tank, containing deionized water, with enough storage racks for 786 assemblies[40]. The pool, shown in Fig. 4, can be served through service plugs located in two of the three covering iron plates around the pool. The dimensions of the service plugs are sufficient for the transport container used to transport fuel assemblies between the AR pool and the top of the reactor or between the AR and the AFR pools. According to the operation limits and conditions (OLCs), to ensure good water quality it is necessary to filter the water using a mobile ion-exchanger whenever the conductivity reaches 3 μS/cm. The pool is served by the reactor hall crane.

3.1.1.1. Storage mode in the AR Pool

The storage mode in the AR pool is referred to as wet storage technology (the SNF assemblies are stored under water). The pool is prepared for VVR type fuel storage only (or VVR fuel type shaped assemblies). The AR pool serves four purposes:

- The first purpose is to ensure enough storage places for complete core unloading in case of emergency. This emergency fuel unloading is a requirement of the OLCs.
- The second purpose is to ensure the removal of the decay heat from the unloaded SNF (the residual heat of the SNF is the reason why the AR pool has, around the pool sides, a cooling pipeline that works as a heat exchanger).
- The third purpose is to ensure safe storage for any damaged and/or instrumented fuel. For this purpose four semi-hermetically closed storage tubes are kept in standby (see Fig. 4)[41].
- The fourth purpose is to store the beryllium reflector elements used in the reactor core.

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[40] In single VVR equivalents.

[41] No damaged fuel assemblies were stored in these tubes, instead, they were used to store the instrumented SNFAs until their encapsulation.

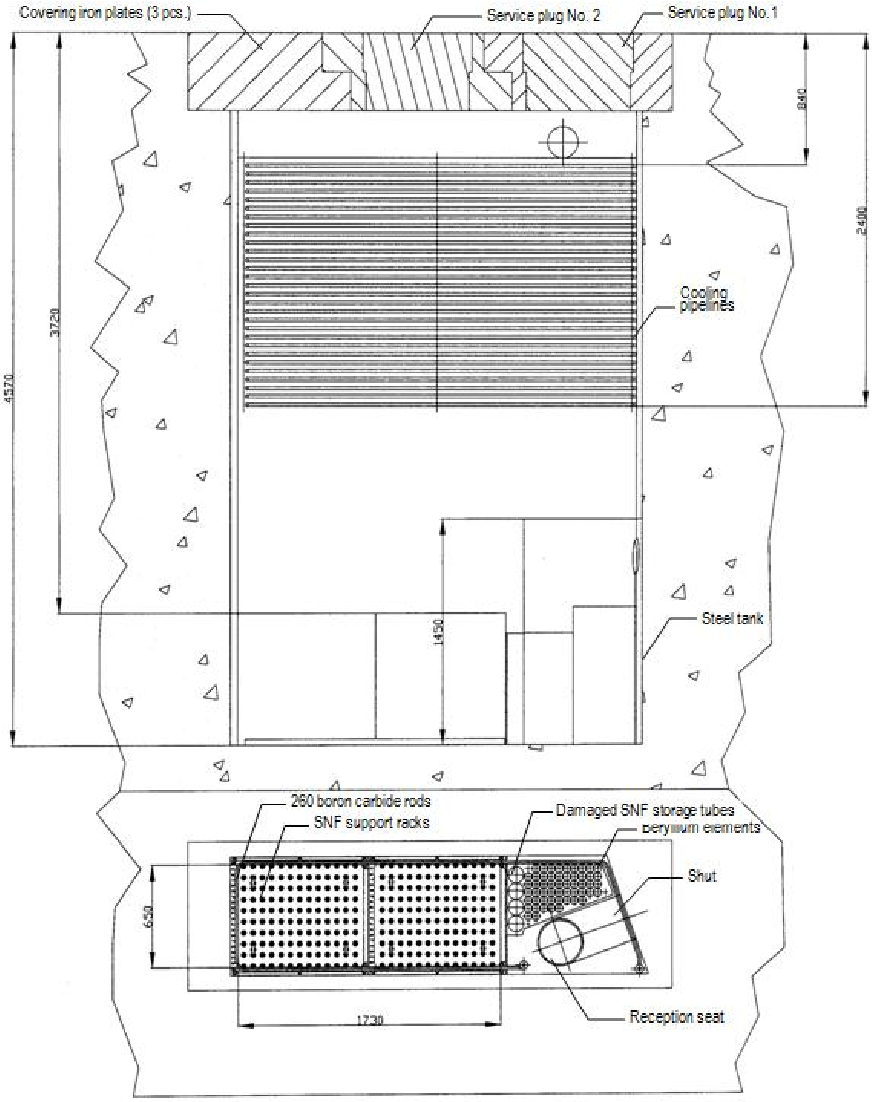
Covering iron plates (3 pcs.)
Service plug No. 2
Service plug No. 1
840
2400
4570
3720
Cooling pipelines
1450
Steel tank
260 boron carbide rods
SNF support racks
Damaged SNF storage tubes
Beryllium elements
Shut
650
1730
Reception seat

FIG. 4. AR pool.

3.1.2. *The Away From Reactor (AFR) Pool*

The Away From Reactor (AFR) pool was constructed in 1967, during the first upgrade of the reactor. The AFR pool is a stainless steel tank filled with deionized water with 188 storage locations (aluminium tubes) where the fuel elements can be stored. The storage tank is embedded within a monolithic, reinforced concrete block located below ground level. The AFR pool has a service plug above the reception seat. The hole and the loading capacity of the service plug and its construction are designed to perfectly fit the transport container. The cross section drawing of the AFR pool is shown in Fig. 5.

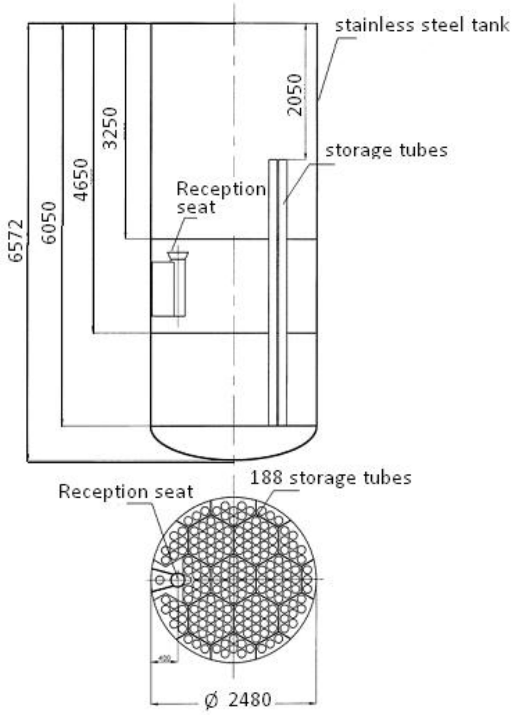

FIG 5. Cross-section drawing of the AFR pool.

Previously the pool had a metal-framed building over it, with a removable roof for access. The pool and its foreground area (AFR pool zone) are approximately 25 × 9 metres that is served by an overhead (angle type) crane with eight tonnes lifting capacity. In 2007, modification work was made to support the SNF shipping cask [1, 2]. Within the frame of this work a pool Service Hall was erected around the AFR pool. The service hall, shown in Figs 5 and 6, was equipped with a bridge crane with 15 tonnes lifting capacity and a transfer trolley. The outline view of the transfer trolley, with deployed railway, is shown in Fig. 8.

To ensure good water quality in the AFR pool, in accordance with the OLCs requirements, it is necessary to filter the water using a mobile ion-exchanger whenever the conductivity reaches 10 µS/cm.

FIG. 6. Service Hall Exterior.

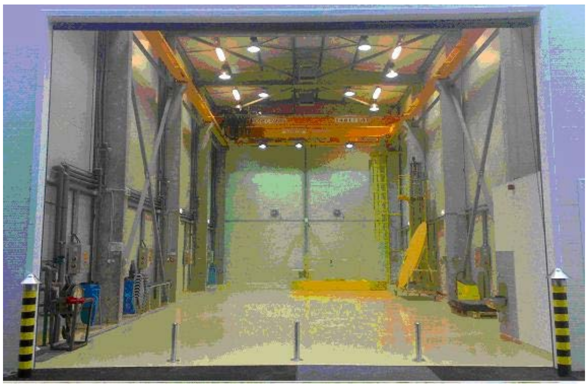

FIG. 7. Service Hall Interior.

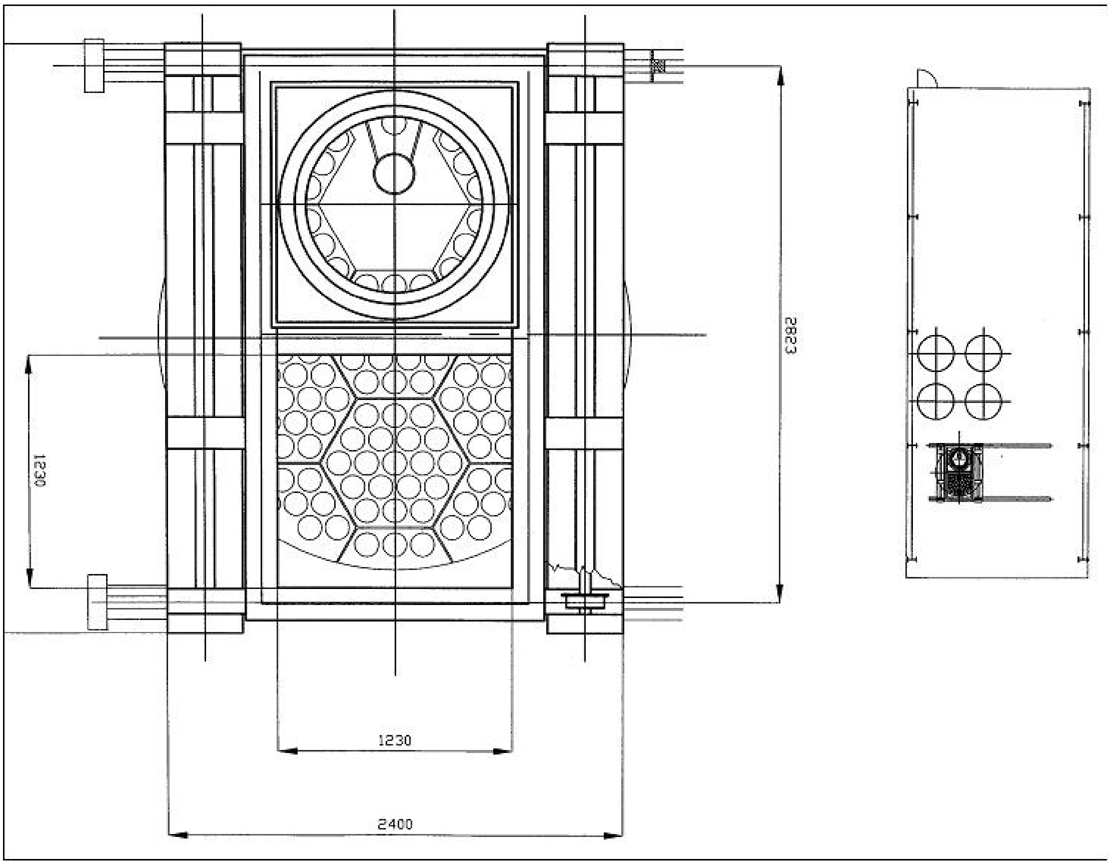

FIG. 8 Outline view of transfer trolley with deployed railway.

3.1.2.1. Storage modes in the AFR Pool

The AFR pool is licensed for wet and semi-dry (or semi-wet)[42] storage modes. The storage arrangement consists of 188 storage tubes made of aluminium. Once the fuel elements are placed within aluminium containers, each tube can accommodate four containers as described in next paragraphs.[43]

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[42] The storage is wet when the fuel assembly clad contacts with water, and semi-dry when a dried fuel assembly is enveloped in a sealed container, filled with inert gas, and this package (encapsulated fuel) is placed back into the storage pool water for biological shielding. Hence it is sometimes called semi-wet or semi-dry storage.

[43] All SNFAs stored in the AFR pool (in wet or semi-dry mode) were shipped back to their country of origin (Russian Federation) in the autumn of 2008 under the Russian Research Reactor Fuel Return Programme of the U.S. DOE National Nuclear Security Administration's Global Threat Reduction Initiative. Thus presently the AFR pool is empty.

3.1.2.2. Wet storage mode in the AR Pool

VVR fuel assemblies stored in wet storage mode are accommodated in aluminium storage tubes of 110 mm diameter. Three single elements or one triple element can be placed in an open container of 90 mm inner diameter, and four containers can be loaded (stacked) on each aluminium storage tube.

3.1.2.3. Semi-dry storage

An encapsulated package contains either one EK-10; one triple VVR, or three single VVR fuel elements. The packages can be stored in the 110 mm diameter aluminium storage tubes on four levels (four packages can be placed in one tube). See Section 3.3 for a more detailed description of semi-wet storage.

3.2. Water chemistry and surveillance procedures

The water quality control and surveillance procedures of both pools (AR and AFR) are regulated in the operation manual of the BRR. The OLCs define the water parameters, as described in the mandatory document of the reactor approved by the Regulatory Body. A summary of the controlled water parameters and the frequency of measurements/inspection, together with OLCs are given in Table III.

As can be seen on Table III, only a few of the parameters are measured only on-line and the majority of the parameters are controlled by sampling technique in laboratory measurements and evaluations. All routine monitoring, sampling and most of the measurements are performed by the reactor staff. The monitoring of corrosion products is made by Instrumental Neutron Activation Analysis (INAA), which is performed by the Materials Department Neutron Physics Research Group, working for AEKI.

Regarding the visual inspection, as there was no experience of long term wet storage, a visual inspection programme was implemented in 1995. The idea was to perform this inspection on comparison basis by decade, whereby the earlier observed cladding topography is compared with its current state. The first visual inspection was carried out in 1999 using the Hungarian Underwater Telescope. The results of this inspection showed different levels of corrosion for some fuel assemblies. The heaviest corrosion was detected in an instrumented fuel having a thermocouple in the cladding. There was no real worldwide experience of wet storage over such a long period but it was well known that aluminium corrosion can accelerate rapidly once it has started. Therefore a decision was made to change the storage mode from wet to semi-dry, that is, to encapsulate all the stored SNFAs and keep them in an inert gas atmosphere in order to slow down or even stop the corrosion process. The encapsulation of the fuel assemblies started in 2002, performed 2002–2004 and completed in 2004.

TABLE III. OPERATING LIMIT CONDITIONS FOR PARAMETERS IN SNF STORAGE POOLS

Measured / Inspected Parameter		AR pool	AFR pool
Continuous measurements (on-line measurements)			
Water level	OLCs	3.7 m	6.0 m
Water temperature	OLCs	Max. 60 °C; normal: 25–35 °C	X
Exhausted air activity	OLCs	Trip value(1)	X
Sample inspection/measurements on monthly basis			
Pool visual inspection	OLCs	Good visibility	Good visibility
Conductivity	OLCs	<3 μS/cm (+ B&A(2) shuffling)	<10 μS/cm
pH	OLCs	>4 and <7 (+ B&A(2) shuffling)	>4 and <7
$\Sigma\beta$	OLCs	Trip value(3) (+ B&A(2) shuffling)	Trip value(3)
γ-spectra ^{137}Cs [Bq/ml]	OLCs	Trip value(3) (+ B&A(2) shuffling)	Trip value(3)
γ-spectra ^{60}Co [Bq/ml]	OLCs	Trip value(3) (+ B&A(2) shuffling)	Trip value(3)
Concentration of corrosion products on quarterly basis (INAA measurements)			
Al [μg/kg]	OLCs	<50	<50
Cl [μg/kg]	OLCs	<50	<50
Cu [μg/kg]	OLCs	<10	<10
Fe [μg/kg]	OLCs	<50	<50
Visual inspection by HUT(4) 10 year			
Surface Cladding topography	OLCs	X	Comparison basis, ≈20 %

Remarks:
(1) Collected air sample activity is measured in the stuck
(2) B&A – Before and after (shuffling)
(3) No limit assigned. A measure is initiated in case of trip value.
(4) Hungarian Underwater Telescope

3.3. Semi-dry storage (SNF encapsulation)

To slow down or even to stop the corrosion of the cladding and thereby to ensure safe storage conditions of the SNF, it was decided to change the temporary storage mode from wet to semi-dry. To change the storage mode from wet to semi-dry, the experts of AEKI developed a canning technology and automatic canning equipment [3, 4]. This equipment was commissioned at the AFR pool of the BRR and the regulatory licence for SNF encapsulation was granted in March 2002.

The canning technology (for details see Ref. [1]) uses a tube-type capsule made of an aluminium alloy with a wall thickness of 3mm as shown in Fig. 9. The capsule is capable of accommodating one EK-10, one triple VVR type assembly or three single VVR assemblies. The EK-10 assemblies are packaged “as they are”, apart from a small part of the aluminium leg of a VVR assembly (≈73 mm) that is cropped off before canning. Encapsulation utilizes a closed technology in which the capsule undergoes a powerful drying procedure (heated by eddy current), is back-filled with dry nitrogen gas, closed with a shrink-fit capsule head and subsequently welded.

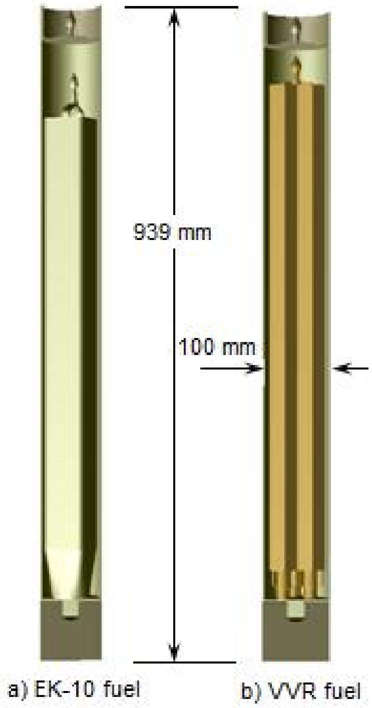

FIG. 9. 3D-drawing of the tube construction.

The canning machine is a programme controlled equipment housed in a compact steel container supplied with a revolver head. The head has five operating positions and a vertical transfer pipe for sucking up and floating down the capsule (Fig. 10). The inner cavity of the cask (operator cylinder) can be closed by using vacuum-tight sealing and valves. The canning steps are controlled by a PLC system. The technology includes a cropping machine that is used for cutting the fuel legs and in case there is a need to open an encapsulated SNF (e.g. in case of a faulty encapsulation). The canning cask is mounted in the hole of the service plug located above the reception area of the AFR pool. The bottom of the cask is submerged under the water about half a meter to provide enough shielding when the loaded capsule is transferred, while the container of the cropping machine is lowered to the highest position of the reception area (around 4 meters under water) ensuring safe underwater manipulation. A more detailed description of the cropping machine can be found in Ref. [3].

The main steps of the canning procedure are: (1) placing the capsule in the reception position and inserting the designated fuel assembly or assemblies into it; (2) sucking up the capsule into the cask through an extendable pipe by intensive water flow; (3) removing the water from the capsule; (4) drying and maintaining the SNF under a heating condition of130 degrees C for about 40 minutes, using a eddy-current heater; (5) vacuuming and filling up the capsule with nitrogen twice (<50 mbar vacuum, 2.5 bar N_2); (6) pressing in the capsule head; (7) welding the capsule head; (8) floating the canned fuel assembly back to the reception area; and (9) placing the encapsulated fuel assembly into its designated storage position (the encapsulated fuel assemblies are stored in the same pool under water). The steps from 2 to 8 are carried out semi-automatically and operator assistance is needed only for control and acknowledgement between steps. Steps 1 and 9 are performed manually (underwater operation).

The cycle time of the whole canning procedure is about 120 minutes allowing about four to five cans to be sealed per working day. The canning machine and its accessories, including the cropping device, form a compact and mobile system that ensures an almost completely automatic, safe and reliable encapsulation of spent fuel.

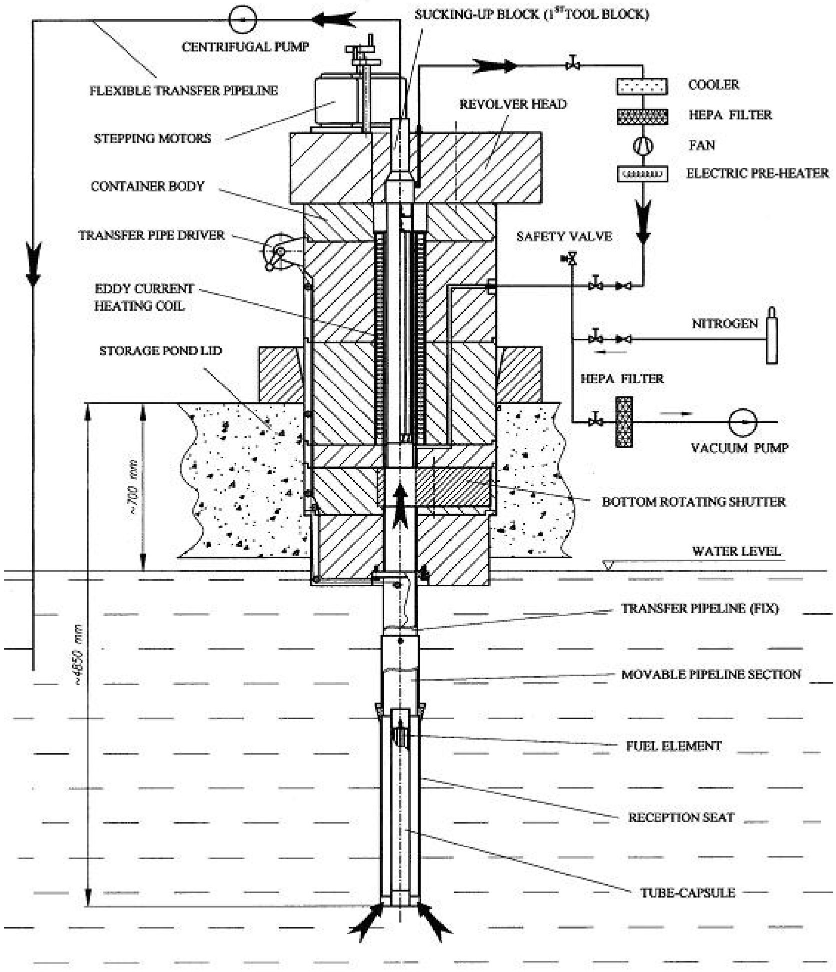

FIG. 10. Conceptual layout of the canning unit.

In the following two photos the upper part of the capsule is presented in the last two phases of the canning procedure. Fig. 11 shows the capsule head pressed into the canning tube before welding, while the capsule head secured by welding can be seen in Fig. 12.

FIG. 11. Pressed-in capsule head before welding.

FIG. 12. Canning tube secured by welding.

The canning process was successfully completed in August 2004 when all SNFAs irradiated before 1986 were encapsulated (82 EK-10, 228 single and 184 triple VVR fuel assemblies, that is altogether 342 cans). The encapsulated SNFAs were then stored under water in four levels, i.e. with four capsules stacked in each aluminium tube of the AFR pool.

4. CONCLUDING REMARKS

Due to the Russian Research Reactor Fuel Return Programme, all SNFAs irradiated before 2005 were shipped back to the Russian Federation in the autumn of 2008. As an integral part of the shipment agreement all 342 capsules had to be uncanned and the fuel assemblies, after undergoing visual inspection were loaded into the transfer containers for shipment. The experience gained during the un-canning and the visual inspection of wet stored fuel, as well as the inspection of the AFR pool, made after the shipment, gave a good basis to draw some conclusions in regards to the applied storage technology:

- Good water quality is a key factor for interim (long term) wet storage of SNFAs. Our experience extends to about a 40 year wet storage period and provides sufficient evidence for safe wet storage if the storage pool water is maintained properly.
- The SNFAs encapsulation technology and storage mode applied at the BRR slowed down or even stopped the corrosion process. Although only five years elapsed from the canning to the crop off cans the experience proved that the semi-dry storage of the fuel in an inert gas atmosphere stopped the cladding corrosion and ensured at least another 50 years of safe temporary storage, even for the fuel that previously exhibited significant corrosion.
- Inspection results of the drained AFR-pool found no fission products in the pool water or in the sludge (deposit) remaining in the bottom of the pool after draining. This provided additional evidence of the proper storage procedure to accommodate the SNFAs in the pool from 1968 to 2008.

It is hoped that the storage technology (primarily the semi-dry storage mode) presented in this paper may serve as a general solution for the research reactor community facing SNF storage problems and seeking to extend the safe temporary storage until a more permanent solution can be found.

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MATERIAL INTERACTIONS ON CANISTER INTEGRITY DURING STORAGE AND TRANSPORT

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Abstract

This paper is a condensed version of report DOE/SNF/REP-104 [1]. It was prepared with authorization of US DOE, and addresses material interaction and degradation mechanisms that could potentially affect the performance of the fuel-loaded US Department of Energy standardized spent nuclear fuel canister, during the interim storage period between canister loading and transportation for final disposition. For purposes of the report, an interim storage period of up to 50 years is considered. The 50 year storage duration is considered sufficient to address the period between canister loading and transport for final disposition. Corrosion damage, gas pressurization, metal embrittlement, and radiation damage are addressed, and based on the review of available literature and empirical data, the report concludes that, if properly dried, sealed, and maintained under controlled storage conditions, the containment functions of the canister are highly unlikely to be jeopardized during the identified period. Regardless the fact that the report only addresses material interactions that may affect physical performance of the DOE standardized spent nuclear fuel (SNF) canister, its content is applicable to most of the processes involving dry interim storage of research reactor spent fuel.

1. INTRODUCTION

This report addresses the potential material interaction and degradation issues that could affect the performance of a fuel-loaded US Department of Energy (DOE) standardized spent nuclear fuel (SNF) canister during the interim storage period between canister loading and transportation for final disposition. For purposes of this report, an interim storage period of up to 50 years is considered. Because the spent fuel canister will be relied upon to remain leak tight during interim storage and transportation, materials interactions are relevant only to the extent that they may affect canister integrity, and since safety credit will not be taken for the form of canister contents, the evaluation does not consider fuel degradation.

Previous analyses of potential material interactions with the DOE standardized SNF canister identified four potential forms of damage: corrosion, gas pressurization, metal embrittlement, and radiation damage. Previous evaluations have indicated that each of these forms has the potential to weaken the canister, thus increasing its susceptibility to breach and the subsequent ingress or egress of materials into or out of the canister. However, actual measured data along with the analyses given in this report illustrate that risks associated with storing SNF can be managed. The objectives of the report are limited to identifying and addressing materials interactions that affect the integrity of the canister. Material interactions that affect fuel, baskets, or other canister internals are outside the scope of this report, which pertains only to material interactions associated with the fuel types designated for packaging in Type 1a baskets in the standardized SNF canisters [2].These fuels represent a broad range of fuels; however, uranium metal, uranium aluminide, uranium aluminium silicide, and uranium oxide—all with aluminium cladding—are the focus of examination. The uranium aluminide fuel type constitutes the majority of fuels that will use a Type 1a basket, and is exemplified by such fuels as the aluminium plate fuels used in the Advanced Test Reactor (ATR), Missouri University Research Reactor, Massachusetts Institute of Technology, and Oak Ridge Research Reactor. Due to its very high surface area, thick corrosion layer on the cladding, a hydrous chemical composition of the corrosion layer, high radiation field, and potential water content [2, 3], the ATR fuel is expected to be the limiting case with respect to challenging the integrity of the canister, and was therefore selected to conservatively represent the fuels within the scope of this report. Because limited data is available regarding potential reactions under extended and off-normal storage conditions and also because of the diversity of the fuels included within its scope, the conclusions of this report are contingent upon implementation of a confirmatory monitoring programme for canisterized fuels that will identify any conditions that are inconsistent with the conclusions of this report. Additional scientific investigation to evaluate possible long term increases in free water or physisorbed water content via equilibration or dehydration/decomposition of chemisorbed water is also recommended.

2. DESCRIPTIONS

With the exception of some commercial SNFs and Hanford fuels (primarily N-Reactor), virtually all other SNFs under the purview of DOE will be placed in a DOE standardized SNF canister before being transported to a repository for final disposal.

There will be four canister configurations. Each will be constructed of stainless steel (316L) with a wall thickness of 0.953 cm, having diameters of 45.72 and 60.96 cm (18 and 24 in.), respectively, and lengths of 304.8 and 457 cm (10 and 15 ft), respectively [4]. The Type 1a basket was designed for the 45.72 cm (18 in.) diameter canister. The shorter of the two 45.72-cm-diameter canisters has a fractional volume and fuel loading of the longer canister; therefore, the analysis given for the longer canister will scale down. Small variations in the actual dimensions of the canisters are not significant to this type of analysis. The operational parameters will be: design pressure 50 psig; temperature range -29 to 315 °C, and a helium backfill.

2.1. Fuel and canister assumptions

The technical discussion and conclusions presented in the report are based on following assumptions:

- Loaded and sealed canisters will contain no organic materials in the form of paints, films, spacers, or other components.
- Internal canister surfaces free of machining oil, solvents, and other organic substances.
- The canister internal guides and support structures for the fuel will be made of stainless steel.
- Type 1a baskets will be made of stainless steel or gadolinium alloy.
- Maximum fuel burnup per canister is 7,150 MWd (947 MWd/kg and 7.55 kg of 235U fissioned).
- The drying process considers pre-existing corrosion products and thus negates the need for specific assumptions regarding the condition of the fuel.
- Loaded canisters vacuum dried at 200–250 °C and satisfying criterion of maintaining a pressure of 3 torr, or less, when isolated from the vacuum pump for a period of 30 minutes.
- Following drying, canisters will be inerted (backfilled with He to achieve a static pressure of ~5 psig at ambient conditions), permanently sealed, and leak tested.
- After canister closure, storage temperatures will remain below the heating temperature achieved during the drying process. The canisters will be stored in a clean, dry environment where they are protected from direct exposure to weather and other adverse conditions.

2.2. Water content

2.2.1. Chemisorbed, Physisorbed, and Free Water

Many potential degradation mechanisms are linked to the presence of water. If complete drying is not feasible, some residual water will remain in the sealed DOE standardized SNF canister [5]. The residual water may exist as chemisorbed, physisorbed, or free water. From a materials point of view, the free water is the most detrimental form, potentially causing corrosion and hydrogen gas production, even at ambient temperatures. Physisorbed water is in equilibrium with free water and can slowly replenish free water even at ambient conditions. Lastly, chemisorbed water may also contribute at elevated temperatures. Each of these forms is discussed below.

Chemisorbed water is the water bound as hydroxyl moieties or the water of hydration in various corroded fuel and fuel components; its concentration is generally limited by the stoichiometry of the hydroxides, oxyhydroxides, or hydrates comprising the corrosion products. For example, boehmite, a potential corrosion product of aluminium, has the chemical structure AlOOH (commonly written as $Al_2O_3 \bullet H_2O$). Complete removal of chemisorbed water generally requires much higher temperatures than are achieved during typical drying processes. In most cases chemisorbed water is effectively bound in the solid matrix, having a low equilibrium vapour pressure, even at elevated temperatures. Because it is chemically bound in corrosion products and has a lower vapour pressure than free water, chemisorbed water does not participate as readily as the other forms of water in degradation mechanisms. However, chemisorbed water may, over the long term, release water via decomposition mechanisms to replenish free and physisorbed forms.

Physisorbed water is physically adsorbed on the exposed fuel and canister surfaces and held by hydrogen bonding. It is in equilibrium with free water. The total amount of physically bound water depends on the total surface area of solid components within the container, their sorption characteristics, the temperature of the surfaces, and the partial pressure of water above the sorption surface. For most surfaces with physically bound water, the outermost water layer is readily removed by heating to 50 °C under vacuum. The removal of subsequent layers becomes progressively harder.

Free water, which is water in liquid or vapour form that is not chemically or physically bound, is more easily removed by adding heat and/or reducing pressure than other forms. The liquid form may be present as pooled liquid or as interstitial (or pore) water held by capillary forces within a particulate corrosion product bed or in porous substrates. At a given heat input rate, the removal of pooled water is constant, whereas interstitial water is associated with a falling rate [6]. Drying rates of interstitial water steadily decrease with time (falling rate) because of the diffusion time required for it to escape from the pores or crevices of a matrix.

3. DRYING METHOD

Before canisters are sealed, fuels need to be dried to remove as much of the free, physisorbed, and chemisorbed water from the system as possible.

The drying process must reliably reduce residual water levels to sufficiently eliminate or safely limit detrimental chemical reactions during the 50-year life of the canister. This can be difficult without supplemental heating of cold spent fuel, especially if ice forms during vacuum drying. Knoll and Gilbert at Pacific Northwest Laboratory developed a drying protocol to address these issues in 1987 [7. 8]. Recommendations for removing bound water are as follows [7]:

The canister should be drained of as much water as practicable and evacuated to less than or equal to 3.0 torr. After evacuation, adequate moisture removal is verified if the evacuated canister maintains a constant pressure over a period of about 30 minutes, while isolated from the vacuum pump. The canister is then backfilled with an inert gas (e.g. helium) for applicable pressure and leak testing. The canister is then re-evacuated and re-backfilled with inert gas before final closure. Care must be taken to preserve the purity of the cover gas and, after backfilling, should verify its purity by sampling.

- The procedures should reflect the potential for blockage of the evacuation system as a result of ice formation during evacuation. Icing can occur from the cooling effects of water vaporization and system depressurization during evacuation. Icing is more likely to occur in the evacuation system lines than in the canister because of decay heat from the fuel. A staged draw down or other means of preventing ice blockage of the canister evacuation path may be used such as measurement of canister pressure not involving the line through which the canister is evacuated.
- A suitable inert cover gas, with quality specification that ensures a known maximum of impurities, should be specified to minimize this source of contaminants.
- The process should provide for repetition of the evacuation and re-pressurization cycles if the canister interior is opened to an oxidizing atmosphere following the evacuation and re-pressurization cycles (as may occur in conjunction with remedial welding, seal repairs, etc.).

The effectiveness and time required for drying depends on variables such as applied vacuum, temperature, rate of heat transfer, and rate of mass transport of water out of the fuel. The decay heat emanating from irradiated fuel with relatively short cooling (decay) times will drive the drying process, whereas older and colder fuel may need significant external heat input to dry.

Alternative processes for drying are possible, provided guidelines such as pressure and temperature limits are not violated. Typical DOE drying processes use the rebound test (evacuation to 3 torr with a 30-minute hold as discussed above) with minor changes to improve factors such as heat transfer.

Hanford established 3 torr at 45 °C as the bounding limit for water pressure in its drying procedure for the multicanister overpacks (MCO). Testing of sealed MCOs confirmed that chemisorbed water or other sources did not release appreciable amounts of free water as internal conditions equilibrated [9]. Follow-up measurements showed a continuous decrease in free water content to an undetectable level [10–12]. This was expected since uranium metal within the canister scavenges free water and hydrogen, and tends to remove both from the MCO headspace. A compelling argument was made that significant fuel corrosion abated after 3 years within the MCOs, based on fission gas measurements. This suggested that the MCOs moved towards a static state internally at least for maximum ambient temperatures of approximately 100 °F (38 °C). The MCO storage and monitoring programme does not include observations at elevated temperatures and thus, cannot be relied upon to confirm that chemisorbed water species remain bound at elevated temperatures or over a 50-year period.

The drying procedure used as the basis for this analysis presumes that loaded canisters will be vacuum dried at high temperature, a minimum of 200 °C and a maximum of 250 °C, to a pressure less than 3 torr. Once a vacuum of 3 torr is established, a pressure rebound test at drying temperature will be done. If, after 30 minutes, the internal pressure in the isolated canister does not exceed 3 torr, then drying is complete. However, if the canister fails the rebound test, then vacuum drying is repeated followed by another rebound test. This procedure will continue until the rebound test is passed or drying is stopped. Once drying is complete, the canister is inerted, sealed, and leak-tested.

4. FREE WATER CONTENT FOLLOWING DRYING

The drying process described above will create a low partial pressure of water at elevated temperature. Assuming a partial water vapour pressure of 3 torr at 20 °C, the dew point is equivalent to -5.8 °C [13].This establishes the maximum temperature at which free water remaining in a sealed canister could condense. In other words, any free water within the canister would remain in vapour phase unless temperatures dropped to -5.8 °C or below.

Calculations consider the 45.72 cm (18 in) diameter, 457 cm (15 ft) long canister. For the estimation of the free water mass (as water vapour), the calculations assume a partial pressure of 3 torr at room temperature. By using the Ideal Gas Law, the calculated free water for the loaded canister with 195 L net void volume (assuming 750 L empty canister to 195 L filled canister), after drying to 3 torr (0,004 atm), could be as much as 0.566 mL (0.032 moles). Thus, at room temperature (298.15 K), the expectation is that, initially, less than 1 mL of water vapour will remain within the canister after sealing. As an absolute limit to the potential residual water mass, drying an empty canister with a void volume of 750 L yields a maximum of 2.18 mL (0.123 moles) of water as vapour. Other constants used are R equal to 0.082057 (L atm/K mol), molecular weight of water equal of 18 (g/mol) and water density equal to 1 (g/ml).

These amounts of residual water vapour are comparable to those in the dried MCO casks, which had an estimated residual water content of 0.04 to 0.72 mL [10–11]. The minimal mass of free water ensures a very limited possibility for water-mediated degradation mechanisms, unless physisorbed and chemisorbed water were able to re-equilibrate to establish higher free-water concentrations.

The level of dryness described above would be sufficient to prevent water related degradation, provided these conditions were static. However, the water vapour concentration may change over time. Free water as a reagent could slowly replenish itself from chemisorbed sources and, over 50 years, produce unacceptable levels of corrosion or hydrogen gas. Evaluation of the significance of potential water from dehydration of existing corrosion product over 50 years is therefore recommended.

5. PHYSISORBED WATER

The amount of initial water present as physisorbed water after drying is considered insignificant. Sustained heating of the canister under vacuum above 200 °C when drying will eliminate essentially all physisorbed water present [14, 15]. A relative humidity of approximately 30%, equivalent to 5.3 torr pressure at 20 °C, will establish approximately one monolayer of physisorbed water on an alumina surface [15]. At 20 °C, a water vapour pressure of 3 torr represents 17% relative humidity and will generate a surface coverage of less than 0.6 monolayers of sorbed water; the endpoint drying condition of 3 torr at 200 °C will result in an insignificant amount of adsorbed water. Less rigorous drying at 100 °C under evacuation for 71 hours resulted in 0.9 monolayers of water [14]. Assuming conservatively that, upon sealing the canister, one water layer exists on the cladding at a coverage of 1.87×10^{-8} mL/cm^2 or 6.25 molecules $H_2O/100Å^2$, the mass of physisorbed water corresponds to less than 0.02 mL of water [14]. Even with a conservative estimate of one water monolayer, physisorbed water sources are insignificant.

6. CHEMISORBED WATER

In view of the high design limit for the canister temperature (315 °C), the chemisorbed water content may have significant ramifications. A significant amount of water is chemisorbed on the alumina corrosion layer of the aluminium cladding. The amount of chemisorbed water calculated below is approximately 1.7 L, based on surveys of the average ATR fuel plates and assuming the oxide layer is boehmite instead of bayerite or gibbsite [16]. The oxide thickness on the aluminium cladding is conservatively 0.00340 cm, based on a 9% thinning of aluminium cladding (0.0380 cm) [16].

ATR plate dimensions: 124.45 cm long, 7.58 cm average arch width

Exposed aluminium: 2 sides, 19 plates per assembly, 30 assemblies per canister

Surface area: 124.45 cm • 7.58 cm • (2) • (19) • (30) = 1.08×10^6 cm^2

Boehmite properties: density 3.01 g/cc, thickness = 0.00340 cm, M.W. = 119.98 g/mol

Boehmite moles: 1.08×10^6 • 0.00340 • 3.01/119.98 = 91.7 moles

Boehmite water content: 91.7 moles of water, 18.0 g/mol (water) = 1650 g or mL

Past working assumptions were that the corrosion layer was mainly boehmite ($Al_2O_3 \cdot H_2O$, monohydrate), that boehmite does not dehydrate below 300 °C, and that no significant pressure increases or water releases would occur at maximum temperature [17]. A protective layer of boehmite is created on ATR cladding and boehmite corrosion forms preferentially under high temperature reactor conditions, 80 °C or above. However, conversion of alumina monohydrate to trihydrate is possible, particularly if fuel is stored underwater for extended periods [18]. During long term basin storage, the boehmite can convert to bayerite or gibbsite, both of which are polymorphs of the trihydrate $Al(OH)_3$ (also written as $Al_2O_3 \cdot 3H_2O$). Analyses of oxide scrapings from basin-stored ATR fuel plates have indicated that the oxide layer consists of $Al(OH)_3$ of unidentifiable phase [19]. Therefore, the alumina corrosion layer on aluminium-clad SNF may actually consist of the monohydrate, the trihydrate, or mixtures of the two. So, depending on fuel history, approximately 1.7 to 3.2 liters of chemisorbed water may be present in each canister. In the calculation below, bayerite properties are used to represent both trihydrates.

Bayerite properties: density ~2.50 g/cc, M.W. = 155.98 g/mol

Bayerite moles: 1.08×10^6 • 0.00340 • 2.50/155.98 = 58.6 moles

Bayerite water content: 58.6(3X) moles of water, 18.0 g/mol (water) = 3170 g or mL

If more highly hydrated morphologies of alumina than boehmite are present in any appreciable quantities, significant free water release could occur below the maximum canister temperature limit of 315 °C. For example, bayerite and gibbsite have 3 moles of chemisorbed water ($Al_2O_3 \cdot 3H_2O$) for every mole of alumina, which will begin to dehydrate at temperatures of ~150 °C [17]. Dehydration can begin at even lower temperatures depending on particle size and the presence of impurities.

The quantity of free water that could be released from the trihydrate $Al(OH)_3$ at maximum drying temperature is uncertain. Anywhere from 2/3 to nearly all of the water in the trihydrate could be released (forming $Al_2O_3 \cdot xH_2O$, with $1 > x > 0.5$), depending on material and process characteristics.

Another source of potentially significant chemisorbed water is hydrated uranium oxides, formed by corrosion of the uranium aluminide (UAl_x, with $2 < x < 4$) fuel meat in breached aluminium fuel. In addition to UAl_x, exposed uranium metal alloys or uranium oxide (UO_2) fuels, some of which also are candidates for packaging in the Type 1a baskets, can corrode to form uranyl hydroxides, $UO_3 \cdot xH_2O$ ($0 < x < 2$), and possibly the peroxides, $UO_4 \cdot xH_2O$ ($x = 2$ or 4). In laboratory tests, when irradiated UO_2 is flooded with water, the hydrate schoepite ($UO_3 \cdot 2H_2O$) forms quickly, and if the fuel remains wet for approximately two years, complete conversion to metastudtite ($UO_4 \cdot 2H_2O$) can occur [20]. Hence, several hydrate forms are possible, depending on the history of the fuel.

The characteristic decomposition temperatures of the hydrated uranium oxides can vary considerably. The decomposition conditions depend on such factors as particle size, specific surface area, residual impurities, and phase. At 150 and <100 °C, respectively, fully hydrated alpha and beta $UO_3 \cdot 2H_2O$ decompose by the reaction:

$$UO_3 \cdot 2H_2O \rightarrow UO_3 \cdot H_2O + H_2O$$

Pure $UO_3 \cdot H_2O$ decomposes to UO_3 at temperatures greater than 350 °C, while $UO_3 \cdot H_2O$ in spent fuel partially decomposes at 230 °C by the reaction:

$$UO_3 \cdot H_2O \rightarrow UO_3 \cdot 0.5H_2O + 0.5\ H_2O.$$

$UO_3 \cdot 0.5H_2O$ decomposes at temperatures in excess of 500 °C.

The uranium peroxides decompose at considerably milder conditions. Studtite ($UO_4 \cdot 4H_2O$) decomposes to UO_3 in a multistep process that begins with the loss of two molecules of water, forming metastudtite ($UO_4 \cdot 2H_2O$) at temperatures in the range 20–100 °C. Metastudtite decomposes to UO_3 at temperatures that may range from 130 to 425 °C; impurities favour the lower decomposition temperature [21].

The proposed drying method (drying at 200–250 °C to 3 torr) is expected to decompose the hydrated aluminium oxides and uranium oxides to AlOOH and $UO_3 \cdot 0.5H_2O$, respectively. However, the validity of the limited water vapour pressure data as a function of temperature for these partially hydrated oxides is uncertain. Water vapour concentration over time may change because of high temperature or slow decomposition (dehydration) of corrosion product on aluminium fuels.

7. ANALYSIS

A literature survey of potential SNF material interactions with the DOE standardized SNF canister has identified corrosion damage, gas pressurization, metal embrittlement, and radiation damage for further consideration. Although some degradation to the canister may occur from corrosion, metal embrittlement and radiation, this analysis concludes that it will not jeopardize the ability of the canister to perform its storage and transportation safety functions over the 50-year period considered. Pressurization due to residual water or hydrogen gas can be managed, provided proper heated drying is conducted before sealing.

7.1. Corrosion damage

Potential corrosion of the canister can proceed by general and localized corrosion. General corrosion pertains to the "uniform" surface oxidation of the steel due to atmospheric or aqueous exposure. Localized or nonuniform corrosion pertains to degradation and oxidation at isolated spots on the steel surface. Localized corrosion is caused by the localized degradation of the passive layer on stainless steel or by a locally enhanced corrosion potential. Localized corrosion includes pitting, crevice corrosion, intergranular corrosion (IGC), and stress corrosion cracking (SCC). Both general and localized corrosion proceed by an electrochemical oxidation mechanism in which the metal is oxidized and dissolved by the anodic reaction:

$$\mathrm{M} \rightarrow \mathrm{M}^{+n} + \mathrm{n\,e}^{-}$$

with the corresponding reduction of hydronium ions or oxygen in solution by the cathodic reactions

$$2\,\mathrm{H}^{+} + 2\,\mathrm{e}^{-} \rightarrow \mathrm{H}_2$$

$$\mathrm{O}_2 + 4\,\mathrm{H}^{+} + 4\,\mathrm{e}^{-} \rightarrow 2\,\mathrm{H}_2\mathrm{O}$$

$$\mathrm{O}_2 + 2\,\mathrm{H}_2\mathrm{O} + 4\,\mathrm{e}^{-} \rightarrow 4\,\mathrm{OH}^{-}.$$

The electrochemical mechanism requires charge balance between the anodic and cathodic reactions to maintain overall electrical neutrality. Because of the electrochemical nature of general and localized corrosion, the corrosion reactions can proceed only in a conductive liquid medium such as liquid water or microscopic surface water films that can form on the steel surface by condensation of water vapour at greater than 70% relative humidity [22].

It should be noted that corrosion can occur without liquid water present. Referred to as gas phase corrosion, this process is comparatively slow relative to other forms of corrosion. Under steam, a corrosion rate of 0.4 mpy[44] was recorded for stainless steel 304 at 371 °C [23]. Gas phase corrosion is not considered a significant degradation process.

Stainless steels that have a composition with more than 12 wt% chromium form a thin (1–5 nm) protective passivating layer of chromia (Cr2O3) on the surface [24]. This layer is the source of stainless steel's corrosion resistance. The standardized SNF canisters will be fabricated from 316L stainless steel because of its resistance to corrosion. The nominal chemical composition of austenitic 316L stainless steel is 16–18 wt% Cr, 10–14% Ni, 2–3% Mo, 2% Mn, <0.03% C with the balance Fe [25].

7.1.1. General corrosion

General or uniform corrosion is the most common form of corrosion and, as the name implies, propagates in a relatively uniform manner over exposed surfaces [22]. General corrosion is categorized by physical environment, atmospheric and aqueous exposure. Of the two, atmospheric exposure best represents nominal conditions seen by the SNF canisters. Under ideal conditions without contaminants and stable non-condensing temperatures, metal surfaces will not corrode below a relative humidity of 100%. The formation of thin condensed surface films of water due to deliquescent contaminants and temperature fluctuations or gradients (>10 °C) can produce corrosion at a much lower relative humidity. This critical humidity level at which corrosion can occur for stainless steels is generally 70% or greater.

Atmospheric corrosion is often divided into four exposure categories: rural, urban, industrial, and marine [24]. Aqueous corrosion is usually categorized as lake, river, and sea water corrosion. The

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[44] mpy - mils (one-thousandth of an inch) per year.

underlying corrosion mechanisms, however, are the same for each of these conditions, consisting of electrochemical oxidation that requires a condensed film of water or water immersion to proceed. The corrosion rates of 316 stainless steels are negligible in long term atmospheric, lake, and river water exposure conditions [26] Alloy 316L is expected to be even more resistant to corrosion. Even if condensing conditions existed in the canister after drying by the recommended method, and even if the contaminants in the canister were comparable to industrial environmental conditions, the expected rate of general corrosion is only 0.03 microns per year or 3.0 microns total internal and external penetration over a 50-year storage period. These values are orders of magnitude lower than the maximum permissible rate of 10 microns per year (0.4 mils per year) or 510 microns (20 mils) over 50 years for wall thinning of the DOE standardized canister [27]. Therefore, general corrosion of the 316L stainless steel standardized canisters is considered of no consequence.

7.1.2. Localized corrosion

The potential localized corrosion types that may be applicable to the DOE standardized canister are: pitting corrosion, crevice corrosion, SCC, and IGC. Pitting corrosion involves localized attack by contaminants such as chlorides, which function as localized acidifiers that destroy the passivating chromic oxide film. Crevice corrosion is associated with shielded or occluded configurations that impede oxygen diffusion. The impeded oxygen diffusion causes a localized dissolved oxygen gradient that establishes a corresponding spatially separated, anode-cathode couple. The oxygen gradient changes the local oxidation potential, and forms an anodic site that oxidizes and dissolves the metal within the crevice, occlusion, or shielded spot. SCC is caused by induced stress and attack by caustic or chloride solutions of sufficient concentration to weaken or destroy the passive film and initiate corrosion at the stress sites. IGC may occur at heat sensitized sites, wherein the chromium oxide layer is weakened as a consequence of heat induced carbon diffusion to the grain boundaries, reaction with the chromium oxide, and precipitation of chromium carbide. The process weakens or destroys the passive chromium oxide layer at the grain boundaries, exposing them to corrosion.

All of these corrosion mechanisms involve an electrochemical process where water, either liquid or a condensed thin film (from atmospheric moisture), is required to conduct the ions and electrons between the corresponding anodic and cathodic half-reactions. In the absence of a conductive medium, these reactions cannot proceed. Therefore, if the canisters are dried to the recommended three torr criterion, the residual water vapour pressure will be insufficient to form condensed films at ambient temperatures. Because a 3 torr water vapour pressure (at 20 °C) corresponds to a dew point (frost point) of -5.8 °C, condensed water films will not form above this temperature. Below this temperature, the condensate will likely form a discontinuous frost layer. Even if the frost layer could support an electrochemical reaction, the corrosion rate will also decrease markedly by such a reduction in temperature. In addition, the localized corrosion mechanisms will be prevented by the lack of ‘initiators’ in the canisters, as discussed below.

7.1.2.1. Pitting and crevice corrosion

Pitting corrosion initiates at a weak spot on the chromium oxide passivating film. Manganese sulfide inclusions, flaws, steps, secondary phases or other physical heterogeneities at the surface are often starting points of pitting corrosion. Once started, pits can continue to grow via a self-sustaining mechanism where halides, specifically chloride, have a critical role. Within the pit, chemical conditions can become quite different from the bulk solution [28]. As the anodic reaction in the pit dissolves the metal and forms M+n concentrations, increasing amounts of Cl- migrate into the pit to balance the charge and form metal chloride, MClx. The metal chloride is then hydrolyzed by water to form metal hydroxide and hydrochloric acid.

$$M+Clx- + xH2O \rightarrow M+OHx- + xH+Cl-.$$

Consequently concentrations of chloride ion, as high as 6 M, and highly acidic conditions, with low pH levels, can form within the pit. Such conditions break down the chromia protective film. The reactions

$$6H+ + 6Cl- + Cr2O3 \rightarrow 2CrCl3 + 3H2O$$

and

$$CrCl3 + 3H2O \rightarrow Cr(OH)3 + 3H+ + 3Cl-$$

accelerate the local corrosion of the steel, and prevent repassivation. However, chloride concentrations greater than 100 to 1,000 ppm are required for pitting corrosion to occur [24]. Chlorine is not a fission product, and chloride contamination in the ambient environment is below 100 ppm for the national average and is particularly low in Idaho (where many of the canisters will be loaded) with approximately 0.07 mg/L (0.07 ppm) for atmospheric sources and 0.2 mg/L (0.2 ppm) from rain at a pH of ~5.5 [29]. Hence, chloride concentrations in DOE standardized canisters are expected to be too low to support significant pitting corrosion, both internally or externally.

Crevice corrosion is caused by the development of a dissolved oxygen gradient within a shielded or occluded configuration, wherein the differential oxygen concentration establishes an oxidation potential gradient [22]. A sufficiently large change in the oxidation potential within the occlusion can set up an anodic-cathodic couple, such that metal oxidation and dissolution can occur at the anode. In addition to the requirement of a liquid medium to support the dissolution reactions and to complete the electrochemical circuit, oxygenated conditions are required at the cathode to balance anode reaction electrically and to drive the anodic reaction. After drying, the DOE standardized canisters will be filled with inert gas, and even if condensed films of water could form in the canisters, the residual oxygen levels will be insufficient to establish a sufficient potential gradient to support corrosion.

7.1.2.2. Stress corrosion cracking

SCC is a degradation mechanism in austenitic stainless steels by which contact with chloride or caustic hydroxide solutions at the site of stress results in corrosion induced cracking. Heat treatment of the canister will limit the effects of SCC to zones near the welds where stress may be introduced [4]. Both 304L and 316L stainless steels have shown excellent resistance to SCC [30]. Further, SCC is precluded because SCC does not occur below 100 °C [31]. Since water condensation does not occur until well below 100 °C in the dried canister, the formation of a condensed water film (and a caustic solution) and temperatures required for SCC are mutually exclusive. Further, even with the presence of water, concentrated hydroxide solutions are unlikely to form in the DOE standardized canisters because long term exposure of DOE spent fuels to leaching in basin storage and to atmospheric carbon dioxide, either in basin or in dry storage, has likely depleted all of the caustics in the fuel or converted them to the carbonates. Therefore only chloride solutions are of concern. As noted previously, environmental chloride concentrations are very low and are not a likely source of concern.

7.1.2.3. Intergranular corrosion and heat affected zone

Thermal sensitization of stainless steel is the reduction of the resistance of the steel towards corrosion by heat induced changes in the steel microstructure. The sensitization is the result of the thermally induced diffusion of carbon in the alloy to the grain boundaries and reaction with the chromium oxide, thereby depleting the protective chromium oxide at the grain boundary by chromium carbide precipitation. The resulting depletion of chromium can allow corrosion at the grain boundary; the process is referred to as IGC. If the corrosion is severe enough, damage can occur in the form of cracking or intergranular stress corrosion cracking (IGSCC). It is generally viewed that once the localized chromium concentration is depleted to 12% or lower by weight, sensitization begins [32]. Sensitization of 316L stainless steel is a diffusion driven process and depends on the temperature and the cumulative time of heating in the sensitization temperature region. It occurs when 316L stainless steel is heated within the range of 500 to 950 °C; occurring most rapidly at 800 °C, and becoming increasingly slower at lower temperatures, requiring exposures of about 100 hours at 550 °C [24].

Sensitization of stainless steels brought about by welding is a well established phenomenon [33–34]. The heat of welding can sensitize the areas adjacent to the welds, forming a sensitized heat affected zone (HAZ) in the steel. The area of maximum sensitization in the HAZ generally extends 1.3 to 2.5 mm from the fusion line [35]. Areas beyond the HAZ or the weld itself are not sensitized because the temperatures are either too low or high to induce chromium carbide precipitation [36]. The onset of sensitization for 316L stainless steel requires a minimum of 1 hour at any temperature within the sensitization range, with the variation in temperature affecting only the DOS [24].

The welding process for DOE standardized SNF canisters has been tested with surrogate components of similar diameter, wall thickness, and composition. The tests were instrumented with thermocouples in the HAZ to record the actual temperature histories created by the multiple weld passes [37]. The resulting maximum cumulative time within the sensitization temperature regime was approximately 500 seconds for five weld passes, or less than 10 minutes; the integrated time at temperature conservatively includes a 50% margin to cover for the potential need of additional weld repair passes. The cooling rate of the HAZ after each of the successive weld passes significantly impacts the cumulative time at the sensitization temperature. The initiation of sensitization in a 316L stainless steel pipe (0.35 in. thick) required cooling rates slower than approximately 0.1 °C per second [38]. The conservative cooling rate for the DOE canister HAZ is expected to be faster than 0.5 °C per second (300 °C range/600 seconds) [37]. Therefore, the final welding of the DOE standardized SNF canister does not represent an additional risk of sensitization, IGC or IGSCC of the HAZ.

7.2. Gas pressurization

Once the canister is sealed, the potential for over-pressurization exists. The only gases of significance, besides water, are hydrogen and stable isotope fission gases xenon and krypton [39]. Hydrogen gas production is possible by the radiolysis of water and direct metal–water reactions. The amount of hydrogen gas produced depends on the quantity of water and the mechanism. The quantity of fission gases remaining in a spent fuel depends on its total burnup, its matrix (oxide or metal), its decay time, and any diffusion losses. Although xenon and krypton are bound within the fuel matrix, diffusion from the matrix over the course of 50 years is addressed. The potential and expected pressure contributions from each of these mechanisms are summarized in Table I.

Because water vapour released during a temperature excursion is likely to be reabsorbed into the corrosion layer following the temperature excursion, the maximum long term pressures are likely those that result if corrosion processes convert water into hydrogen gas, which, in the worst case, could produce 22.7 atm (334 psia) of pressure in a canister containing only 195 L of headspace. As shown in Table I, the potential pressure rises from radiolytic hydrogen and fission gases are expected to be negligible. Even in this worst case, pressures would remain well below the 571 psig that can be accommodated without jeopardizing canister integrity. Hence, pressurization of the sealed canister will not jeopardize its ability to remain leak-tight during interim storage and transportation operations.

TABLE I. SUMMARY BY SOURCE OF POTENTIAL AND EXPECTED PRESSURE IN THE DOE STANDARDIZED CANISTER

Mechanism of Pressure Generation	Potential Pressure at 315 ºC	Expected Pressure at 315 ºC	Discussion
Thermal Equilibration from water vapour	22.7 atm (334 psia)	4.5 atm (66 psia) after drying	Section 3.2.1
Corrosion (conversion of water into hydrogen)	22.7 atm (334 psia)	<<22.7 atm (but data not available for calculation)	Section 3.2.2
Radiolysis–(decomposition of water into hydrogen)	1.88 atm (27.6 psia)	.094 atm (1.38 psia)	Section 3.2.3
Fission gases (Xe and Kr)	2.01 atm (29.6 psia)	negligible	Section 3.2.4

7.2.1. *Pressurization by water vapour in a thermal excursion*

The drying process prescribed in Section 2.2.2 is meant to limit the water content of the canister and of any corrosion product associated with the spent fuel as much as reasonably possible. Drying will remove significant quantities of water from any alumina trihydrate, but leave water in the monohydrate form behind. It is the remaining chemisorbed water ($Al_2O_3 \cdot H_2O$, monohydrate) under thermal excursion events (up to 315 °C) that may present a pressurization issue. As indicated in Section 2.2.5, if the entire corrosion product after drying consists of $Al_2O_3 \cdot H_2O$, the potential mass of chemisorbed water is 1.65 L. Even if all of the $Al_2O_3 \cdot H_2O$ decomposed completely (at lower temperatures than expected) to form 1.65 L free water, the resultant water vapour pressure at the maximum temperature of 315 °C would be 22.7 atm. This value was estimated by the Ideal Gas Law (considering P = nRT/V) as follows:

- Reaction stoichiometry: Al2O3 • H2O → Al2O3 + H2O
- Moles of water: 91.7 moles (from Section 2.2.5)
- Gas constant, R: 0.082 L-atm/mol K
- Temperature: 315 °C or 588 K
- Net void volume: 195 L (from Section 2.2.3)
- Water vapour pressure: 22.7 atm

As indicated by the sparse literature data, $Al_2O_3 \cdot H_2O$ is relatively resistant to dehydration at elevated temperatures, requiring temperatures in excess of 500 °C to dehydrate completely. The limited literature has indicated that extended drying of bayerite and gibbsite for tens to hundreds of hours at 300 °C yields end states of approximately $Al_2O_3 \cdot 0.4H_2O$ and $Al_2O_3 \cdot 0.6H_2O$, respectively [21, 40]. The documented difficulty of the complete decomposition of $Al_2O_3 \cdot H_2O$ to Al_2O_3 means that in reality, the $Al_2O_3 \cdot H_2O$ will release substantially less than its theoretical water content in heating to the canister's upper temperature design limit of 315 °C.

Because of differences in the morphology and composition that are unique to spent fuel, the dehydration of the corrosion product may behave differently than synthetic or pure material. Scrapings of the corrosion product from stored ATR fuel were analysed by X ray diffraction (XRD) and thermogravimetry [19]. The thermogravimetry was performed isothermally from 100–500 °C at 50 °C intervals with hold times of 50 hours at each temperature. The XRD analysis yielded anomalous results with no discrete diffraction peaks or phase identification, indicating that the particle sizes were extremely small. The thermogravimetric data indicated that the initial phase of the corrosion product consisted primarily of the trihydrate ($Al_2O_3 \cdot 3H_2O$) which decomposed to the monohydrate by 170°C and to $Al_2O_3 \cdot 0.5H_2O$ by 500 °C. These data indicated that the temperature interval between the minimum drying temperature of 200 °C and the transient accident temperature of 315 °C would release 0.2 moles H_2O per mole of $Al_2O_3 \cdot H_2O$, not the theoretical 1 mole H_2O per mole $Al_2O_3 \cdot H_2O$ or 1.65 L H_2O as calculated in Section 2.2.5. This means that the potential release of water vapour during a thermal transient to 315 °C could amount to as much as 330 grams H_2O and a pressure increase of 4.5 atm.

7.2.2. *Hydrogen gas production by corrosion*

As discussed in Section 3.2.1, some of the chemisorbed water in the monohydrate could be released as free water, which in turn could react with available metal in the canister to produce hydrogen. In principle, as calculated in Section 2.2.5, as much as 1.65 L water could be liberated from the monohydrate following drying. The worst scenario is the complete reaction of the liberated water with available metal to form hydrogen, which assumes that, even at 20 °C, a finite equilibrium vapour pressure develops over the monohydrate and the vapour thus developed can react with the metal in the system concurrent with regeneration of the water vapour by the remaining chemisorbed water and sustaining the reaction. This mechanism will most likely, however, be limited to the reaction of available aluminium and uranium metal [41], not stainless steel (as discussed in Section 3.1) or zircaloy [42]. (In addition, reaction with uranium metal will not result in canister pressurization, as the uranium will react with hydrogen to form the solid uranium hydride.) In this worst case scenario, if all of the chemisorbed water that could possibly remain (in the monohydrate form) in the corrosion layer after drying is released and reacted, the eventual hydrogen pressure could be as high as 11.3 atm at 20 °C or 22.7 atm at 315 °C. Again, considering the Ideal Gas Law, we have:

- Corrosion reaction: Al + 1.5 H2O → 0.5 Al2O3 + 1.5 H2;
- Available water (from boehmite decomposition): 1.65 L (or 91.7 moles, from Section 2.2.5);
- Potential hydrogen production: 91.7 moles;
- Temperature: 20 °C, or 293 K;
- Net void volume: 195 L;
- Potential hydrogen pressure: 11.3 atm at 20 °C (or 22.7 atm at 315 °C).

A more accurate (less conservative) calculation of hydrogen generation by reaction of metal with the equilibrium water vapour pressure would be possible with additional data. First, the equilibrium water vapour pressure of Al2O3 • H2O is not known as a function of temperature. Secondly, the equilibrium water vapour pressure is not known as a function of hydrate composition. While the equilibrium water vapour pressure is expected to increase substantially with temperature, it is also expected to decrease substantially as the hydrate composition of Al2O3 • xH2O decreases. The data from the thermogravimetry of the corrosion product scrapings from the ATR fuel indicated that the Al2O3 • H2O is difficult to dehydrate, decreasing to the equivalent of only Al2O3 • 0.8H2O after being heated at 300 °C for 50 hours [19]. This suggests that the equilibrium water vapour pressure is very low at this temperature and hydrate composition resulting in very slow drying kinetics.

Finally, while the reaction kinetics are available for the reaction of liquid water with aluminium metal [41], the reaction kinetics of water vapour with aluminium metal are not known, particularly as a function of water vapour pressure. If the equilibrium water vapour pressure in the canister were high enough to form condensed films on the fuel and the water vapour equilibration were not rate-limiting, then the rate of reaction of aluminium metal with the water film is fast enough to consume all of the available chemisorbed water in 50 years of storage and would generate 11.3 atm hydrogen pressure. However, drying the spent fuel at 200–250°C to a final pressure of 3 torr will ensure a very low final equilibrium water vapour pressure in the canister, well below condensable levels. The aluminium corrosion reaction rate is expected to be significantly lower at low water vapour pressure than with condensed film or immersion conditions. In fact, the residual water vapour pressure produced by the drying procedure may be low enough that the reaction rate of a subsequent water vapour-metal corrosion is inconsequential.

7.2.3. Hydrogen gas production by radiolysis

The absorption of energy from alpha, beta, and gamma radiation can break the bonds in water molecules to form a complex mixture of high energy intermediate radicals and ions of H, OH, and combinations thereof. These can react to form H2, H2O2, or reform water [43]. Molecular oxygen is not a major product of radiolysis and generally forms at concentrations several orders of magnitude less than hydrogen. The underlying mechanisms for product distribution are complex and vary with radiation types; and for the purposes of this report, gamma radiation is the only form considered. Assuming intact cladding, alpha and beta particles are effectively blocked and contribute energy only by collisions with cladding to produce X ray emissions. Neutrons with energies greater than thermal motion tend to give up energy to the hydrogen in the water molecule in the form of elastic collisions, and interaction of water with thermal neutrons can result in additional gamma emissions [44]. In a closed system with little to no head space, these radicals eventually form a steady state where approximately 20 µmol of dissolved hydrogen forms from 1 liter of pure water at 25 °C under gamma and neutron irradiation within a reactor [45–46]. The pressure created from hydrogen gas would be of little concern under these ideal pure water conditions; however, the presence of impurities, decomposition products, and a void volume for hydrogen gas expansion deviates from that ideal.

Total water content and the strength of the radiation field are the two important factors in potential hydrogen gas production from SNF. Other factors such as water purity, water form (free, physisorbed and chemisorbed), free volume for gas expansion, presence of catalytic species, and hydrogen sinks can amplify or retard the overall amount of hydrogen produced [43, 46]. Uncontrolled impurities leached from ampoule walls during testing have been the cause of inconsistent experimental data since the first measurements were done. Impurities such as halides in liquid water can increase steady state hydrogen concentration by an order of magnitude [43].

A rigorous drying procedure, which effectively removes free liquid water, can reduce the problem of accelerated hydrogen production via dissolved catalysts described by A. O. Allen [43, 46]. The goal is to minimize the sources of hydrogen production such as water vapour and physisorbed and chemisorbed water. Although the SNF drying procedure can never completely remove all water, especially chemisorbed water, rigorous drying does limit the amount of remaining water vapour and sorbed water, which may be the source of hydrogen production. There will, therefore, always be some potential for hydrogen gas production within the sealed SNF canister.

Hydrogen gas production via radiolysis of water within a SNF canister encompasses a complex set of reactions. Output of hydrogen gas can be calculated based on *G* value. The *G* value is the number of molecules created (positive) or destroyed (negative) for each 100 eV absorbed by a substance. For each roentgen of gamma rays approximately 6.08×10^{13} eV per gram of water is absorbed [43]. So if $G = 1$ for compound X, approximately 1 micromole of X is created per liter of water for every 1,000 roentgens. Pure water has a *G* value of 0.45 for hydrogen gas production, but the presence of oxides can skew the *G* values dramatically [4749]. In general, oxides may have no effect or they can enhance or retard the radiolysis of surface water. In terms of *G* values, ZrO_2 (10 to 80) enhances, Al_2O_3 (0.2) has no effect, and Fe_2O_3 (0.09) retards hydrogen radiolytic yield [49]. The exact mechanism of energy transfer from oxide to water is unclear. Unfortunately these *G* values change with the number of water layers, particle size, surface area, and morphology. In general, fewer water layers, smaller particle size, greater surface area, and, in the case of ZrO_2, tetragonal morphology (rather than monoclinic) favour increased *G* values and increased hydrogen yield.

The total output of hydrogen gas and the ultimate pressure produced within a SNF canister cannot be readily predicted. A rough calculation of total hydrogen output from the SNF canister can be performed using ATR fuel as a model, the estimated radiation field for ATR fuel, the available averaged $G(H_2)$ value for radiolysis of surface water films on Al_2O_3, and by assuming radiolysis only of the mass of water physisorbed on and chemisorbed in the alumina corrosion product layer. The calculation assumes that hydrogen production from water vapour is not significant because of the low total mass of water as vapour achieved by drying. The calculation assumes that only AlOOH ($Al_2O_3 \bullet H_2O$) remains after drying, and considering ideal Gas Law we have:

- Amount of water: 1,700 grams (assuming drying converts trihydrates to monohydrate;
- G45(H2) on Al2O3: 0.2 molecules /100 eV;
- Radiation field: 100,000 roentgens /hour;
- Time frame: 50 years or 438,000 hours;
- Total dose: 4.38 × 1010 roentgens;
- Energy conversion: 1 roentgen = 6.08×10^{13} eV /gram (value from water);
- Total energy dose: 4.53 × 1027 eV;
- Molecules of H2: 9.06 × 1024;

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[45] The $G(H_2)$ for Al_2O_3 is for surface waters only and the chemisorbed water of alumina has arguably a much lower *G* value (~0.01) [50–51].

- Avogadro's No. 6.02 × 1023 molecules/mole;
- Moles of H2: 15.0 moles;
- Temperature: 298.15 K;
- R constant: 0.082057 (L atm/K mol);
- Volume (V): 195 L;
- Hydrogen pressure: 1.88 atm. or 27.6 psi after 50 years.

This estimate is based on approximated data. Back reactions to reform water were not considered. The radiation field is an estimate based on unpublished measured data from ATR fuel elements after removal from the reactor and cooled for a couple months.

Previous reports have attempted to conservatively estimate the hydrogen gas production within these canisters [5, 39]. Without actual data, these have tended to significantly overestimate the potential pressure increase. Two recent articles indicate that very little hydrogen forms by radiolysis under the conditions expected within the canisters (see Table II). Work by Icenhour [5] demonstrated that wetted UO_2 and U_3O_8, as well as hydrated UO_3 at water loadings above our specified conditions, produced typically a 5 psi spike in pressure upon initial exposure to radiation [50, 51]. The pressure spike was followed by a fast decay to or below the initial fill pressure. Because no strongly hydriding metals are present, absorption of hydrogen gas did not appear to be a factor for the observed pressure decay. Icenhour speculated that the observed pressure decay was caused by oxidation of uranium oxide by the air backfill in the test vessel. The SNF placed into the DOE standardized canister should not experience the spike observed by Icenhour, whose samples were subjected to a very high dose rate upon placement within a fresh, hot high-flux isotope reactor (HFIR) spent fuel assembly or near a ^{60}Co gamma source.

Whether wetted UO_2, U_3O_8, or hydrated UO_3, the irradiation of wet oxide with 2.5–330 MGy produced less than 11 vol% H_2 and converted less than 0.06 mol% of the available water to molecular hydrogen. The relative radiolytic yields of hydrogen for gamma radiolysis were in the range of 0.0003 to 0.14 of the yield calculated from the nominal *G* value for liquid water.

The Icenhour experiments are considered to conservatively model the conditions in the SNF canisters for the following reasons:

- When the water concentration in the canister is expressed as the ratio of water content to net free volume, the DOE standardized canister has a residual water to free volume ratio of 0.0087 (1.700 L/195 L). By comparison, one example of Icenhour's samples, the HFIR-4 sample (3.35 mL), has a ratio of 0.093 or 0.048 depending on which vessel was used (70.5 or 36 mL) [50–51]. Therefore, the normalized water contents in several of Icenhour's tests are substantially higher than those expected for the SNF canisters and bound the SNF canisters.
- The DOE standardized SNF canister specifies a maximum exposure of 107 rad/hr [52], which is closer to the activity of an ATR fuel freshly removed from the reactor. This is slightly less than the exposure rates used in the Icenhour experiments (107–108 rad/hr, HFIR) [50–52].
- Icenhour's sample HFIR-4 was irradiated to a total dose of 370 MGy, or slightly less than 3.7 × 1010 roentgens, which is comparable to the 50-year calculated total dose of 4.38 × 1010 roentgens of fuel in the DOE canister. As an oxide, uranium does not enhance or reduce the radiolytic yield of hydrogen from surface films of water, similar to the effect of alumina [49–50].
- Lastly, the G value suggested by Icenhour for hydrogen output is 0.01, which is much lower than the estimated value for the rough calculation. In many ways, the Icenhour experiments are arguably harsher conditions than those seen by the standardized canister. The major difference is that the materials used by Icenhour are pure materials, free of the fission products found on SNF.

TABLE II. SUMMARY DATA FOR GAMMA RADIOLYSIS OF URANIUM OXIDES
(*Icenhour, Refs. 50. 51*)

Container	Material	Dose (rads)	Cover Gas	Final Pressure psi, (torr)	Final Temp. (°C)	%vol H_2	Estimated H_2 Yield mmol H_2/g Sample
S-7	U_3O_8 dry	1.8×10^8	He	Not taken	Not taken	<0.01	0
S-15	U_3O_8 with 0.7 wt% H_2O	2.5×10^8	Air/He	14.3, (738)	27	0.01	9.10×10^{-6}
S-19	U_3O_8 with 0.89 wt% H_2O	2.2×10^8	Air/He	14.1, (727)	28	0.01	5.07×10^{-6}
HFIR-5	U_3O_8 with 1.04 wt% H_2O	8.2×10^{10}	Air/He	24.5, (1,267)[a]	40	Not taken	Not taken
HFIR-6-2	U_3O_8 with 6 wt% H_2O	3.2×10^{10}	Air	14.7, (760)	49.7	1.44	1.96×10^{-3}
S-14	$UO_3 \bullet xH_2O$ with 1.8 wt% H_2O	2.5×10^8	Air/He	13.9, (718)	27	0.25	9.10×10^{-6}
S-18	$UO_3 \bullet 2H_2O$ with 2.47 wt% sorbed H_2O	2.2×10^8	Air/He	13.9, (718)	28	0.83	2.55×10^{-4}
HFIR-4	$UO_3 \bullet 2H_2O$	3.7×10^{10}	Air/He	33.7, (1,742)[a]	40	10.6	1.99×10^{-2}
HFIR-6-3	UO_3	3.3×10^{10}	Air	12.4, (641)	49.7	0.007	5.64×10^{-6}

a. Initial sample pressure about 10 psig at room temperature, operational requirement for HFIR.

More on the Icenhour data and Hydrogen gas production by radiolysis can be found in Refs [50, 51, 53].

7.2.4. *Fission gas releasen*

Fission gases xenon and krypton are also found within SNF. Stable isotopes ^{131}Xe, ^{132}Xe, ^{134}Xe, ^{83}Kr, ^{84}Kr, ^{86}Kr, and long half-life ^{136}Xe form and should not be confused with shorter lived isotopes used in curie measurements. The amount of fission gases found within various fuels depends on the matrix and also on the reactor operating temperature. Metallic fuels will not release as much xenon and krypton during relatively cooler reactor operations and would retain more fission gases than oxide fuel operating at higher temperatures. In theory, these fission gases trapped within the SNF could escape over time and increase pressure within a sealed canister, but this does not occur in practice. The MCO gas measurements showed that little fission gases were released from the fuel (~10 ppm) and as an indirect indicator, very little corrosion damage has occurred to the fuel during storage [10–12]. If it were possible to break down the fuel matrix, the total fission gas content released from high burnup ATR fuel is estimated to be 15 psi at room temperature. This pressure is based on (using the ideal gas law) a maximum 25% molar production of xenon and krypton from ^{235}U fission, a free volume of 195 L in a loaded canister, and a maximum burnup of 7.55 kg (32.1 moles) of uranium [54]. Collection of xenon and krypton during reprocessing suggest smaller amounts of these fission gases are present than calculated, closer to 10 psi [55]. Even at maximum canister temperatures of 315°C, fission gases will not be released based on safety measurements of fission product movement from high-temperature testing of metallic and oxide fuels [56, 57]. Therefore, based on the measurements of the MCOs and experimental data no significant pressure increases are expected from xenon and krypton fission gases.

7.3. Metal embrittlement

Various steel embrittlement failure mechanisms were screened for applicability based on the metals present in the DOE canisters, with particular focus on environmentally-assisted embrittlement mechanisms. Of these, the canisters were considered to be susceptible only to the environmentally-assisted mechanisms of hydrogen embrittlement and liquid metal embrittlement (LME). These mechanisms were evaluated in detail. Hydrogen embrittlement occurs when hydrogen absorbs in solid metals, which results in a decrease in toughness or ductility. Hydrogen embrittlement is strongly dependent on the amount of available hydrogen. Therefore, its potential role is determined in part by the amount of residual water that can produce hydrogen.

LME, is the interaction of liquid metals on metal and alloy surfaces. Penetration of these metals into each other creates weaknesses and increasing susceptibility to mechanical failure; the embrittlement effect is greater for steel that has been heat treated or alloyed for greater strength.

7.3.1. Hydrogen embrittlement

Metals or alloys respond differently to exposure to hydrogen gas. Steels with a body centred cubic structure are often susceptible to brittle fracture under applied stress and hydrogen exposure, whereas austenitic steels typically experience a much smaller loss of ductility. In the absence of active corrosion, damage can occur to 304L and 316L stainless steel from hydrogen stress cracking. In the presence of active corrosion and hydrogen exposure, hydrogen-assisted stress corrosion cracking is possible [58]. These degradation mechanisms would most likely occur around the final closure welds of the DOE standardized SNF canister where residual stresses remain from the heat of welding. The bulk of the canister is more resistant because it has been annealed. However, for welded seams of stainless steel 304, a pressure of 240 atmospheres (3,527 psi) of hydrogen at 200 °C was needed to produce measurable changes in strain rate or ductility. Studies done on stainless steels 316L and 304L of various grain sizes before and after sensitization have shown 316L to be more resistant towards hydrogen-assisted stress corrosion cracking, even at pressures of 10,000 psi [59]. Therefore, hydrogen embrittlement is not considered significant because the canister pressure limit will be exceeded well before hydrogen embrittlement would appear in this event.

7.3.2. Liquid metal embrittlement

Liquid metal embrittlement (LME) is the interaction of liquid metals on alloy surfaces upon wetting. Penetration of these liquid metals into alloys reduces ductility and can ultimately result in cracking and fracture below normal yield stress. The mode of attack varies depending on the species of metals; penetration can occur either transgranularly (through the grain), intergranularly (between grains) or uniformly. In the case of stainless steels, the attack tends to be intergranular.

The available data indicate that of the fission products, only Cs-Te mixtures, cadmium, and antimony are potential agents for the LME of steel. These agents are effective only at temperatures greater than 450 °C, well above the service temperatures of the DOE Standard Canister. Therefore, the fission products of the spent fuel will not cause the LME of the steel walls of the DOE canister, because the canister upper temperature limit (315 °C) will be well below the minimum temperatures required for LME. Additional assurance is provided by the fact that the potential embrittling agents in the fission products are isolated from the canister wall by the cladding of intact fuel. For fuel rods that were breached during reactor service or underwater storage, the exposed fission product metals are converted to the corresponding hydroxides or high melting oxides by reaction with water or air, thus rendering them ineffective in the embrittlement mechanism. While the water reaction would immobilize the cadmium and Te as insoluble (and high melting) oxides, the Cs hydroxide and antimony oxide are soluble and would have been leached from the oxidized fuel surfaces prior to emplacement in the DOE canister.

7.4. Radiation Damage

Damage to stainless steel occurs from neutron exposure. Damage from alpha, beta, or gamma radiation is relatively insignificant. High energy neutrons may impact metal atoms, displacing them from their matrix or lattice, which results in defects. Enough of these defects create weakness in the metal or alloy structure resulting in degradation of physical properties such as strength. Estimates for the first signs of changes in tensile strength of stainless steel range from a total exposure of 10^{18} to 10^{22} neutron/cm^2 with energies greater than 1 MeV [60–61]. Within the canister, spontaneous fission of isotopes may produce 10^7 neutrons/second per fuel assembly. This number is estimated based on the measured neutron flux from similar fuel, with an order of magnitude increase to add conservatism [62]. For 30 assemblies per canister, it is calculated that over 7 million years would be required before radiation damage affects structural performance of the canister. Hence, radiation damage to the stainless steel is not expected based on the limited neutron flux from the SNF.

Dimensions of canister: r = 22.85 cm, h = 457.0 cm
Surface area of canister: $A = 2\,\pi r^2 + 2\,\pi rh = 68{,}890\ cm^2$ (internal ≈ external)
Neutron flux from SNF: 10^7 neutrons per second • 30 fuel assemblies = 3×10^8 neutrons/sec
Damage point of Steel: 10^{18} neutrons/cm^2
Canister neutron rate: 3×10^8 neutrons/68,890 cm^2 sec ≈ 4400 neutrons/cm^2 sec
Resulting timeframe: 2.27×10^{14} sec. or 7.2 million years until damage could be expected.

Radiation can also cause organic materials such as polymers and solvents to decompose into gases such as hydrogen, methane, carbon dioxide, and acid halides, which could pressurize the canister and/or corrode its internal surface [61]. Therefore organic materials are prohibited within the canister.

8. CONCLUSIONS AND RECOMMENDATIONS

Adequate drying of the fuels is the key to assuring that sealed DOE standardized SNF canisters can be safely stored for extended periods and subsequently transported and handled. Based on the analyses provided and on literature findings, this analysis concludes that there are no credible degradation mechanisms that would significantly degrade canister performance during a 50-year period of interim storage and subsequent transportation and related operations, provided proper heated drying is conducted before sealing and that temperatures of sealed canisters remain below the drying temperature. This analysis also concludes that any pressurization by residual water or hydrogen gas, sufficient to jeopardize the performance of the canister, is highly unlikely over the 50-year period, contingent upon implementation of confirmatory monitoring of the pressure within a representative sample of sealed canisters over 10 years or more.

Because limited information exists regarding chemistries of corrosion products on SNF and also because there appear to be some inconsistencies in the available data, this analysis recommends future work to evaluate possible long term increases in free water or physisorbed water content via equilibration or dehydration/decomposition of chemisorbed water and whether this water is of any consequence. Future work should determine:

– The amount of free water vapour that can be produced by thermal equilibration of hydrated corrosion products in a closed system.
– The rate of hydrogen production (pressurization) by corrosion of remaining aluminium cladding with the free water vapour generated by thermal equilibration of hydrated species, particularly at the high temperatures.

These results will confirm that free water content will remain sufficiently low to prevent degradation and/or over pressurization that could jeopardize the leak tightness of the DOE standardized SNF canister and may also provide a basis for relaxing the temperature and time constraints associated with the conclusions given above.

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DUAL PURPOSE CASK FOR DRY STORAGE OF RESEARCH REACTOR SPENT FUEL IN LATIN AMERICA

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Abstract

Since 2001 Brazilian researchers have participated in a regional initiative, with researchers from other Latin American countries whom operate research reactors, to improve the regional capability in the management of spent fuel elements from these reactors. A dual purpose cask for transport and storage was selected as the best option for the long term dry storage of this material, and a half-scale model was designed, built and tested. Although the model failed the tests, its overall performance was considered very satisfactory and design and constructive features were changed as a result of the tests. A new test sequence with the modified cask model was scheduled for the first quarter of 2010.

1. INTRODUCTION

Since the beginning of this decade three Latin American countries whom operate research reactors – Argentina, Brazil and Chile – have joined forces to improve the regional capability in the management of spent fuel elements from these reactors (Mexico and Peru participated in the initial phase of the programme). A main drive for this initiative, sponsored by the International Atomic Energy Agency, was the fact that has been no definitive solution regarding the back end of the research reactor fuel cycle made by any one of the participating countries. Although all three countries have in the past repatriated Latin American origin fuel to the United States (Table I), they have in-house built elements or elements from other suppliers whose fuel cannot be traced back to the USA and therefore do not qualify for repatriation.

TABLE I. SPENT FUEL ASSEMBLIES SHIPPED BACK TO THE USA [13]

Country	Year	Number and type of spent fuel	Total
Argentina	2000	207, MTR HEU	**249**
	2007	42, MTR HEU	
Brazil	1999	127, MTR HEU and LEU	**160**
	2007	33, MTR HEU	
Chile	1996	28, MTR HEU	**58**
	2000	30, MTR HEU	

Therefore, a proper management programme to guarantee the maintenance of fuel integrity for a long storage period, with adequate shielding, assurance of sub-criticality, which conformed to internationally accepted safety, security and safeguards standards, was sought to be achieved at a regional level. The main activities developed in the scope of this initiative were the characterization of the research reactor spent fuel and the analysis of feasible options for spent fuel storage. The characterization studies focused on the identification of spent fuel corrosion mechanisms under wet storage conditions, spent fuel burnup determination, visual inspection, sipping tests and the establishment of a data base for the spent fuel types prevalent in the region. As for the long term storage options, it was widely recognized that, instead of the currently options used for storage of research reactor spent fuel in pools, dry storage is the most adequate option, given the local conditions and, amongst the different options for dry storage, storage in metallic casks was deemed the safest and most cost-effective. Considering this, it was decided to develop a cask for the transport and storage of spent fuel from research reactors, starting with the design of a half-scale model that was built in Argentina and tested in Brazil. Although the model failed the tests, its overall performance was considered very satisfactory and design and constructive features were changed in light of the test results. A new test sequence with the improved cask model was scheduled for the first quarter of 2010.

2. CASK DESCRIPTION

The cask was designed to meet transportation and long term storage criteria. On one hand, the cask has to be sturdy enough to resist transport accidents, a free drop from 9 m; a one-meter high fall onto a flat pin; a fire yielding a temperature of 800°C for 30 minutes, and submersion to a 15 m depth of water. On the other hand, its materials of construction have to offer long term stability and compatibility (between them and the radioactive contents) and there needs to be access to its internal cavity, for periodical checks such as gas sampling, draining or pressurization.

In addition to the above criteria, the cask design features were constrained by the characteristics of the research reactor facilities in the participating countries. Therefore, the maximum cask weight, when loaded, was established as 10 metric tonnes and the maximum external diameter (to allow in-pool loading) as one meter. Likewise, as the main research reactor types operating in Latin America are MTR and TRIGA reactors, the cask was designed with a storage capacity to accommodate interchangeable baskets either 21 MTR spent fuel assemblies or 78 TRIGA spent fuel elements.

The cask consists of a sturdy cylindrical body provided with an internal cavity to accommodate the basket that holds the spent fuel elements. The cask body has a sandwich-like shielded wall consisting of stainless steel outer and inner surfaces and lead inbetween. A double lid system guarantees the required containment and the internal lid (which is part of the pressure boundary) is provided with two concentric metallic seals. The structure of the basket is made of square tubes. For protection against mechanical and thermal loads, the cask is provided with top and bottom external removable impact limiters. These are structures made of an external stainless steel skin and an energy-absorbing filling material. The filling material chosen was an wood composite named Oriented Strand Board (OSB), which is an engineered, mat-formed panel product made of strands, flakes or wafers sliced from small diameter, round wood logs and bonded with a binder under heat and pressure. A schematic view of the cask is shown in Fig. 1.

As part of the dual purpose cask design, both shielding and criticality analyses were performed for the most reactive MTR and TRIGA fuel elements in the region and for the most conservative assumptions likely to be encountered [4] . For the shielding analysis, the gamma and neutron sources were estimated considering a burnup of 50% (235U) for the MTR fuel and 25% for the TRIGA fuel. Furthermore, a 5-year cooling time was adopted in the model. The QADS and SAS4 modules of the SCALE 4.4A package [5] were used in the calculation. The maximum values found for surface dose rate were 186 and 367 μSv/h in the radial and axial directions, respectively. These values are well below the limits established by IAEA standards, which are 2,000 or 10,000 μSv/h, depending on whether the transport is under exclusive use [6].

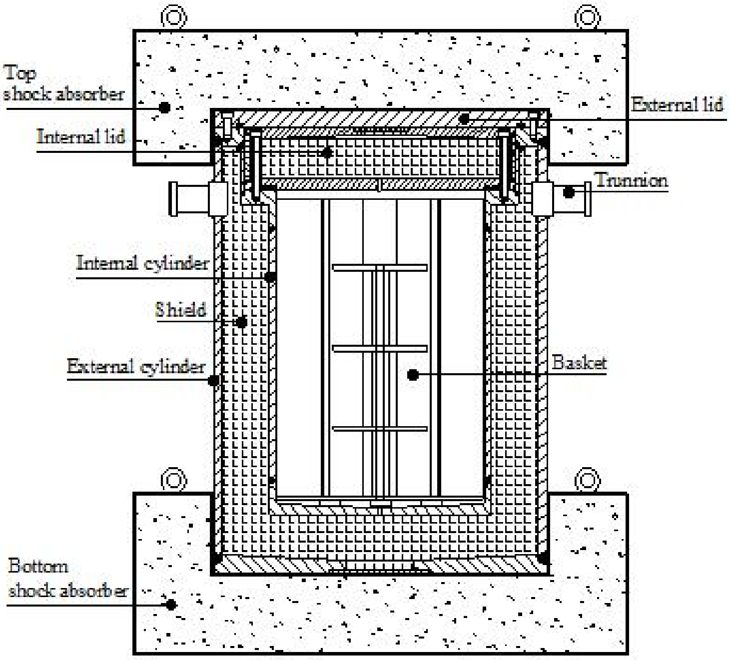


FIG.1. Dual purpose cask for dry storage of research reactor spent fuel.

The criticality safety analysis of the cask was carried out using the Monte Carlo transport code MCNP4B [7]. It was conservatively assumed that the fuel elements were fresh, i.e. no burnup was considered, that the cask was completely submerged in water (water inside the cavity and surrounding the cask) and the calculated neutron multiplication factor (k_{eff}) required to be, for both normal and accident conditions, lower than 0.95. The maximum values found were 0.87165 ± 0.00070 for TRIGA elements and 0.89890 ± 0.00082for MTR [4], demonstrating that the sub-criticality condition is guaranteed for the cask loaded with any one of the two types of fuel elements. Thus, the fuel elements can be stored and transported safely even in the hypothetical situation of the cask being completely flooded.

3. CASK MODEL DESIGN, FABRICATION AND TESTING

3.1 Cask model design

As previously stated, the cask is required to resist accident conditions that may occur during transportation without permitting any dispersion of its radioactive contents or a significant increase in its surface dose rate.

To verify the compliance of the proposed cask with this requirement, a half scale model for testing purposes was designed and manufactured according to an established Quality Assurance (QA) Programme. As part of the QA Programme, technical specifications containing the requirements for the manufacture of the cask model were developed. The design followed basically the guidelines of the Oak Ridge National Laboratory's *The Radioactive Materials Packaging Handbook* [8] and the ASME B & PV Code, Section III, Division 1, Subsection NB [9]. The items most relevant to safety were calculated analytically. The model's external wall thickness was calculated to resist the penetration of a flat pin, whereas the cavity wall thickness was verified against buckling due to the radial pressure resulting from the contraction of the lead due to its cooling during the model fabrication. Also the internal lid, which is part of the cask's containment system, was dimensioned to resist an internal cavity pressure of seven bars, established as the maximum operation pressure. The verification included also the 24 bolts connecting the lid to the model's main body. Another critical feature was the model's welding parameters and quality. All welds were designed according to the above mentioned ASME Code.

3.2 Cask model fabrication

The cask model was manufactured in Argentina and shipped to Brazil for testing. The main milestones in this phase were:

- Material procurement: Aug–Sept 2005;
- Manufacturing start: October 2005;
- First inspection by client (manufacture authorization): 17–21 October 2005;
- Lead shielding pouring: 7 December 2005;
- Manufacturing end: March 2006;
- Second inspection by client (acceptance tests): 21–25 April 2006;
- Shipping to Brazil: May 2006.

The model was submitted for acceptance tests at the manufacturer's premises. The tests, detailed in the technical specifications for fabrication, consisted of gamma scanning test, helium leakage test and a pneumatic pressure test. The test sequence was designed to establish if that the model shielding and containment features met the required quality, that the model was leak tight, and that its mechanical strength met the specifications. Figs 2 to 4 show the cask undergoing the acceptance tests.

FIG. 2. Gamma scanning test.

FIG. 3. Helium leakage test.

FIG. 4. Pneumatic pressure test.

3.3 Cask model testing

The cask model was subjected to the sequence of tests prescribed at the IAEA´s standard for the safe transport of radioactive material [6]. The tests were carried out to simulate both normal conditions of transportation (NCT) and hypothetical accident conditions (HAC). The first group of tests consisted of a free drop from 1.2 m height and a penetration test. The prescribed water spray and stacking tests were not performed, the former were deemed to be irrelevant for the tested item and the latter because, due to its geometry, the cask is not stackable.

The tests to simulate HAC were water immersion test, puncture test onto a vertical bar, free drop test from 9 m height, and thermal test. The latter was carried out only for design development purposes, since scaled models are normally not submitted to real thermal tests. The practice recommends that if it is not possible to carry out thermal tests in a natural scale prototype, due to high cost or lack of a proper infra-structure, then numerical simulations should be used, since the thermal phenomena that occur during the test are not reproducible in a scaled model.

The approval criteria established in the regulations in terms of contamination release rate were translated into gas (helium) leakage rate. A maximum helium leakage rate of 10^{-6} mbar l / s was established as an approval value. An initial helium leak test was carried out prior to any other test and the cask model was found to be leak tight.

3.3.1 Tests to simulate NCT

A series of three 1.2 m drop tests were carried out in different positions, since no previous experience existed to ascertain which would be the most critical drop condition. The drop positions, shown in Fig. 5, were:

- Slanted, CG-corner aligned in vertical (a);
- Horizontal on cask body (b);
- Vertical on lid end (c).

(a) CG-corner *(b) Horizontal on cask body* *(c) Vertical on lid end*

FIG. 5. 1.2 m drop test positions.

The visible damage of the CG-corner and horizontal drop tests were dents to the impact regions of the shock absorbers and a slight bulging of each impact limiter's flat surfaces, whereas the vertical drop caused no visible damage on the tested model. As shown in Fig. 6, the maximum g-level (approximately 33) was registered during the CG-corner drop. It shall be mentioned that, due to the scaling physical laws, the acceleration levels in the natural scale cask are expected to be half of the values observed in the present case.

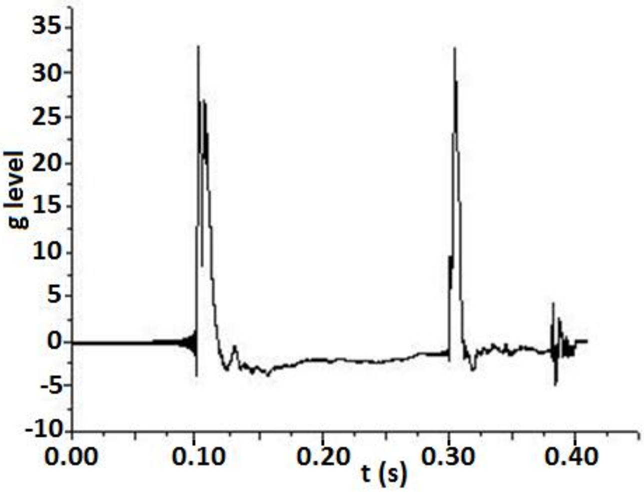


FIG. 6. Acceleration levels during the 1.2 m CG-corner drop test.

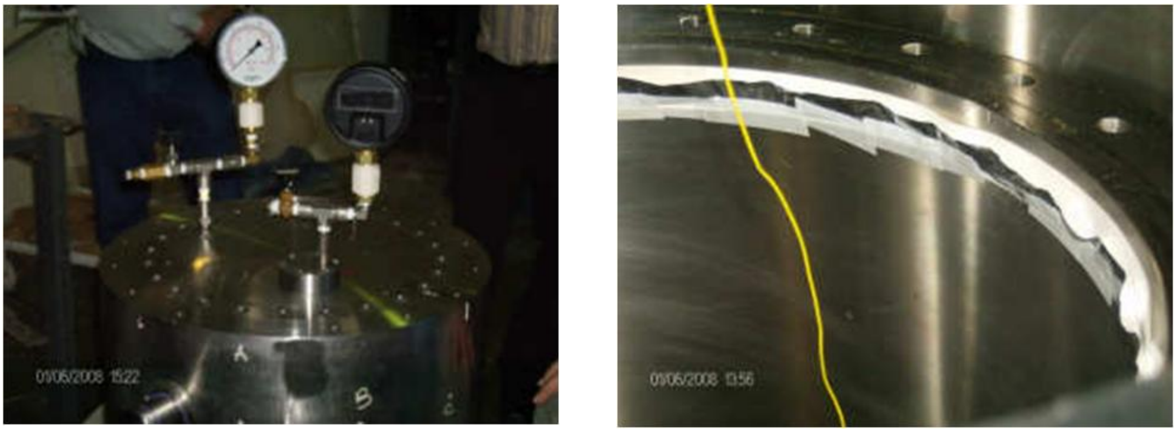
The penetration test consisted of an impact of a 6 kg steel bar striking the most vulnerable region of the model, caused only a negligible dent at the impact point.

3.3.2 Tests to simulate HAC

The tests to simulate hypothetical accident conditions were water immersion test, puncture test onto a vertical bar, free drop test from 9 m height, and thermal test.

The water immersion test was not carried out by physically submerging the test model in water neither placing it inside of a pressure vessel. Instead electronic circuits that could detect the presence of water by sensing the variation of the electrical resistance were mounted around the model´s most critical parts, which are its lid and ports, as shown in Fig. 7.

The test consisted in applying a hydraulic pressure of 150 kPa to the annular space between, respectively, the seals of the internal lid and the pressurization and draining ports. Any water intrusion would reduce the electrical resistance of the circuit, which was open (infinite resistance) in the absence of water.



(a) Manometers for hydraulic pressure recording *(b) Electric circuit at the top rim of the cavity*

FIG. 7. Setup for immersion test simulation.

The cask failed this test, possibly due to lack of gasket compression deformation.

The puncture tests consisted of releasing the cask model twice onto a vertical metallic bar from 1 m height. In the first test, the drop angle was selected to make the bar impact the top impact limiter in an attempt to remove it. As a result, the targeted impact limiter was torn open at the welded joint near the point of impact but it was not removed The second drop test targeted flatly the cask body side, causing an indentation with a depth of 11.8 mm to the cask body at the point of impact. However, the cask body outer wall suffered no tearing. Figure 8 shows the results of the puncture tests.

FIG. 8. (a) Cask model after the puncture tests showing: damage due to the first puncture test;
(b) Indentation on cask body side due to the second puncture test.

The free drop tests from 9 m height were conducted in the same positions as the previous drop tests. However, due to the damage sustained by the shock absorbers, the most affected shock absorbers were replaced after each drop. As shown in Fig. 9, the different damage sustained by the shock absorbers were indentation along the impacted edges; bulges on the surfaces near the impact regions; opening of the most stressed weld seams; and tearing of the impact limiter skin due to impact of the lifting trunnions. Also a bending of the tensor bars which connected the top and bottom shock absorbers was noted. As shown in Fig. 10, the maximum g-level recorded was 272 g, and half of this value is expected to be observed in the natural scale cask.

(a) Opening of welds *(b) Bulges and skin tear* *(c) Indentation of impacted dges*

FIG. 9. Damages from the 9 m drop tests.

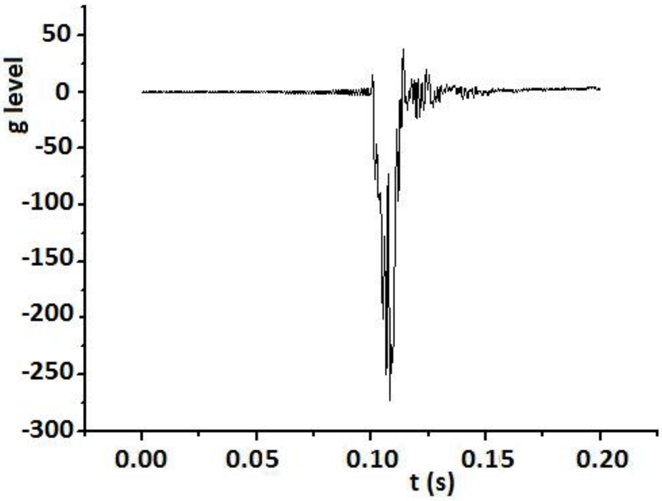


FIG. 10. Acceleration levels due to the 9m upside down drop test.

The thermal test consisted of submitting the test model for 30 minutes to a temperature of 800 ^{o}C. The test was carried out in an industrial electrical furnace preheated to 804 ^{o}C and the model was fitted with temperature indicating labels and thermocouples. The labels were installed in different positions inside the cask cavity, including the cavity walls, basket and some of the dummy fuel elements. The thermocouples were mounted in the following positions: two inside holes existing on the external lid; two inside the cask cavity (through especially designed passages); and two inserted into especially designed passages through the cask body wall (one inserted 1/3 deep on the wall thickness and one 2/3 deep on the wall thickness). The two last sensors were meant to record the temperatures of the lead during the test.

After heating in the furnace for 30 minutes, the cask was removed and suspended in air for about 10 minutes and then laid to rest on thermal insulating pads. The maximum temperatures indicated by the thermocouples at each monitored position were:

- External lid: 395 oC;
- Cask wall, 1/3 thickness: 436 oC;
- Cask wall, 2/3 thickness: 405 oC;
- Cask cavity: 254 ^{o}C.

The maximum temperature range registered by the labels on the basket and dummy fuel elements was 163 ^{o}C < T < 177 ^{o}C.

These values indicate that, although the lead might have melted (a suspicion corroborated by a bulge observed on the bottom of the cask, as seen in Fig. 11), the cask wall as a whole provided good thermal isolation, preventing the fuel elements from reaching excessive temperatures.

FIG.11. Bulge on the bottom of the tested cask model.

3.3.3 Final leak test

A final leak test was carried out to verify the cask model containment capacity. A helium leak test was initially performed, but it was soon evident that the model had gross leakages. The helium test equipment was disconnected, a compressed air line attached to the cask and a soap bubble test performed. Leaks through the lid rim were then observed, as shown in Fig. 12, and the test interrupted with unsatisfactory results.

FIG. 12. Leaks detected during the final bubble test.

4. CONCLUSIONS AND FUTURE ACTIONS

It has been demonstrated that the overall cask design is robust and has good mechanical strength. Despite having been subjected to a total of nine drop tests, no noticeable structural failure occurred. The only significant plastic deformation sustained by the tested model was produced by the impact against the vertical bar during the second puncture test, which did not affect the internal basket or the dummy fuel elements. The visual inspection after the tests showed that neither the basket nor the dummy fuel elements experienced distortions or displacements; therefore, the configurations used for the criticality safety analyses were adequate.

As for the shock absorbers, the deformations, distortions and other damages sustained corresponded to the expected scenario. After all, they are sacrificial items intended to absorb energy during the impacts and protect the cask´s main parts.

Some punctual improvements however need to be implemented in some aspects of the cask design. The shock absorber shape has to be modified, basically replacing its square external edge by a chamfered edge, thus avoiding high stress concentrations during the impact. Also the orientation of the OSB boards in the central region of the shock absorber has to be changed from horizontal to vertical (considering the cask in its normal vertical position). During the upside-down drop test (the one that resulted in the highest acceleration values) more energy will be absorbed by the shock absorber, less rebound effects will be experienced by the cask body and, consequently, less load will act upon the cask lid bolts.

A final important modification is the replacement of the used primary lid bolts material by a similar material with higher mechanical strength.

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DRY STORAGE OF SPENT FUEL DISCHARGED FROM RESEARCH REACTORS IN CANADA

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Abstract

Atomic Energy of Canada Limited (AECL) manages the spent fuel discharged from most of Canada's research reactors. These research reactors have been operated to support Canada's nuclear R&D programmes and medical isotope production. The spent fuel inventory consists of dozens of types and configurations, including intact and non-intact fuels and totalling approximately 95 MTHM. The fuels are a mixture of both high and low enrichments, and are typically aluminium-clad with various fuel core materials, including uranium metal, uranium dioxide, uranium-aluminium alloy, and uranium silicide-aluminium dispersion fuels. The discharged spent fuels are initially wet stored at reactor fuel bays for approximately two years before being transferred to dry storage facilities, at AECL's Chalk River Laboratories site and Whiteshell Laboratories site. After 50–100 years of interim dry storage, the spent fuel will be sent to permanent geological disposal as part of Canada's long term management programme for spent nuclear fuel. This paper presents the strategy and practices with regard to the management and storage of spent research reactor fuel in Canada.

1. INTRODUCTION

The Canadian strategy for managing the spent nuclear fuel has been to provide safe interim storage at the reactor site using a combination of pool and dry storage technology, and in parallel to develop the technology for disposal [1]. When the spent fuel is discharged from the reactor, it is first transferred to a pool system. The water in the pool removes the residual heat produced by the spent fuel and provides radiation shielding for workers. After a few years in wet storage, the fuel's activity and rate of heat generation have decreased sufficiently to allow the fuel to be transferred to dry storage, if desired. Compared with extended wet storage, dry storage is considered to be more efficient. Dry storage facilities generally remove decay heat by passive cooling and have lower operating costs. They also provide the advantage of incremental storage capacity expansion by allowing additional storage capacity to be constructed on an as-needed basis.

2. RESEARCH REACTORS IN CANADA

There are currently eight operating and six shutdown research reactors in Canada, as summarized briefly below. Three other reactors have been decommissioned, SDR-Slowpoke Demo reactor (first critical in July 1987, and shutdown in June 1993), Slowpoke-2 Toronto (critical in June 1971 and shut down in December 1998) and Zeep (critical in September 1945 and shut down in October 1970).

2.1. Operating research reactors in Canada

2.1.1. National Research Universal (NRU)

Located in Chalk River, Ontario, NRU research reactor is a thermal neutron, heavy water moderated and cooled reactor, with a nominal power up to 135 MWt.

During many decades it was world's highest producer of medical and industrial radioisotopes, including molybdenum-99. NRU's large irradiation space has also been an important factor in the testing of fuel bundles and fuel-channel components for CANDU reactors. NRU is used for research into reactor fuels, materials and components, and is the centre for neutron beam research in Canada. The original fuel assemblies of NRU, a 10 feet long bar, consisted of natural uranium metal flat bars clad in aluminium sheaths, with no bonding between the bar and the sheath. The irradiation history of the non-bonded fuel assemblies in NRU showed failures at an unacceptable rate, requiring the unbounded fuel to be replaced by a recently developed fuel assembly obtained by cladding the flat uranium bars with aluminium by extrusion [2]. Figure 1 shows a sketch of the original NRU flat type uranium rod.

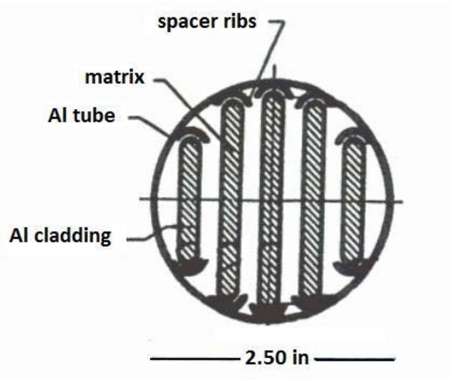

FIG.1. Cross section of original flat type NRU natural uranium rod [2].

Originally designed for operation with natural uranium metal fuel, in 1957 NRU started a conversion process to use an alloy made of 93% U-Al high-enriched uranium (HEU). The transition core included some annular assemblies, until reaching the final equilibrium core, made of 12 pin assemblies as shown in Fig. 2. The conversion process from natural uranium to HEU was completed in 1964 [3]. In the 80´s AECL joined the international efforts to reduce HEU fuel utilization (RERTR Programme), and started a programme to reduce the enrichment of NRU fuel. U_3SiAl with aluminium cladding was the selected fuel, and gradual conversion from HEU-Al fuel to LEU fuel began in 1991 [4, 5]. The majority of HEU fuel irradiated in NRU was historically sent to the United States, with the remainder being dry stored at CRL. Initial storage of the spent fuel rods is in water filled bays located within the NRU. After an appropriate time to allow for radioactive decay and cooling, the spent fuel is dry stored in tile holes at Waste Management Area ‘B’ at Chalk River.

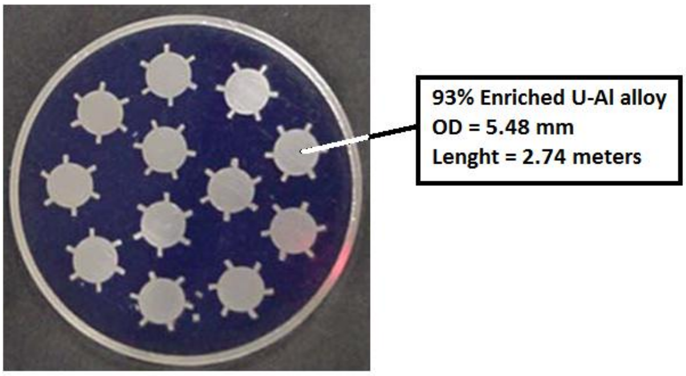

FIG.2. Configuration of HEU assembly fuel used in NRU research reactor.

2.1.2. Zero Energy Deuterium-2 (ZED-2)

Also located in Chalk River, Ontario, ZED-2 is a D2O-moderated critical facility, that started operation in 1960, and which is nowadays operated occasionally at very low power (~200 W).

The reactor as shown in Fig. 3, is a large cylindrical aluminium tank (calandria) with 336 cm ID and 334 cm deep, surrounded by graphite reflector, 60 cm radially and 90 cm on bottom. Fuel rods and/or channels are suspended from steel beams at top of calandria at various spacings: 28.6 cm for CANDU and 24.0 cm for ACR-1000. Criticality of the reactor is achieved by adjusting the moderator level within +/- 0.2 cm. The fuel channel consists of calandria tube, pressure tube, fuel assembly and coolant (D2O, H2O of air). The facility was built to test the fuel arrangements of Canada's power plants [6], and since its inauguration, ZED-2 has supported the development of the CANDU industry by testing a wide range of fuel bundle designs and fuel arrangements at low power (usually between 5 to 100 watts). Over the years, experiments included the early seven element fuel used in NPD, the 19 fuel element of both NPD [6] and Douglas Point, and the subsequent larger diameter fuel elements used in the commercial CANDU reactors, shown in Figure 4. Test fuel included natural uranium fuels and Slightly Enriched Uranium (SEU) fuel, with enrichment between 0.9 and 1.2%, like the Low Void Reactivity Fuel (LVRF) [7] and CANFLEX 43, also shown in Fig. 4.

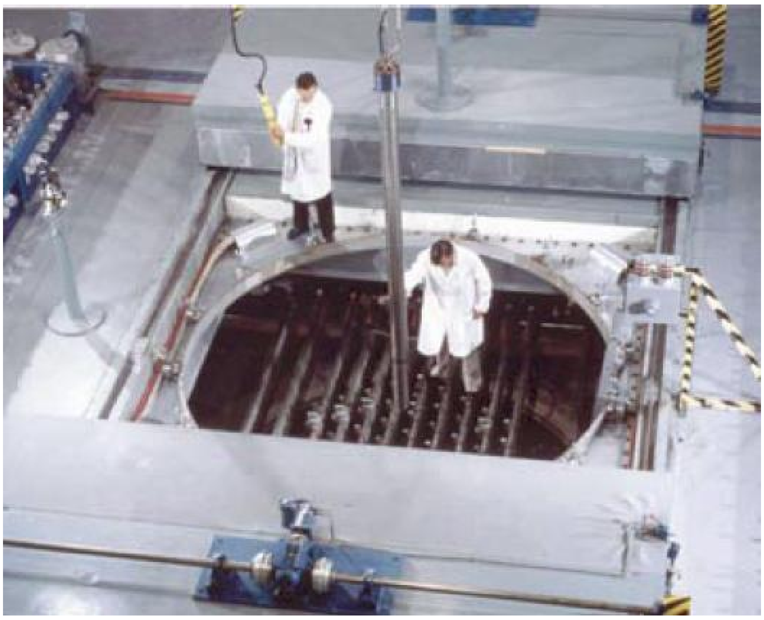

FIG. 3. Top view of ZED-2 research reactors [8].

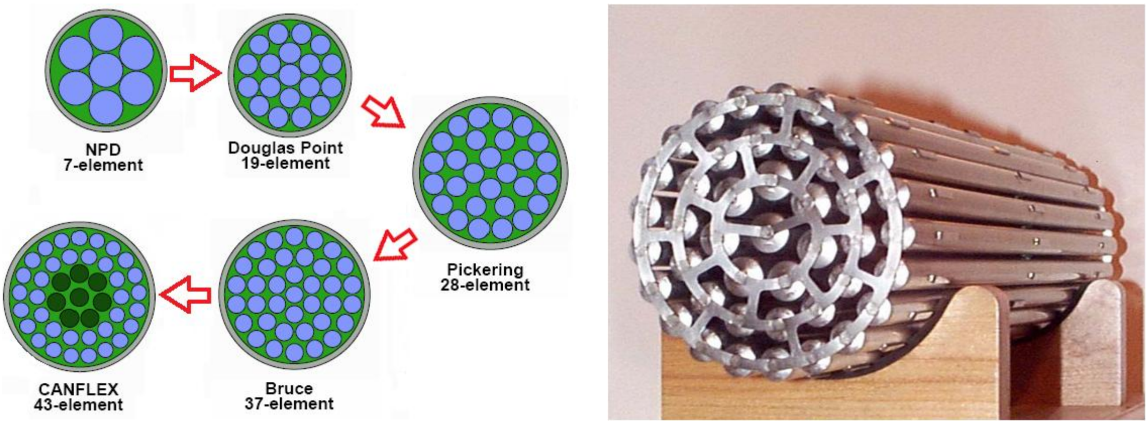

FIG. 4. CANDU fuel bundles used in ZED-2 (left) and the CANFLEX 43 fuel bundle (right) [8].

ZED-2 continues to operate today, actively supporting prototype testing of fuel to determine fuel characteristics. The fuel includes natural uranium, slightly enriched uranium, and low enriched uranium as well as natural uranium metal clad in Al. ZED-2 is also used to calibrate neutron detectors for use in power reactors, and the spent fuel, with near-zero burnup, is dry stored in air at the reactor site.

2.1.3. McMaster Nuclear Reactor (MNR)

Located in McMaster University, Hamilton, Ontario, the MNR is a pool-type reactor moderated and cooled by light water, which operates at powers up to 5 MW.

MNR uses plate type fuel assemblies. Standard fuel is an 18-plate fuel assembly composed of 16 fuelled plates and two, outer, solid aluminium, "dummy" plates. Control-fuel assemblies contain nine fuelled plates, with a central space replacing the nine removed plates, to accommodate an absorber rod. An alternative design used as standard assembly in the early years of the reactor was a 10-plate fuel assembly, with thicker plates and coolant channels. The plates have some curvature in radius and are supported by aluminium sideplates. The plate dimensions of the control-fuel are identical to those of the standard assembly. Figure 5 shows the sketch of the standard 18-plate and control fuel assemblies of MNR, and Figure 6 shows a cutaway view of a standard fuel assembly.

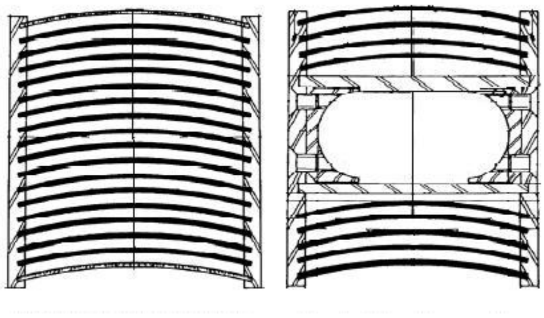

FIG. 5. Standard and control(left) (right) fuel assemblies of MNR research reactor [9].

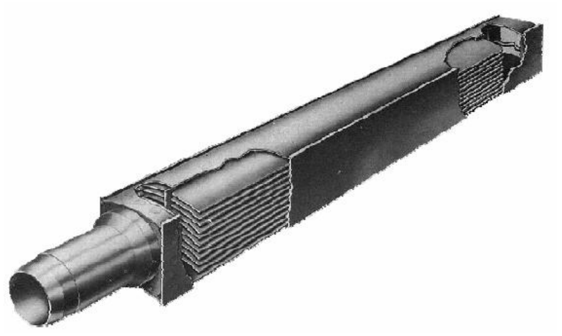

FIG. 6. Cutaway view of standard fuel assembly of MNR research reactor [9].

The original MNR fuel was HEU, nominally 93% enriched with fuel meat UAlx-Al or U3O8-Al, depending on the manufacturer. The conversion process of MNR research reactor started in January 1999, when the first LEU fuel, 19.75% enriched, was installed in the reactor core. It was a gradual process with replacement of spent HEU assemblies with fresh LEU assemblies [9]. Full LEU loading occurred in April 2007 [10]. The meat of the LEU fuel is U3Si2-Al. The LEU standard and control fuel assemblies are geometrically identical to the HEU standard and control assemblies, respectively. After full conversion of the reactor, the remainder of the HEU fuel and other irradiated material as well as the new LEU fuel is initially stored under water for a period of time before being sent to the United States, or to Chalk River Laboratories for storage.

2.1.4. Safe Low Power Critical (K) Experiment (SLOWPOKE-2)

Actually there are five SLOWPOKE-2 research reactors in operation, one in the Royal Military College of Canada; in Ontario; the second in the École Polytechnique of Quebec; the third one in Alberta University, in Alberta; the forth one in Dalhousie University, in Nova Scotia, and the fifth in the Saskatchewan Research Council, in Saskatchewan. The SLOWPOKE-2 research reactor is a low-energy, pool-type nuclear research reactor, designed by AECL in the late 1960s, for small scale radionuclide production, neutron activation analysis (NAA), research, and teaching. Most SLOWPOKE-2 reactors are rated at a nominal 20 kW, although operation at higher power for shorter durations is possible. The original SLOWPOKE-2 uses 93% enriched uranium in the form of U-Al alloy, with aluminium cladding. The core is an assembly of 296 fuel pins, only 22 cm diameter and 23 cm high, surrounded by a fixed beryllium annulus and a bottom beryllium slab. In total, the reactor has about 1 Kg of 93% HEU. Criticality is maintained by adding beryllium plates in a tray on top of the core, to compensate the fuel burnup. In addition to passive cooling (natural circulation), the reactor has a high degree of inherent safety (high negative void and temperature coefficient). These characteristics are so dominant, that the SLOWPOKE-2 reactor is licensed to operate unattended overnight (but monitored remotely). The HEU reactor core was designed to last the time equivalent to the life of the reactor, about 20–25 years, which means that there was no provision for refuelling. At the end of its life, the complete core is removed and the spent fuel is either sent to AECL CRL for dry storage or to the United States. The original core of the Slowpoke-2 research reactor and its supporting structure are shown in Figure 7.

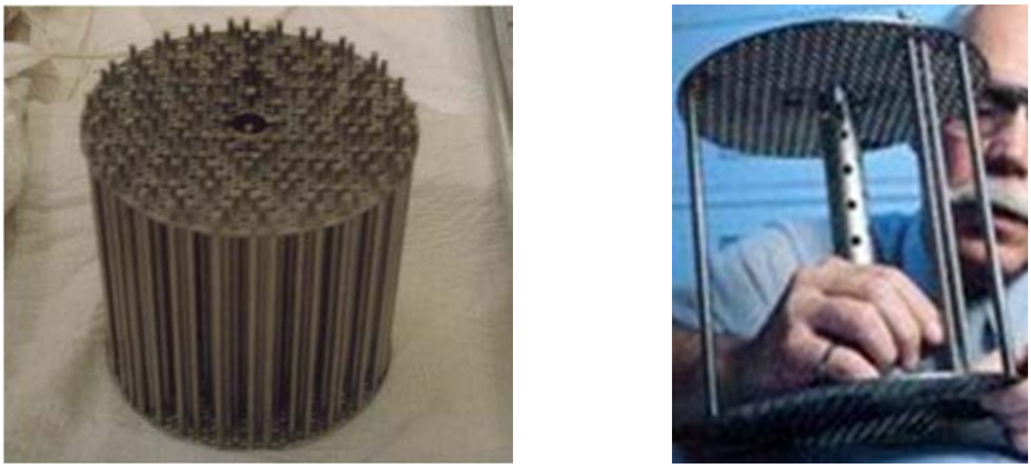

FIG. 7. Core of SLOWPOKE-2(left) and its supporting structure (right).

In the 80´s, following the tendency to decrease the use of HEU fuel, AECL started studies to convert the core of the SLOWPOKE reactor to LEU fuel, and in 1985 the first LEU fuelled SLOWPOKE-2 reactor was commissioned at RMC in Kingston, using 198 pins of Zircaloy-clad fuel, with 20% enriched UO2, similar to the fuel used in CANDU reactors. Some modifications made for the conversion process allowed to extend the life of the reactor. The reactor at Ecole Polytechnique, Montreal that had been installed in 1976 and operated with HEU fuel for 21 years, in 1997 showed no remaining excess reactivity to compensate the fuel burnup. The exhausted fuel was removed and replaced with LEU fuel in the same beryllium reflector, making it essentially identical to the reactor at RMC. The lifetime of the two new reactors are expected to be about 40 years [11].

The SLOWPOKE-2 reactor at Dalhousie University is expected to be shutdown shortly.

2.2. Shutdown research reactors in Canada

2.2.1. National Research Experimental Reactor (NRX)

Located in Chalk River, Ontario, NRX was a heavy water moderated, light water cooled, nuclear research reactor which came into operation in 1947 at a design power rating of 10 MW (thermal), increasing to 42 MW by 1954.

Original fuel used in NRX reactor was made of uranium and thorium metal solid cylindrical rods clad in aluminium, 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, as used in NRU [12]. NRU also used uranium oxide “long rod” fuels, either solid or annular cylindrical pellets clad in aluminium and also approximately 3.35 m long.

For the first 15 years of NRX operation natural uranium metal rods were the standard fuel. These consisted of a solid uranium-metal core 1.36 inches (34.43 mm) in diameter and 10 feet (3 048 mm) long. The uranium was clad with a drawn aluminium sheath. The clad uranium was inserted into an outer aluminium coolant tube to provide a water cooling annulus. The fuel showed some instability during irradiation. A decision was made to introduce a new fuel design in NRX, and 10-foot long, solid –pellet uranium oxide rods were designed and fabricated for irradiations. The sheaths were made of aluminium alloy, to provide the greater strength required by the building of pressure during irradiation. Annular pellet uranium-dioxide rods were also developed for the hotter (Higher neutron flux) positions of NRX. A seven element booster assembly was developed, to maintain the reactivity with increased loads. 93% enriched uranium metal pieces were dissolved in molten aluminium, to yield a uranium-aluminium alloy. The standard fuel in NRX was subsequently converted to enriched U-Al fuel. (7 elements per coolant tube) [2].

The NRX research reactor was retired in 1992. The majority of the fuel historically irradiated in NRX was returned to the United States, with the remainder being dry stored at CRL.

2.2.2. Pool Test Reactor (PTR)

Also located in Chalk River, Ontario, this 10 kW_{th} pool-type reactor was built at Chalk River in 1957. It used 93% enriched uranium-aluminium plate-type fuel. The reactor, which was retired from service in 1990, was used for burnup measurement of fissile (irradiated) samples from NRX.

All the spent fuel from PTR has been returned to USA under the Foreign Research Reactor fuel return programme.

2.2.3. Whiteshell Research Reactor (WR-1)

WR-1 (originally Organic Test Reactor – OTR) was located in AECL, Whiteshell, Manitoba. The WR-1 was an organic test reactor designed by AECL to demonstrate the feasibility of an organic-cooled CANDU power reactor. WR-1 operated at a power rating up to 60 MW_{th}. WR-1 achieved criticality in 1965, and was shutdown in 1985.

Original fuel of WR-1 was a bundle with 18 element pins made of uranium-dioxide clad in zirconium, in a design similar to 19-element bundle shown in Figure 4, except that the centre element was omitted so that the bundle would fit over a central support shaft. The uranium was enriched in the range 1.2 to 2.4%. Two fuel types of this general design were made, one used Zr-2.5 Nb sheathed fuel elements assembled into 19.5 inch (495 mm) long bundles with each reactor fuel string having 5 bundles suspended on the central hanger tube; the other used sintered aluminium product (SAP), sheathed fuel elements assembled into 32.1 inch (813mm) long bundles, with each reactor fuel string having 3 bundles suspended on the central hanger tube. About half of a full reactor charge of the SAP-clad fuel had been made by the time WR-1 started up, and this was eventually irradiated, but no more was made. [2]

WR-1 used the uranium-dioxide fuel from startup in 1965 until 1973, when uranium-dioxide was gradually phased out in favour of uranium-carbide fuel. About 1,100 bundles of uranium dioxide were used in WR-1. Full conversion of WR-1 from uranium-dioxide to uranium-carbide was completed by the end of 1977 [2]. The uranium monocarbide fuel, clad in Zr, is slightly (1.3 ~ 2.4%) enriched. All the spent fuel has been dry stored in concrete canisters at the WL site.

SLOWPOKE – Three SLOWPOKE reactors have been shut down, AECL's Whiteshell Laboratories, Manitoba; University of Toronto, Ontario and Nordion International Inc, Ottawa.

The two SLOWPOKE cores removed from Whiteshell, Manitoba and Nordion, Ottawa have been dry stored in waste management area "B" at Chalk River; the core of SLOWPOKE from University of Toronto, Ontario was returned to USA, under the Foreign Research Reactor fuel return programme.

3. DRY STORAGE OF SPENT FUEL FROM CRL RESEARCH REACTORS

AECL has operated research reactors at the CRL site for the purpose of nuclear research and production of radioisotopes. Spent research reactor fuels have been accumulated from operating these reactors. The fuels are a mixture of both high and low enrichments, and are typically aluminium-clad with various fuel core materials, including uranium metal, uranium dioxide, uranium-aluminium alloy, and uranium silicide-aluminium dispersion fuels [12].

These spent fuels are initially stored in water filled fuel storage bays for a period of several years before being placed in storage containers and transferred to the CRL Waste Management Area "B"

(WMA “B”), where they are stored in below grade, vertical cylindrical steel and concrete structures called “tile holes”, shown in Figure 8.

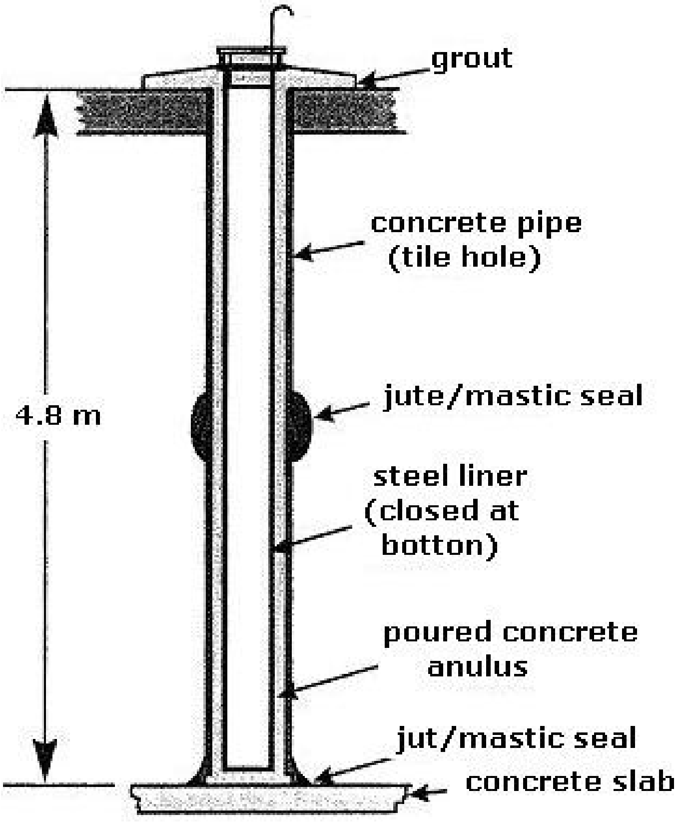

FIG. 8. Sketch of “tile holes”, a cylindrical steel concrete structure used for interim storage of research reactor spent nuclear fuel in Waste Management Area B of Chalk River Laboratory.

These tile holes are constructed in arrays, on a concrete pad foundation located at the bottom of a trench. The bottom of the trench is at least 1.2 m above the water table. Two concrete tiles are stacked vertically, and sealed at the joint. A mild steel liner with closed bottom is inserted, and the annulus is backfilled with concrete. The liner tube has a gasket sealed flange, which is closed by inserting a vented shield plug. The array structure is packed with sand between the tile holes, and capped with a concrete pad.

During the process of storage, the spent fuel rods are placed under water, in steel containers that may contain one or more rods. The containers are typically an open steel cylinder. After the fuel is loaded, the containers are drained of water and then loaded into the tile holes.

Tile holes storage structures are engineered to shield the radioactivity of the spent fuel, dissipate heat and to contain any contamination, and thereby prevent the spread of radioactivity into the environment. An ongoing physical inspection programme and the Groundwater Monitoring Programme have confirmed that the tile holes continue to meet their design intent [13].

However, some of the fuel in early designed tile holes, which were built and loaded in the 1960’s and 1970’s, is known to be degrading due to ingress of moisture. Monitoring and inspection of these older fuel types have shown that some of the fuel storage containers and the fuel itself are corroding. The fuels stored in these older tile holes are “legacy fuels” that were generated during the nuclear development programme in Canada in the 1950s to early 1970s. The legacy fuels are primarily uranium metal and uranium dioxide, clad in aluminium.

Further investigations of the state of the early tile holes and their contents found that [14]:

- Water is present inside some of the older tile holes. The water samples taken from some of the holes contain several hundred ppm of chloride and have a conductivity value in the order of mS/cm.
- In some of the older tile holes an elevated and increasing 137Cs activity was found in the tile holes water.
- The surfaces of mild steel components of the tile holes such as liners and fuel containers are corroded in the wet or damp conditions.

The presence of water with elevated levels of impurities is believed to be the main cause of the fuel corrosion in the older tile holes. Most of these fuels have aluminium cladding. Although protected by a natural barrier of oxide film, the aluminium cladding is subject to continued degradation via general corrosion and pitting corrosion in high humidity condition with elevated level of impurities. For example, chloride ions break down the passive film and promote metal dissolution. Optimum corrosion protection of the aluminium cladding in water requires maintaining a chloride content below 1 ppm and a water conductivity of 1–3 μS/cm [15]. Conductivity values near 200 μS/cm are known to be aggressive, causing pitting corrosion of the aluminium clad.

In elevated radiation fields, gamma radiolysis of air produces various gaseous species, including nitrogen oxide gases. When moisture is present, nitric acid can be produced that will accelerate the corrosion.

Once localized penetration of the cladding from corrosion occurs, the fuel matrix immediately beneath the penetration is exposed to the corrosive environment. The uranium metal fuel is particularly active, and hence more vulnerable to corrosion than other fuels. Fission products (e.g. ^{137}Cs) are released from this portion of the fuel matrix as a result of the continuous corrosion process of the fuel.

The potential sources of water/humidity in the older tile holes are considered to be:

- Ingress of water vapour through vent lines, which subsequently condenses on the colder surfaces inside the tile hole.
- Ingress of water through degraded flange gaskets at the top closures.
- Incomplete water removal when the storage containers were originally taken out of the water filled fuel storage bay.

The top closure of the older tile holes consists of a shielding plug with a top flange which is bolted close to grade level to a corresponding flange at the top of the liner, as shown on the left side of Fig. 9. Exposed to the environment of snow, ice, rain and corrosive materials (e.g. road salt during winter), the top closure may no longer be leak-tight. The near grade-level elevation of the top flanges in the older tile holes would make it susceptible to ingress of water in the form of melted snow and ice contaminated with road salt. The design also makes it impossible to detect and remove any accumulated water in the tile hole without opening the closure.

The tile holes of newer design have a raised closure and a separate shielding plug and vented flange. They also have sampling (sucker) lines, running from the base of the liner up through the concrete infill to emerge adjacent to the closure, as shown on the right side of Figure 9. The sampling line allows the detection and removal of any water that might have accumulated without the need to open the top closure.

The newer design of tile holes, which constitute the majority of tile holes in the WMA “B”, have been used to store the more corrosion resistant uranium/aluminium and uranium silicide fuels. These later tile holes are inspected annually for dryness via the sampling lines, and are seen to be resistant to water ingress, representing a significant improvement over the older tile holes. A small number of the newer tile holes have been found to contain small amount of water, especially during the warm and humid summer months. It is suspected that improper installation of the top closure of these tile holes compromised the seal, allowing exchange of the air inside with the ambient air via the leaking top closure and the vent lines with atmospheric pressure changes. The moisture entered would condense on the colder surfaces inside the tile hole and accumulate over time. The water is subsequently removed from these tile holes when detected.

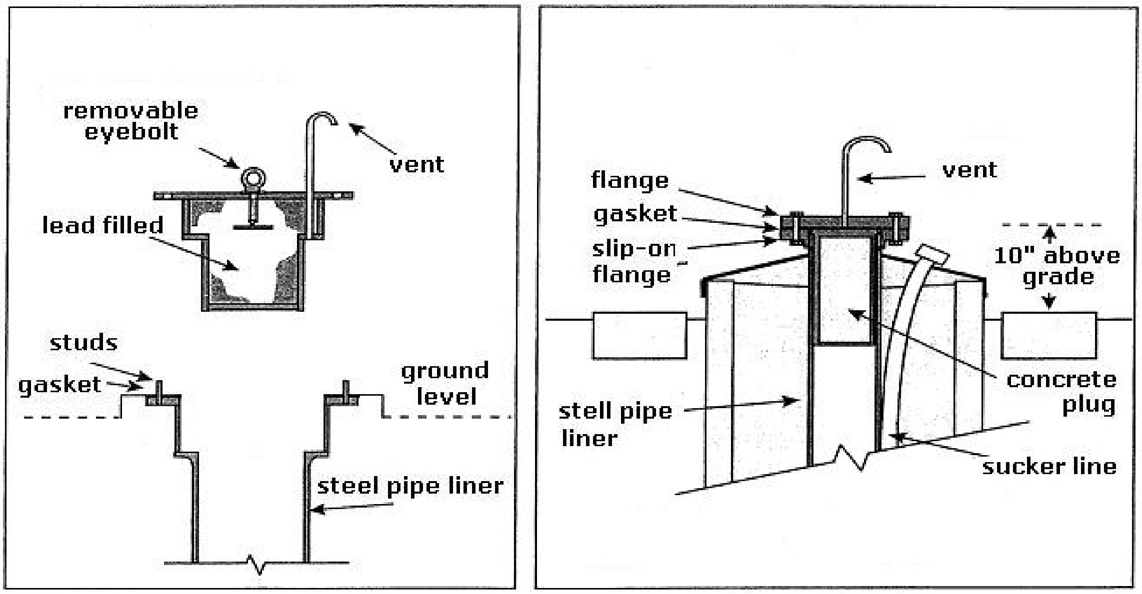

FIG. 9. Closure Designs on Early Tile Hole (Left) and on Newer Tile holes (right).

For the older tile holes, short term remedial actions have been taken to remove water from the holes, and to limit further water ingress. The top closure gaskets have been inspected and replaced, and surface drainage around the concrete array caps has been improved. A removable weather shield to shed precipitation has been used to cover the arrays for the spring and winter seasons. Tests on the historic fuel containers were conducted to confirm their integrity. Those historic fuel containers that are closed cans were punctured following specific procedures to remove pressurization risks and to allow passivation of any uranium hydride formed. The corrosion rates of aluminium and steel were assessed using corrosion coupons placed in the tile hole environment over a period of 13 years.

Recognizing that the older tile holes, which are approaching the end of their design life, are not suitable for longer term storage of the legacy fuels, AECL is currently in the process of implementing the Fuel Packaging & Storage (FPS) project to:

- Retrieve the legacy long-rod fuels from the older tile holes;
- Package and dry these retrieved fuel;
- Store the packaged fuels in a new aboveground dry storage facility.

The main construction will be a Fuel Packaging and Storage building, as illustrated in Figure 10, that will contain two Packaging and Drying Stations and a monolithic concrete Storage Block. The fuel will be retrieved in the existing storage containers (cans), which will be placed in a new stainless steel container with a vented closure, and will then be vacuum dried before emplacement in the monitored storage block.

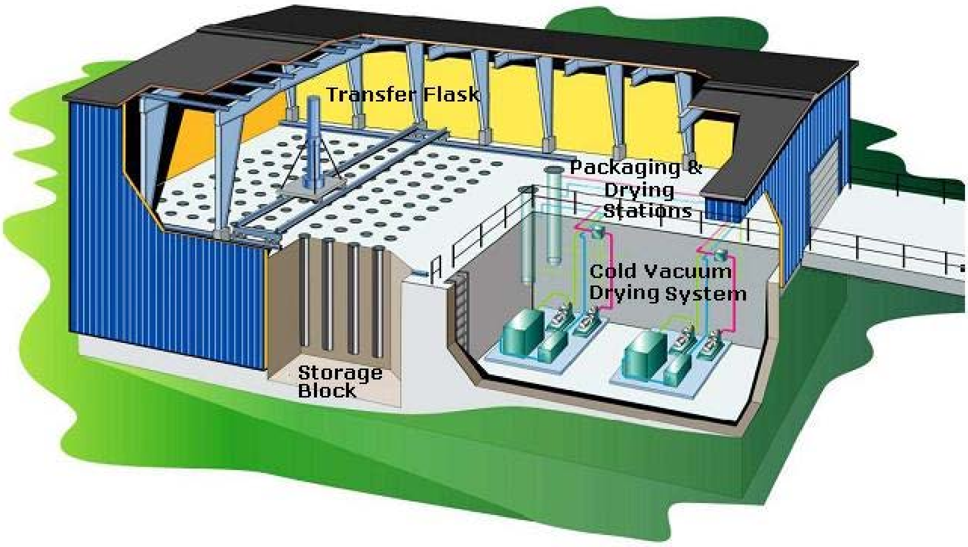

FIG. 10. Conceptual view of the FPS building.

As shown in Figure 11, the top of the storage container is closed with a cap equipped with sintered metal filters. These filters, while confining radioactive particulate, will allow flow of gas/moisture during the vacuum drying process. After vacuum drying in the Drying Station, the container is placed into the storage block, which is subsequently backfilled with an inert cover gas.

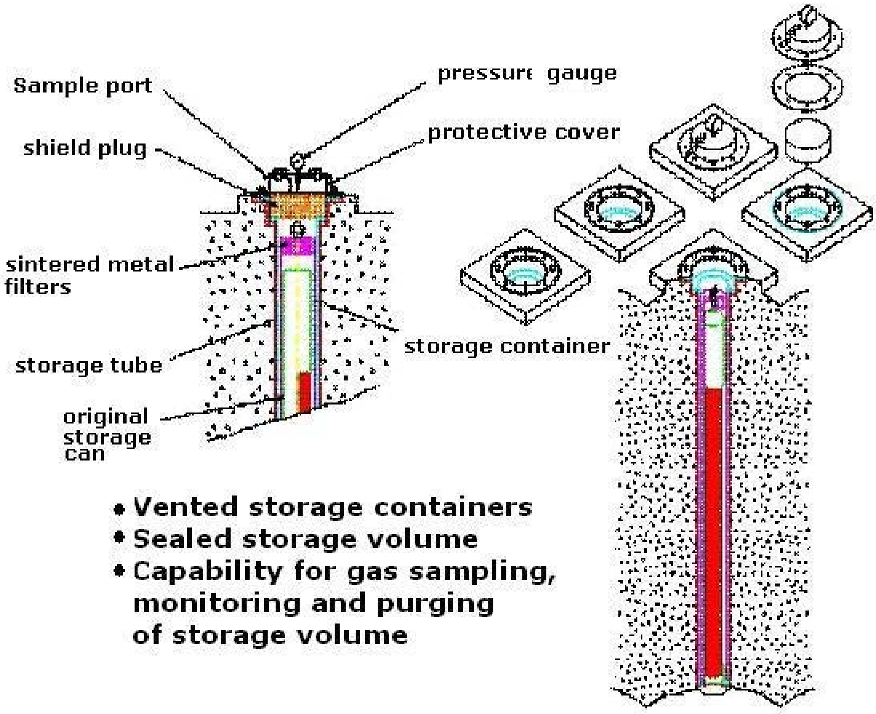

FIG. 11. Fuel Packaging and Storage (FPS) dry storage concept.

The top of the storage block is closed by a shield plug equipped with a pressure gauge and a sample port. An increased reading of the pressure gauge is indicative of the presence of excessive residual water/moisture that continues to corrode the contents within the sealed storage volume. If confirmed by analysis of gas samples taken via the sample port, the storage volume will be purged, re-dried and refilled with inert gas. The FPS storage structure will provide shielding and a secondary confinement boundary for the stored fuel, and facilitate monitoring and inspection. The Storage Block design utilizes many features of AECL's Modular Air Cooled STORage (MACSTOR) system for dry storage of spent CANDU fuel.

4. SUMMARY

Over the past several decades, AECL has been engaged in the technology development and implementation of dry storage facilities for a variety of spent nuclear fuels.

AECL manages a diversity of spent fuels from the research reactors at CRL site. Performance monitoring programmes have identified the need to implement an improved storage system for the less corrosion-resistant spent fuels from research reactors. To accommodate the difficulties in completely removing the residual water from a loaded storage container, a vented dry storage system is selected that facilitates the performance monitoring and remediation during long term dry storage.

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STORAGE OF RESEARCH REACTOR SPENT FUEL IN NORWAY

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Abstract

Since 1951, Norway has experience in management of research reactor spent nuclear fuel (RRSNF). In total, four research reactors were built, all owned and operated by Institute for Energy Technology (Institutt for Energiteknikk, IFE), an independent foundation located at Kjeller, east of Oslo, which is also responsible for the management of the RRSNF. IFEs activities are controlled by Norwegian Radiation Protection Authority, the Norwegian governmental authority on matters concerning radiation protection and nuclear safety and security. IFE keeps most of the spent fuel from the four research reactors in three dry storage facilities, two (JEEP I and Met Lab II) located in Kjeller and one located in Halden. Traditionally the fuel is moved to the dry storage after being cooled for some time in a water pool. The spent fuel stored in the Kjeller site includes 3.1 tonnes of natural metallic U fuel in aluminium (Al) cladding from former JEEP I and NORA reactors, and about 1 ton of 3.5% enriched UO_2 fuel from JEEP II reactor. The Halden site has about seven tonnes of naturally enriched metallic U fuel, from HBWR. At JEEP I storage facility fuel management includes periodic visual inspections of the fuel, to assess any eventual degradation of the fuel cladding. This is important because, considering that most of the spent fuel is in the form of metallic U in aluminium clad, an eventual degradation of the cladding can expose the metallic U to water, allowing a chemical reaction that liberates Hydrogen, which, depending on the concentration, is explosive. In addition to IFEs 60 years of experience on management of metallic U fuel, the paper summarizes also the work which is being done in Norway to find solutions for extending the interim storage of the RRSNF, and to find methods to stabilize the spent metallic U fuel, to render it acceptable for final disposal.

1. INTRODUCTION

Norway is not a nuclear energy producing country. However, since 1951 Norway has some heavy water research reactors which are operated by IFE. Actually only two Norwegian research reactors are in operation, JEEP II, located in Kjeller, near Oslo and the Halden Boiling Heavy Water Reactor (HBWR), located in Halden, 140 km south of Kjeller, near the border to Sweden. JEEP II started operation in 1967, and has a nominal power of 2 MW, and HBWR started operation in 1959, with a licensed nominal power of 25 MW, although its regular operation is at 20 MW. The Department of Nuclear Materials Technologies (NMAT) at Kjeller, supports both reactors with fuel, hot cell, and fuel storage services. The two other research reactors were JEEP I, which operated from 1951 until 1967, and NORA, which operated from 1961 until 1968. JEEP I was decommissioned in 1969, and NORA was decommissioned in 1979 [1].

The research reactor spent nuclear fuel (RRSNF) stored in Norway comes from all 4 research reactors, JEEP I, NORA, JEEP II, and HBWR. In total Norway has some 16 tonnes of RRSNF, whereof six tonnes are stored at Kjeller and ten tonnes in Halden. About ten tonnes is in the form of metallic U, and six tonnes is in the form of oxide fuel. Most of the metallic fuel stored in Halden and Kjeller arose from the early years of operations, and actually, it does not require active cooling. Since 1951 Norway has experience in handling, managing and storing the research reactor nuclear spent fuel, keeping in storage all spent fuel from the four national research reactors, JEEP I, NORA and JEEP II at Kjeller plus HBWR at Halden.

The Norwegian nuclear installations at Kjeller and Halden, namely the research reactors and the storage sites, are owned and operated by Institute for Energy Technology (IFE), Norway [2]. IFE, formerly IFA (Institutt for Atomenergi), is an independent foundation. Its headquarters is located in Kjeller, northeast of Oslo, and according to IFE´s policy, the spent fuel from the reactors is kept in a storage facility built at the reactor site, close to the reactor building. IFE manages three dry storage facilities, two in Kjeller and one in Halden. One storage site contains what is called “historical fuel”, exclusively. This fuel has been stored for 40 to 60 years from the two decommissioned reactors. To date, there has been no reported incident at Norwegian facilities with respect to these activities.

IFEs activities are regulated by Norwegian law. IFEs nuclear activities are controlled by the Norwegian Radiation Protection Authority (NRPA). In Norway, all nuclear activities, including transboundary movements, are regulated by the Norwegian Act on Atomic Energy Activity of 1972 and the Act on Radiation protection and Use of Radiation of 2000. NRPA is the Norwegian governmental authority on matters concerning radiation protection and nuclear safety and security. Norwegian general safety requirements, for the safety of spent fuel management, follows IAEA recommendations, and all facilities owned by IFE are under IAEA safeguards.

2. STORAGE OF SPENT FUEL IN NORWAY

In Norway there are two sites with facilities related to research reactors, Halden and Kjeller. Facilities in Halden include the HBWR bunker, a temporary storage pool, shielded cells and handling facilities, plus a dry storage facility. Facilities in Kjeller include the JEEP II reactor, a fuel production laboratory; a facility for conditioning low and medium active waste; two dry storage sites (JEEP I and Met Lab II), and a hot laboratory facility. The fuel production laboratory (a pilot plant for production of UO_2 fuel, mechanical enrichment, and sintering), is used for the production of driver and experimental fuels (oxide fuel) for use and test in the research reactor. The hot laboratory offers the necessary cells for hot handling, workshop and PIE services, and is connected to one storage site by an internal transport system.

2.1. Kjeller site — JEEP I storage

The dry storage facility JEEP I "stavbrønn" in Kjeller, shown in Fig. 1, has vertical storage pits. The JEEP I fuel, also called historical fuel, is more than 40 years old, and does not require active cooling. The burnup is from 10 – 1000 MWd/t U; mostly in the interval 200 – 400 MWd/t U. This storage site has no activity at present. Most of the fuel stored at the JEEP I dry storage site is metallic fuel. The spent fuel assemblies (SFAs) from JEEP I represent about three tonnes of natural metallic U in Al cladding.

FIG. 1 Kjeller site — JEEP I Spent fuel dry storage facility (JEEP1, NORA).

The SFA are stored in baskets made of stainless steel. Each pit position contains one inner steel basket, only. In the dry pit the stainless steel basket stands in an "outer steel storage basket", a 2800 mm long 3 ¼" steel tube. In some pits two spent fuel assemblies are stored in one inner steel basket. In 1982 all spent nuclear fuels (SNFs) were inspected and repacked into the respective stainless steel baskets. Actions were taken to dry up moisture/condensed water found in the pits, to reorganise the dry pit storage inventory and to re-load the fuel on the respective pits. In 2000 a transfer of fuel elements and slug inventory was performed from the "bunker" to the pits in the JEEP I dry storage facility.

The fuel elements were inspected and repacked/transferred from Al baskets into IFE designed and fabricated stainless steel baskets in the campaigns of 1982 and 2000. The stainless steel basket material is AISI 304. The new baskets are in total 2905 mm long, the free rod length below the lid is

2760 mm, and the diameter of the baskets is 75 mm (di:72 mm). The lid outer diameter is 90 mm. The baskets were leak tested.

The storage site contains in total 96 baskets, whereof 88 baskets contain Al clad metallic U fuel, six baskets contain Al clad LEU (1.7%) oxide fuel, and the last 2 baskets contain radioactive sources. The JEEP I dry storage facility has in total 97 pit positions, one position is empty.

The inventory of the JEEP I dry storage facility consists mainly of spent fuel elements from JEEP I, and NORA. The materials of the spent fuels are natural U metal, low enriched UO_2, ThO_2 & UO_2, and radioactive sources. The fuel cladding in all fuel is Al (Al2S). In low enriched UO_2 rods (JEEP I SEED fuel) end pellets are MgO slugs.

2.2. Kjeller site — Met Lab II

The dry storage facility Met Lab II at Kjeller shown in Fig. 2, is in active use and contains mainly oxide fuels in Al or Zircaloy cladding. The total amount is about 2 tonnes of Al clad standard fuel from JEEP II, more precisely 96 JEEP II Al clad LEU oxide fuel elements. The JEEP II fuel has a minimum pool cooling period of 6 month prior to transport, packing and storage in the dry storage site. The JEEP II fuel has an average BU of 15–16000 MWd/tU after 10 years operation in the research reactor. Additionally, the Met Lab II storage site also contains Zircaloy clad experimental oxide fuels from HBWR, the Halden research reactor. From this reactor, 186 baskets contain fuel with accounted amount (whereof 173 contain a mixture of complete fuel rods with non-intact fuel, and 13 contain only complete fuel rods) and 96 additional baskets have fuel without the accounted amount. Some of the non-intact fuels, from post irradiation examination, is mounted in epoxy and has to be treated prior to transport to the intermediate storage site.

FIG. 2. Kjeller site — "Met. Lab II" Spent fuel dry storage facility.

2.2. Halden site — HBWR bunker

At the HBWR, spent fuel is stored in a bunker, a building outside of the reactor hall. The 48-year old metallic natural U fuel is stored inside of the bunker within a dry storage compartment. The horizontal dry storage site in Halden has no activity at present. It contains 6.9 tonnes of metallic U fuel clad in Al. The fuel has a burnup around 11 MWd/t U and is from the first load of the Halden reactor, removed from the core in 1961. Figure 3 shows the Halden bunker.

FIG. 3. The Halden bunker.

The remaining HBWR spent fuel is oxide fuel, a part of which is stored in a dry storage, but the majority of the spent fuel is stored in a pool underneath the floor.

3. SPENT FUEL IN NORWAY

Some properties of the Norwegian heavy water research reactors and their standard reactor/ driver fuels are listed in Table I.

TABLE I. PROPERTIES OF NORWEGIAN HEAVY WATER RESEARCH REACTORS AND THEIR STANDARD REACTOR FUEL/DRIVER FUELS[a]

	JEEP I (Kjeller)	NORA (Kjeller)	HBWR (Halden)	JEEP II (Kjeller)
Operation	1951–1967	1961–1967	1959–	1967–
Power	Before 1956: 100 kW (t) After 1956: 450 kW (t)	< 0.1 kW (t)	Licensed: 25 MW (t) Regular operation: < 20 MW (t)	2 MW (t)
Fuel	Metallic	Metallic/ Oxide	Before 1967: Mostly metallic Presently : Oxide	Oxide
Cladding	Aluminium	Aluminium and Zircaloy	Zircaloy	Aluminium
Enrichment	Natural	From natural to 3,41%	6% for driver fuel and < 20% for experimental fuel	3,5%

[a] experimental fuel is not included in the list

3.1. HBWR fuel specification

The "first load" of HBWR was fully made of metallic natural U fuel. The total mass is 6.7 tonnes natural U. Each fuel element has one fuel pin (ø 25.4 mm) with an active fuel length of 2375 mm. The reactor operating conditions were 1000 hours at low temperature (appr. 150°C) and low power (max. 6 MW) in the period from June 29, 1959 until 1962. The total generated power in the period was approximately 79 MWd, which represents about 0.27 MWd per element. The average burnup was approximately 12 MWd/tU.

3.2. JEEP I / NORA fuel Specification

The JEEP I and NORA fuel elements contain 2 fuel rods each with an active fuel stack of 1900 mm. Each fuel element had an original U mass of 35.6 kg per rod unit and an Al mass of 8 kg per rod unit. 88 pits of JEEP I storage facility contain one SFA each. The total Al mass of the SFAs is 0.7t, and the total U fuel mass is 3.1 tonnes (mass of the U-fuel slugs). The dimensions of the U-slug are 25.32 mm × 298.2 mm. U-slug weight is 2739 gram. The total weight per rod-unit is 44 kg. Overall specifications of the Norwegian spent research reactor nuclear fuels are given in Table II.

TABLE II. NORWEGIAN RESEARCH REACTOR SNF SPECIFICATION

	HBWR	JEEP I / NORA	JEEP I SEED fuel	JEEP I Void fuel	JEEP II
Fuel	Metallic U	Metallic U	UO_2	Natural U	UO_2
Fuel rod diameter, cm	2.54	2.54	1.27	2.54	1,28
Fuel length, cm	237.5	190 (active fuel stack)	96.5 (active fuel stack)	190	90
Number of fuel rods	308	176	60	7	1050
Initial U enrichment	Natural	Natural	1.7%	Natural	3.5 %
Burnup, MWd/t U	12	1 – 1000 (mostly 200 – 400)	570–1000	unknown	Average 10.000
Total mass U, tonnes	6.7	3.1	Approx 75kg	37 kg	Approx 2.0
Cladding material	Al	Al	Al	Al	Al
Irradiation period	1000 hours (1959–1962)	1951–1967	1951 – 1967	1951–1967	1966 to present
Storage[a]	Horizontal dry storage, at IFE, Halden No.[b]	Vertical dry storage at IFE, Kjeller	Vertical dry storage at IFE, Kjeller	Vertical dry storage at IFE, Kjeller	Vertical dry storage at IFE, Kjeller

[a] inside stainless steel basket; [b] No. Al baskets

4. MANAGEMENT OF RESEARCH REACTOR SFAS IN NORWAY

The SFAs from the early Norwegian reactors were subjected to interim inspections during reactor stops. When interim inspection showed signs of cladding degradation, such as blisters, spallation of the surface oxide or cracks, the fuel was not returned to the reactor core; instead, it was left aside, to be moved to the dry storage. From the time the fuel was removed from the reactor core until it was placed into the dry storage, it was cooled in water pools in the reactor hall. According to IFE practice, the spent nuclear fuel element is first cooled in a water pool, and then it is encapsulated in a waste basket for dry storage. An overview of the spent fuel management in Norway is given in Table III.

4.1. Hazards with Al clad metallic U fuels

It is important to highlight that Norwegian research reactor spent fuel has one additional issue that needs to be considered, the hazards that Al clad metallic U represent, because, as explained before, in Norway there is about 16 tonnes of research reactor spent fuel, whereof about 10 tonnes is Al clad metallic U fuel.

As reported in the literature, the main hazards from corrosion of Al clad metallic U fuel are explosion and fire [3]. Al clad metallic U fuel corrodes in the presence of water, leading to Al cladding degradation with pittings and cracks, and eventually exposing the metallic U to the water, giving rise to another corrosion process. U corrosion by water at temperatures below 100°C leads to U hydriding (UH_3 formation), and finally to Hydrogen formation when the U further reacts with water. This occurs in sections where the metal shows deep localized corrosion. Hydrogen formation was reported when U was in direct contact with another metallic element e.g. Al, suggesting electrolytic factors to be involved [4]. Corrosion of such fuels can lead to severe structural degradation with eventual fragmentation and powderization, as reported from historic Al clad metallic U fuel after long time storage. There are some reported situations when such damages were observed during visual inspection, repacking operations, and during spent fuel processing and preparation for transport, or for reprocessing [5–7].

TABLE III. OVERVIEW OF NORWEGIAN RESEARCH REACTOR SNF MANAGEMENT

Reactor	JEEP I	NORA	HBWR	JEEP II
Irradiation Period	1951–56: 100 kW(t); 1956 - 67: 450 kW(t)	1961–67 <0.1kw(t)	1959–62(1000h): < 6 MW(t) 1962 to present: 20MW(t)	1966 to present 2MW(t)
Wet storage / cooling in water pools[a] — Clean water. Rinsing circuit.	Max.10 years Al-baskets after wet storage + repacking into steel baskets	Al-baskets after wet storage + repacking into steel baskets	Max 3 years Al-baskets after wet storage	0.5–1 years in water with rinsing circuit packing into steel baskets
Visual Inspections / interim inspections[b] on. Documentation[c]	Some elements had defects. Discharged from reactor. In 1982 visual inspection of all elements and repacking.	Some elements showed defects in 1982 visual inspection	All rods were intact prior to packing into baskets	All rods were intact prior to packing into baskets
Long term dry storage location Packages:	JEEP I dry storage 1961–2000	JEEP I dry storage 1962–2000	HBWR bunker since 1962	Met. Lab II dry storage since 1967

[c] From the time when the fuel was removed from the reactor until it was placed into dry storage, the fuel was kept cooled in water pools, in the reactor hall.

[b] Integrity of fuel elements and fuel rods. When visual inspection showed any sign of cladding degradation such as blisters, spallation of the surface oxide or cracks, the fuel did not return to the reactor core.

[c] Documentation of the state of all rods and fuel elements is done on so called “log-cards”.

4.2. Fuel management at JEEP I storage facility

The fuel management at JEEP I storage facility includes safety and security actions. Activities include monitoring and inspections of dry storage, with respect to water ingress; periodic visual inspections of the reactor fuel; fuel log-cards on each spent fuel assemblies, with dates on fuel handling & storage, data on burnup; plus remarks on findings from visual inspections of the state of the Al cladding, e.g. oxide spallation, balloon, pitting, blister and cracks.

The visual inspection data gives an indication on the fuel integrity or fuel degradation. However, the last inspection was performed in 1982. The types of defects found during visual inspection of the spent fuel assembly (SFA) from JEEP I were oxide spallation, balloons, blisters, pittings and cracks in the Al. Most defects were observed in the very earlier SFAs, the ones discharged in 1951. It was understood that the reason was that the Al cladding of these early fuel elements was not well oxidized (anodized) to achieve a protection surface oxide layer as good as the cladding of the later fuel elements. An overview of the defects of the SFAs is given in Table IV.

TABLE IV. DEFECTED SFAS IN JEEP I STORAGE SITE (TYPES, QUANTITIES AND DATES)

Oxide spallation	Balloon	Blisters	Pittings	Cracks
2 SFA	7 SFA	27 SFA	5 SFA	31 SFA
discharge	discharge	discharge	discharge	discharge
1951[a]	1951[a]	1951[a]	1951[a]	1951[a]
(65)[b]		(52, 54, 59, 63, 64)[b]	(52, 54)[b]	(52, 54, 60, 63, 64)[b]

[a] Year in which the majority of the defects were observed.

[b] Years in which additional defects were observed.

4.3. Management of HBWR spent fuel rods from first loading

The HBWR SFAs from the first loading had a wet storage period of less than 3 years. The low burnup fuel, 12 MWd/t U, was loaded into the Al basket before 1962. The rods were dried before loaded into the Al storage basket. By late 1962 all fuel elements had been placed in the dry storage. All elements were intact prior to conditioning for long time dry storage. The state of the fuel rods was recorded / documented on log cards for each rod. The storage facility is air cooled by natural circulation from the outside. Observations showed that the storage positions were kept dry. The storage was sealed (by the IAEA) until 2006, and no further inspections were done before that. After 47 years in dry storage, one rod of Al clad metallic natural U-fuel was selected randomly from the HBWR first loading for visual inspection. The inspection showed that the rod has an intact cladding, however with clear signs of incipient corrosion, as shown in Fig. 4.

5. DISPOSAL OF NORWEGIAN RESEARCH REACTOR SPENT FUEL

The management of spent nuclear fuel in Norway has gone through various phases. The first core loading from the HBWR was dry stored after its discharge in 1961. In that decade reprocessing was an emerging technology, and spent fuel from JEEP I was used as loading material in a pilot reprocessing plant, at the Kjeller site. This plant was in operation from 1961 to 1968, then later decommissioned. The second core loading of HBWR was reprocessed in Belgium in 1969. The U and plutonium gained from the reprocessing was sold for civilian use, and the waste was disposed of in Belgium. When the third core loading was discharged, reprocessing was no longer a politically viable option, and a decision was taken to keep this and later discharged spent fuels from HBWR, stored on site, together with the discharged first core loading. Since the end of the 1990s several strategy working groups were established in Norway, with the purpose to define viable options to deal with the Norwegian research reactor spent fuel, namely the Bergans Committee (2001) [8], the Phase I Committee (2004)

[9], and the Phase II Committee (2010). Since IFE owns several tonnes of metallic U fuel clad in Al and UO_2 fuel clad in Al, and because both metallic U and Al metal react with water, forming unstable reaction products and hydrogen gas, these fuels should be stabilized before they can be stored and placed in final repository. Considering this, Phase I Committee recommended the establishment of an additional Technical Committee, with the specific purpose to investigate the safe storage of metallic fuel and fuel with Al cladding. This Technical Committee, formed with members from IFE, Studsvik and IAEA, was established in January 2009.

FIG. 4. Cladding of a randomly selected rod from the HBWR first loading, after 47 years on dry storage.

6. THE MANDATE OF THE PHASE II COMMITTEE

The mandate of Phase II Committee is to present, by the end of 2010, and based on the findings of the previous Committees, including the Technical Committee, the most suitable technical solution and location for the extended interim storage of Norwegian research reactor spent fuel and long lived waste. It is expected that the Technical Committee delivers, by January 2010, its report to the Phase II Committee, about the recommend treatment options for metallic U fuel and Al-clad fuel, to render them stable for long term storage and disposal. Phase-II-Committee will then recommend the technical option and location of an interim storage facility. Based on safety, security, ethical, technical and economic reasons, it is expected that no further treatment is required when transferring the spent fuel from the interim storage to final disposal. This means in practice that the fuel should be stabilized in such a way that it is acceptable for final disposal.

7. MANAGEMENT OPTIONS FOR INTERIM STORAGE AND FINAL DISPOSAL OF NORWEGIAN RESEARCH REACTOR SPENT FUEL

There are at least five management options for the interim storage of the Norwegian research reactor spent fuel, whereof at least two present methods to treat the unstable metallic fuel, to render it acceptable for interim storage and final disposal.

(1) *Pack in* steel/Cu/concrete without treatment. This option has the disadvantage that there can be a potential contact between metallic fuel and water, and measures have to be taken to avoid Al-concrete interactions, that could lead to the corrosion of clad. The method and its limitation is discussed in details in literature [6].
(2) Reprocessing. Reprocessing plants in France and the Russian Federation offer commercial services. Reprocessing in Norway is not cost effective, but needs to be considered if there is no other cost effective solution.
(3) Oxidise the fuel to UO2 / U3O8. This is a possible option that needs to be considered, using as reference the Korean dry processing of CANDU fuel. A feasibility study is necessary for the Norwegian spent fuel [10].
(4) Return of fuel to the country of origin. Actually there are several programmes that allow the return of the fuel to the country where it was originally enriched/processed, with the possibility to extend the programme to other fuels. If the fuel meets the necessary requirements, this is an option that needs to be considered.
(5) Exchange for other fuel. Look into possible options for exchange of unstable metallic fuel for fresh fuel.

8. CONCLUSION

Since the end of the 1990s strategy work has been undertaken in Norway to decide how to deal with the 16 tonnes of Norwegian research reactor spent nuclear fuel (SNF). The dry stored SNF is from the Norwegian heavy water reactors, namely JEEP I from the periods 1951 to 1967, NORA reactor from 1961 to 1968, the two MW JEEP II from 1966 to presently, and the 25 MW Halden Boiling Heavy Water Reactor in Halden (HBWR), from 1959 to presently. The amount of metallic U fuel from the early years of operations amounts to about 10 tonnes whereof three tonnes are dry stored in Kjeller and seven tonnes in Halden. Metallic U and Al clad fuel is unstable and has to be treated, in order to render into a stable form prior to interim storage and final disposal. Hazards from aqueous corrosion of Al clad and metallic U fuels are explosion and fire. A Technical Committee, established in January 2009, is working on a proposal of methods to treat unstable fuel to render it acceptable for interim storage and final disposal. All technical options remain “open”. The Technical Committee will deliver its report and recommendations to the Phase II Committee and the Ministry in Norway by 15 January 2010, and the Phase II Committee has to deliver a suggestion for a suitable solution and location for an intermediate storage of the Norwegian research reactor spent fuel by the end of 2010.

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GOOD PRACTICE OF INTERIM STORAGE OF RRSNF[1] INSIDE CASTOR® MTR-2 FLASKS IN AHAUS, GERMANY

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Abstract

In Germany the interim storage of RRSNF[46] in away-from-reactor facilities is subject to the Atomic Act, as it is the case for the storage of commercial reactor spent nuclear fuels too. The central storage facilities in Germany, such as Ahaus, are usually licensed in accordance to the atomic law for a period of 40 years, after having undergone a thorough application procedure referring to all applicable nuclear safety aspects and subsidiary regulations. The Ahaus facility was built between 1984 and 1989. It was first licensed in 1987, and in 1997, it successfully underwent a re-licensing procedure, obtaining a valid licence until 2036. It is a dry storage facility for spent nuclear fuel loaded in transport and storage flasks. It has been designed and licensed for a capacity of 420 LWR flasks. In 2005, 18 CASTOR® MTR2 flasks with RRSNF from VKTA Rossendorf were shipped to Ahaus. They were stored amongst 305 CASTOR® THTR/AVR flasks containing spherical fuel elements from the decommissioned Thorium-High-Temperature-Reactor from Hamm-Uentrop. The CASTOR® MTR2 is a smaller version of the dual purpose flask concept suitable for transport and storage, specially designed for research reactor spent fuel elements. It fulfils the long term storage as well as the international transportation safety requirements. Particular attention is paid to quality assurance during flask manufacturing and surveillance also on quality control of the proper technical function during the entire storage phase, and of course on safeguards. Furthermore, an integrated management system, that covers all aspects of quality, safety and the protection of health and environment has been established, and supports the management by the classical structural organization and personal responsibilities. Internal and external audits contribute to a high level management performance. In 2007, the facility took part in a WENRA[47] pilot project on checking conformance with international safety and management standards.

1. BACKGROUND AND LEGAL FRAMEWORK OF THE DRY INTERIM STORAGE CONCEPT

The dry storage concept arose from political discussions back in the seventies, when a reprocessing facility was planned in Germany. Passive safety features and aircraft impact resistance were required for the spent fuel reception area. This led to the idea of storing spent fuel elements inside flasks that were suitable for storage as well as for transportation. Storage of these flasks would take place in simple but solid buildings naturally cooled by a self regulating air flow instead of using a pond with a forced water circulation. As a consequence, two central storage facilities for the collection of spent fuel, prior to reprocessing were constructed, in Gorleben and in Ahaus, both in operation since the beginning of the nineties [1].

The basis for long term storing of spent nuclear fuels and for all other nuclear activities in Germany is the Atomic Act. The fundamentals are laid down in the Atomic Energy Act. Based on the Atomic Energy Act, several subsidiary regulations specify the general rules and set up the requirements for the licensing and operation of an interim storage facility, such as

- Radiation Protection Ordinance;
- Nuclear Licensing Procedure Ordinance;
- Nuclear Safety Officer and Reporting Ordinance;
- Environmental Impact Assessment Act;
- Rules of the Nuclear Standard Safety Commission;
- Etc.

From the set of regulations and technical standards, we derive the following physical protection goals to be fulfilled by the applicant, respectively the operator of a nuclear installation for storing spent nuclear fuels. These are:

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[46] Research Reactor Spent Nuclear Fuel.

[47] Western European Nuclear Regulator´s Association.

- The protection against the ionising radiation from the radioactive sources;
- The safe encapsulation of the radioactivity;
- The safe removal of the heat induced by the radioactive decay;
- The criticality safety;
- The protection against disruptive actions or other interference of third parties.

Under these prerequisites, the Ahaus facility, shown in Fig. 1, was first licensed in 1987. Independently from the nuclear licence, the planning, licensing and construction of the building took place in the eighties. 305 CASTOR® THTR/AVR flasks bearing the spherical fuel elements from the decommissioned Thorium-High-Temperature-Reactor were the first nuclear inventory to be stored in Ahaus, between 1992 and 1995.

In 1995 a new nuclear licence was applied for, taking into account the new developments in flask design which allowed a significant increase in the amount of spent fuel inside of a single flask, i.e. the CASTOR® V-series. This 1997 licence is still valid (expiring 2036) and was extended by the third amendment, allowing the storage of 18 CASTOR® MTR2-flasks with spent nuclear fuel from the decommissioned research reactor of the VKTA Rossendorf in Saxonia. Thus, the RRSNF are treated equally to the fuels from commercial reactors during interim storage.

FIG. 1. The Ahaus facility.

The good practice of interim storage of spent fuels is based on four main components:

- A transport and storage flask highly tight and long term suitable;
- A high profile quality management accompanying the flask development from the design to the manufacturing of each single flask;
- A simply designed and solidly constructed storage facility equipped with reliable safety systems, auxiliary infrastructure and a sophisticated security system;
- A high reliability organization of well trained staff and approved operation procedures for all aspects of the safe handling and storage of the inventory and protection of staff, public and environment.

2. THE CASTOR® MTR2 FLASK

The CASTOR® MTR2 flask [2] is a smaller version of the dual purpose flask concept suitable for transport and storage, specially designed for RR spent fuel elements such as the Russian design fuel type elements licensed for Ahaus:

- WWR-M (single);
- WWR-M (triple);
- WWR-M 2 (single);
- WWR-M 2 (triple);
- EK-10.

The CASTOR® MTR2 flask is shown in Fig. 2. Its main dimensions (without shock absorbers and protection plate) are as follows:

- Overall height: 1631 mm;
- Outer diameter: 1430 mm;
- Cavity height: 920 mm;
- Cavity diameter: 721 mm.

The flask consists of a monolithic body made of ductile cast iron, casted in one piece and machined afterwards. The radiation shielding is provided by a wall thickness of about 35 cm. Due to the low heat load of the RRSNF there are no cooling fins machined into the surface of the MTR2 flask as are in the outer surface of the bigger flask types such as the CASTOR® V. The fuel elements are positioned in a basket made of a massive aluminium cast body with seven shafts for two types of loading units. The type A loading unit consists of a cylindrical aluminium cast body capable of bearing fuel elements in seven positions and equipped with a borated absorber rod in its centre to ensure the subcriticality condition. The type B loading unit is capable of bearing four fuel elements, type EK-10, or three of these elements combined with one fuel rod container bearing EK-10 fuel rods. The rectangular positions are also manufactured within a cylindrical aluminium cast body, equipped additionally with a borated absorber rod in its centre.

The flask cavity is closed with double lids, and the space between the lids is filled with helium at six bar. This pressure is permanently monitored during storage. The lids are made of stainless steel, bolted and tightened by means of metallic seals. Figure 3 shows the closure system of the CASTOR® flask.

FIG. 2. CASTOR® MTR2.

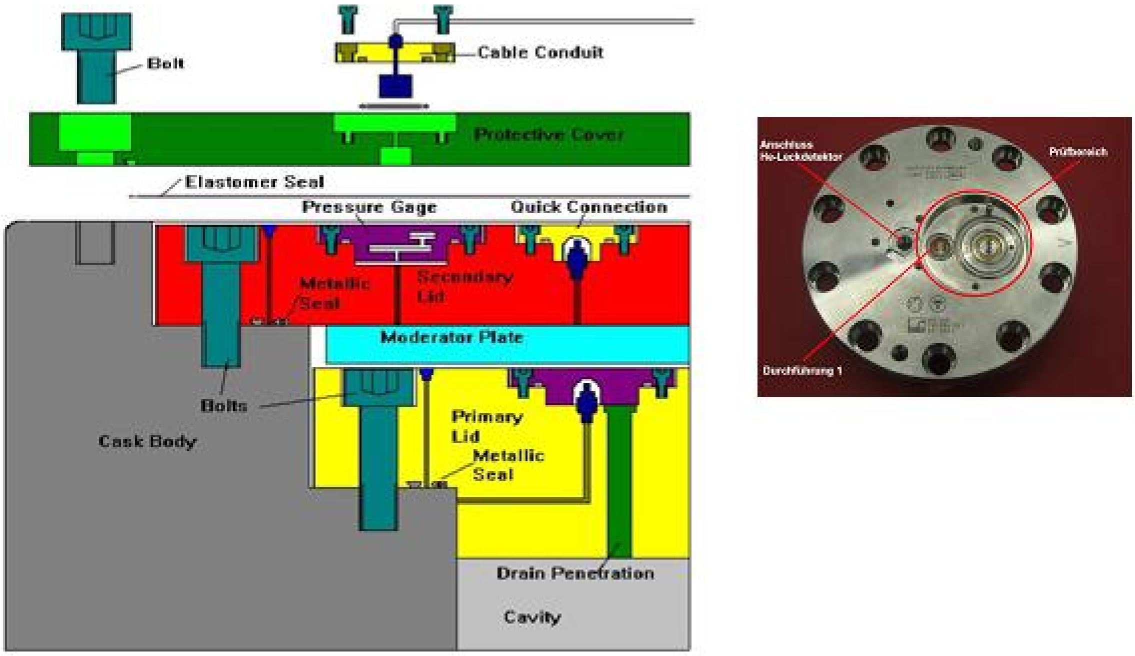

FIG. 3. Monitored Double Lid System.

3. QUALITY MANAGEMENT IN FLASK MANUFACTURING

The double lid sealed flask of all types has to fulfil the long term storage as well as the international transportation safety requirements.

GNS holds a certificate of qualification for the manufacturing of containers for transportation, interim and final storage in compliance with the national regulation TRV 006, issued by BAM, the Federal Institute for Materials Research and Testing, Germany's top competent authority for material research and approval. The GNS QM system is furthermore in compliance with the international standard ISO 9001:2008 and the German nuclear power plant specific standard KTA 1401 [3].

In practice, the procedure is as follows:

Application for flask approval by BAM:

- Safety Analysis Report;
- Drawings, parts lists;
- Material specifications;
- Test procedures (e.g. ultrasonic tests, drop tests).

Approval by BAM with additional requirements for manufacturing and use of the flasks:

- Fabrication and Test Plans, pre-examined by GNS Quality Control and BAM
- Definition of witness and "hold points" for tests and examinations by supplier, GNS QC and BAM.

Flask fabrication and assembly:

- Performance of tests by the suppliers of the components, GNS QC and BAM;
- Performance of witnessing and "hold points" during fabrication and assembly;
- Final test of the flask, i.e. leak tightness test.

For each successful assembly of a flask BAM issues a Certificate of Conformance with the Flask Type Approval for Transportation on Public Routes.

This is a prerequisite for the flask being loaded with spent fuel, provided of course, there is a valid storage licence for the facility of destination.

4. THE AHAUS INTERIM STORAGE FACILITY

The Ahaus interim storage facility TBL-A comprises a storage hall, auxiliary buildings and infrastructure. The storage hall is divided into two storage areas by a central reception area, the latter containing a maintenance compartment and a two-storey social section. The reception area is accessible for transports from the north and from the south side, either by railway or by road. A 140-tonne bridge-type crane handles loading and unloading operations, and carries the flasks to any point of the storage hall [4].

The overall dimension of the hall is 196 m (length), 38 m (width) and 19 m (height). The storage area measures 2.800 m² each wing, equivalent to 420 storage positions for big flasks of the CASTOR® V-type. Concrete wall and ground plate dimensions vary from 25 to 55 cm, thus further reducing radiation from the flasks. The building is earthquake-proof according to German nuclear standard KTA 2201.1.

Every flask is connected to a flask tightness surveillance system. At its heart is the pressure switch monitoring the barrier space between the primary and the secondary lids of the double lid system. The barrier space is charged with helium under a pressure of 6 bars. A loss of barrier space gas would lead to a loss of pressure, detected by the pressure switch at the pre-defined and pre-adjusted pressure level of three bars. The resulting electrical signal would be displayed by the surveillance system at the observation points. No such incident, caused by leakage, has ever been recorded so far. In case of failure of the pressure switch itself, detection is ensured by means of a referential pressure chamber within the switch.

The facility is generously equipped with auxiliary systems such as independent electric power supply and battery power supply for safety and security relevant systems, to ensure non-interruption of availability. The interim storage is laid out for certain defined internal and external impacts and even resistant to a certain extent of residual risks, such as an impact of a fast military jet.

Various kinds of radiation protection systems and devices monitor the radiological safe situation inside of the storage hall, and in the near environment. In the frame of an environmental monitoring system, trained personnel routinely and systematically controls ground, plants, water and air at fixed points in 24 sectors up to a radius of 5 km. At 12 points permanent dose surveillance is guaranteed by thermoluminescent dosimeter components. A weather station registers and provides weather and climatic parameters at the site.

5. INTEGRATED MANAGEMENT SYSTEM

In 2009 an integrated management system (IMS) has been introduced company-wide. It unites the formerly independent systems for quality management, environmental care and health and safety oriented procedures with the classical structural organization and personal responsibilities. The IMS covers all specifications, regulations and organizational aids that are provided within the company to manage the relevant tasks for the success of the company under controlled conditions, and to control and improve the achievement of objectives [5].

To support the persons responsible for the operation of all sections of the company, various specialists have been authorised for special care on

- Quality management;
- Environmental management;
- Radiation protection;
- Dangerous goods;
- Nuclear transports;
- Fire protection;
- Occupational health and safety;
- IT security;
- Test devices;
- Etc.

They are coordinated by the manager of the central auxiliary department, reporting directly to the chairman.

The structure of the various regulatory systems is defined and described in the IMS manual as schematically shown in Fig. 4.

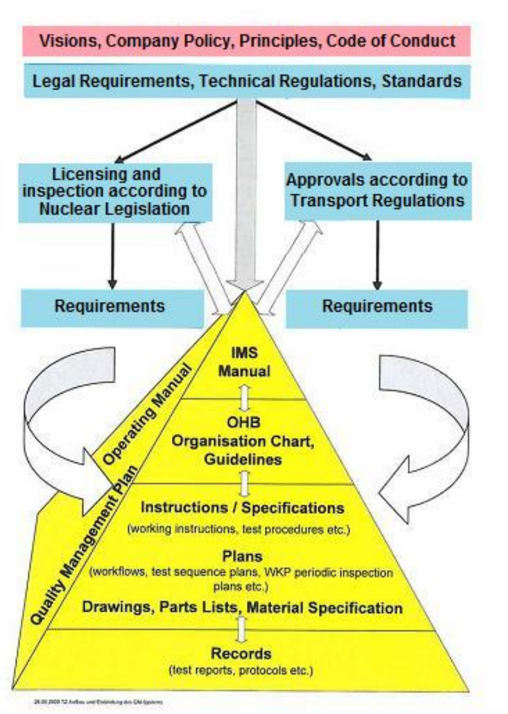

FIG. 4. Regulation pyramid.

The main part of the IMS manual is dedicated to the description of processes, which are distinguish between lead processes, main processes, auxiliary processes and other procedures, as shown in Fig. 5.

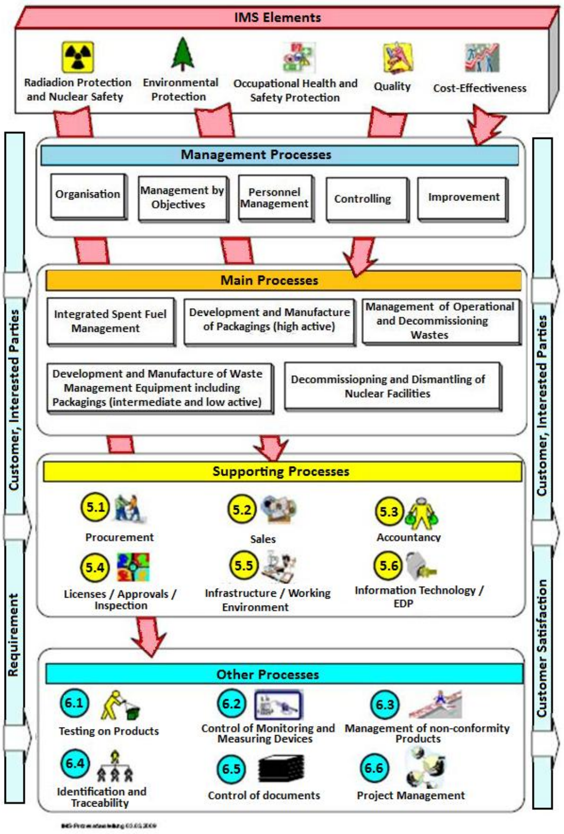

FIG. 5. IMS processes.

Inside this system, the Ahaus interim storage is embedded in the main process "Integrated Fuel Element Disposal". The main process "Storage Operation" is shown in Fig. 6.

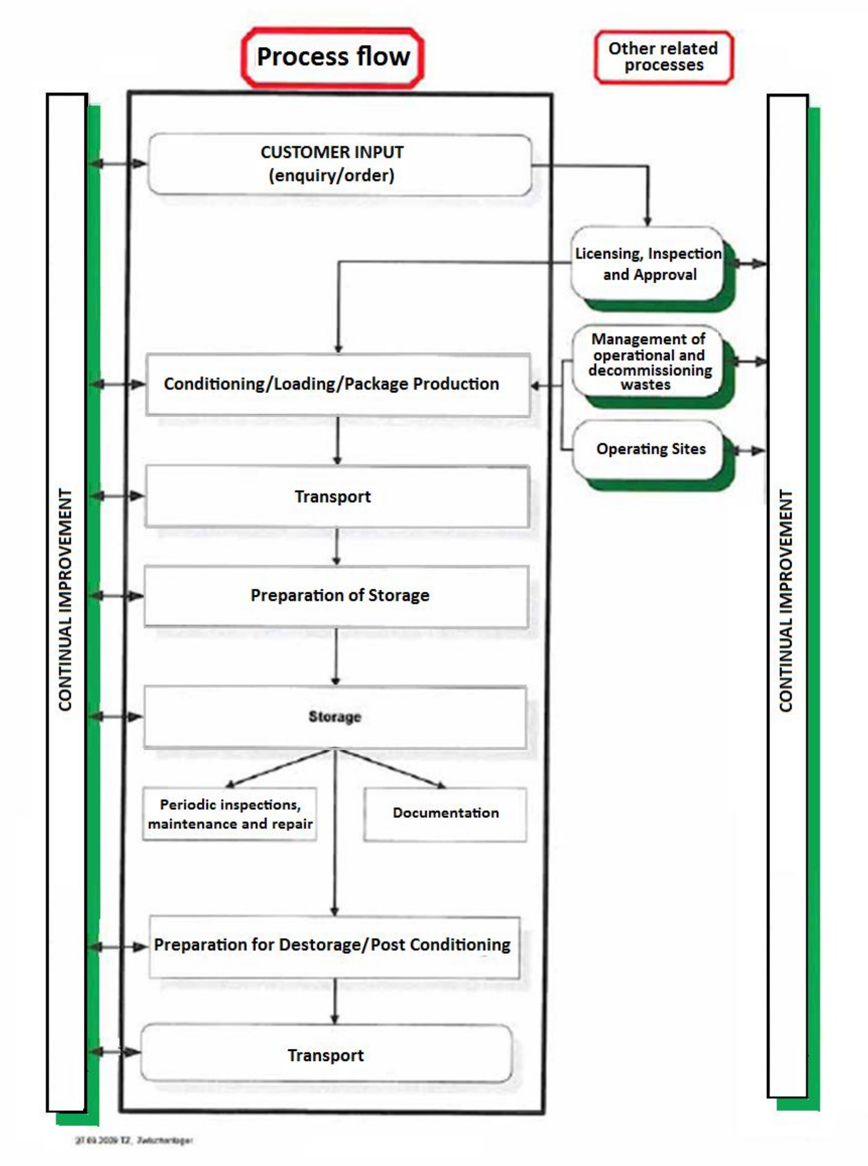

FIG. 6. Main process "Storage Operation".

The storage operations in Ahaus are regulated in detail by the "Operation Manual", licensed and supervised by the competent authorities. Minor amendments of the processes and consequently the manual are permanently accompanied by the supervisory authority and its expert institution. Fundamental amendments lead to an amendment of the licence by the licensing authority.

Such an amendment had to be obtained previous to the shipping of the RRSNF elements from Rossendorf to Ahaus in 2005.

Before storing a flask in the licensed position, radiological measurements are performed to ensure the conditions of the licence and that dose rate and contamination value limits are complied with. After that, the flask undergoes certain preparations inside the maintenance compartment such as mounting a protective cover plate and the cables for the flask tightness surveillance system, checking the conservation condition, temperature and radiation parameters. In the case of the MTR2, as a licence condition, one of each convoy (6 flasks per convoy) had to be selected for a tightness measurement of the secondary lid system. Finally the flasks were stored amongst the 305 CASTOR® THTR/AVR flasks containing spherical fuel elements from the decommissioned Thorium High-Temperature

Reactor from Hamm-Uentrop. The last steps are to connect the flasks to the tightness monitoring system and apply the safeguards seal. From that point on, no regular handling is necessary during the long term storage of the flasks, apart from regular EURATOM/IAEA inspections. Only after ten years of storage, a representative percentage of individual flasks has to be picked up for a check-up of its general condition.

Technical measures are available in the unlikely case of a failure of the sealing of one of the two lids of the flask. Should the sealing of a secondary lid be affected, the lid would be removed in the maintenance compartment and the sealing replaced and once again closed tightly. Should the primary lid barrier be affected by a sealing failure, a third lid could be welded over the flask top, overtaking the function of the second tight barrier. Alternatively, the flask could be prepared for shipping to another nuclear facility equipped with a “hot cell” where the primary lid could be removed for exchanging its defective sealing. Once again, licensed procedures for the pre-shipment preparations could be applied. Analogously these procedures have to be applied whenever the flasks are to be shipped to a final repository, which is, to date, not yet in operation.

To ensure the permanent readiness and proper condition of the technical equipment and procedures, skilled and trained personnel has to be provided and organised. The actual organization chart of Ahaus interim storage is shown in Fig. 7.

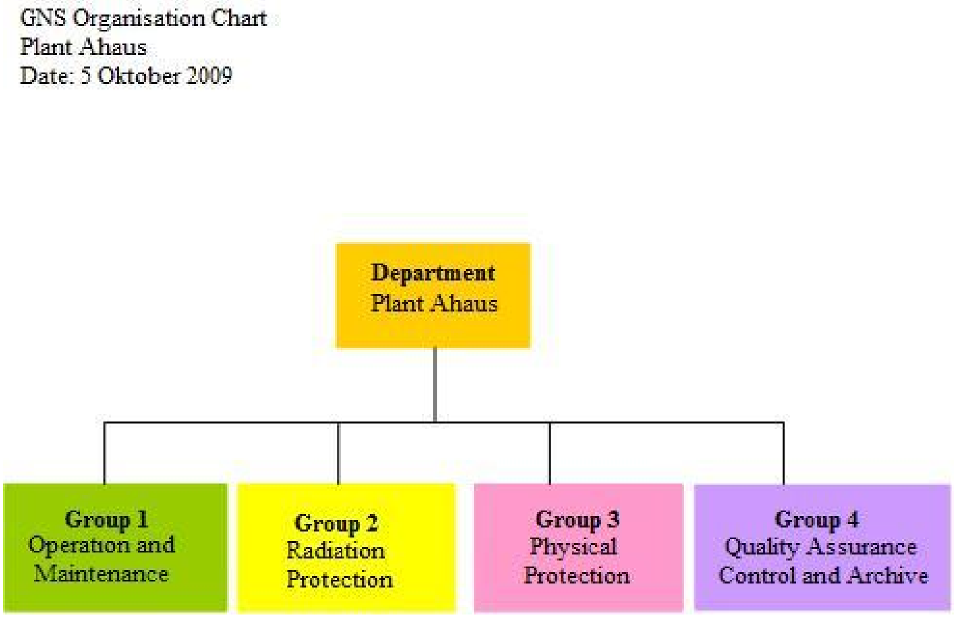

FIG. 7. Organization chart of Ahaus.

Special skills for handling of flasks and welding of a tertiary lid in the above mentioned case are provided by the group of operations. The radiation protection staff operates independently from the technical operations staff and is responsible for minimizing dose uptake by the personnel and supporting the operation group in optimizing the handling procedures to fulfil the radiological goals. The security staff of the third group leads and organises the guard service all around the clock throughout the 365 days of the year. Finally, the fourth group has been installed in 2008 to supervise the application of the qualified procedures and help and give advice to staff and their managers to correctly implement the ideas and principles of IMS and comply with all derived regulations. The systematic return of experience from the groups as well as from outer sources is one of the tasks of the plant manager himself. Regular as well as situational meetings serve, apart from other functions, the continuous optimisation process as one of the central concepts of the IMS. Another element of enhancing a good management and operation performance by learning from each other is the benchmarking with other operators, as it was performed in 2008 amongst the sister installations Gorleben and Ahaus.

Also contributing to a high level management performance are internal and external audits. At least one general internal audit is performed in the facility every year. Its goal is to evaluate the efficiency and adequateness of the practised management of the plant, its personnel and other resources. External audits serve the permanent compliance with the preconditions of the certification or the requirements of the customers. An extraordinary project took place in 2007, when the facility took part in a WENRA pilot project on checking conformance with international safety and management standards in development for waste and spent fuel storage safety [6]. Involved in the German self-assessment were the federal ministry for environment and reactor safety and the competent supervisory authority of North Rhine Westphalia, as well as the operator of Ahaus in cooperation with other storage operators. It could be demonstrated that the degree of compliance with the safety standards is high. Although the facility has been in operation for approximately 20 years and was firstly licensed already in 1987 the compliance with nearly all Reference Levels from 2007 could be proved.

Nevertheless, no lessening of care and good practice may take place at any time. Then, the task of a long term storage of spent fuel elements, amongst them of course the RRSNF type of elements, can be fulfilled to the customers' and the competent authority's satisfaction without doubt.

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