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* 19 June Session

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IAEA Nuclear Fuel Cycle Databases: Relevance to Spent Nuclear Fuel Management

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Reliable statistical data on spent fuel management would be essential for the global nuclear community, e.g. for approaches related to international cooperation, as well as for the needs of individual countries. Compilation of data on large amounts of spent fuel located at various nuclear facilities around the world is a challenge. It is not a trivial exercise to collect and compile spent fuel inventory data as they are subject to dynamic change.

Spent fuel inventory data are important to various national and international spent fuel management activities, especially for planning and regulatory activities. Recently, security issues became an additional factor to be considered in the information management associated with spent fuel or radioactive waste. The specific need for spent fuel inventory data varies depending on the ultimate purpose:

- International Level – compilation on a gross tonnage (in heavy metal basis) mainly for statistical purposes and global trend analysis both for use by IAEA and at the request of Member States;
- National Level – compilation for industry and regulatory purposes on either a gross tonnage or individual assembly basis to assist in planning and public awareness; and
- Operator Level – the origination and maintenance of detailed data on individual assemblies by the utility for operational needs or to meet regulatory requirements.

There is, in general, a global trend towards greater transparency of information with the general public which may require more information to be made public on spent fuel management, including data on inventories or transportation. With the increase in the commercialisation of the nuclear industry, the trend is away from national governments operating nuclear facilities, including spent fuel management. This results in the spread of information on spent fuel as it is not concentrated at government level, but is instead held by various organizations .

Spent fuel information may also have to be considered in the context of nuclear knowledge preservation. In addition to current inventories, historical and projected data are important for various purposes such as consistency analysis.

The IAEA in its role as a focal centre of global cooperation on nuclear activities, established a variety of databases, including the Nuclear Fuel Cycle Information System (NFCIS) which encompasses a global list of nuclear fuel cycle facilities. The NFCIS database was published in hardcopy two times in the past [1, 2]. The database includes a long list of spent fuel storage facilities because of the large number of facilities with spent fuel inventories. It is a challenge to maintain a reliable updating of spent fuel invenory data in those storage facilities, requring a good mechanism in place for the collection and compilation of data to be obtained from the operators or national authorities.

The NFCIS is supported by a simulation software tool named Nuclear Fuel Cycle Simulation System (VISTA) which provides a method for versatile analyses of nuclear fuel cycle systems for all types of commercial nuclear power systems [3]. It provides, among others, for the calculation of varous quantities involved in the operation of nuclear fuel cycle facilities resulting from a given set of assumptions applicable to the various fuel cycles, including the statistics on the generation and management of spent nuclear fuel.

The Nuclear Fuel Cycle Simulation System (VISTA) was developed to calculate fuel cycle material and service requirements. This subject was one of the key issue papers at an International Symposium [4]. The purpose was to develop a simple and fast calculation tool in order to compare different fuel cycle options. The need for a small number of input parameters was a very critical part of the development. Later the model was expanded to enable actinide tracking. A simplified isotopic calculation program (CAIN) was added to the system. By adding CAIN, VISTA became capable of calculating the isotopic composition of spent fuel for any existing reactor type, for a given fresh fuel composition and a given discharge burnup.

A possible use of VISTA might be: i) to estimate actinide accumulations in spent nuclear fuel; ii) to calculate nuclear fuel cycle material and service requirements for selected scenarios; and iii) to compare the different options for future nuclear fuel cycle developments.

NFCIS and VISTA have recently been upgraded to a more user-friendly service which is immediately accesible on the web [5].

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Conceptual Design of a Modular Facility for the Long-Term Dry Storage of PHWR Atucha Spent Fuels

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The reduced storage capacity available in the two spent fuel pools of the Argentine PHWR Atucha-I power plant, along with the desire of extending the reactor operation beyond its present lifetime, i.e. 2014, have motivated the evaluation of an out-of-pool dry storage alternative for the long-term interim management of spent fuel assemblies. Based on different design requirements established by an expert working group [1], a modular facility is being designed in agreement with the Argentine National Regulatory Authority Legislation, as well as the IAEA and the Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials (ABACC) Safeguard Regulations.

The facility is composed of an arrangement of reinforced concrete structures into which metallic welded canisters containing 37 spent fuel assemblies each are stored horizontally. Every reinforced concrete module is passively cooled by air circulating in natural convection to satisfy the following two temperature limits: 200 ºC for the fuel rod cladding surfaces, and 100 ºC for the concrete inner surfaces.

The metallic canister is designed to provide primary containment for 37 spent fuel assemblies located in a triangular array, as shown in Figure 1. The canister consists of a stainless steel cylindrical shell and two welded end plug assemblies that provide the necessary axial shielding at the top and bottom of canister, in order to reduce the occupational doses during handling operations. The canister also contains an internal basket assembly formed by 37 guide tubes that are held in position by eight axially distributed spacer disks welded to ten support rods. The basket geometry provides criticality control and structural support of spent fuel assemblies, both in axial and lateral directions. After sealing, draining and drying operations in the decontamination area, the cavity of the canister is backfilled with helium to improve the heat transfer and prevent corrosion during storage. In Figure 2, a general view the upper part of the metallic canister is presented.

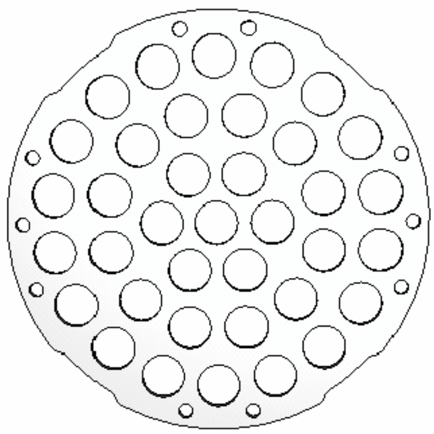

FIGURE 1: Triangular array of 37 spent fuel assemblies per canister and 10 support rods.

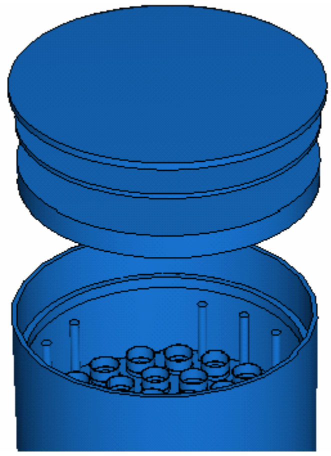

FIGURE 2: General view of the upper part of canister design.

The reinforced concrete module is a low-profile, impact-resistant structure designed to withstand all normal condition loads as well as abnormal condition loadings postulated to occur during design basis accidents. It also provides the necessary biological shielding and means to remove the spent fuel decay heat by a combination of radiation, conduction and natural convection. Ambient air enters the module through ventilation inlet openings located in the lower side walls, circulates around the canister and exits the module through outlet openings in the upper side walls. Adjacent modules are spaced to provide a ventilation flow path between modules. The canister is located horizontally inside the module resting on a railed structure manufactured from structural steel.

In addition to these two components, the modular system also comprises transfer equipment to move the metallic canisters from the plant's fuel/reactor building, where they are loaded with spent fuel assemblies, to the storage facility site where they are stored in the concrete modules. The transfer system consists of a transfer cask, a lifting yoke, a hydraulic ram system, and a transport trailer, and it is designed to maximize the use of existing plant features and equipment.

The on-site transfer cask is a metallic containment where the canister is stored before loading the spent fuel assemblies, and it is designed to provide radiation shielding and structural protection from potential hazards during the canister closure operations and transfer to the concrete module. Four support trunnions are provided on the cask cylindrical surface for pivoting the transfer cask from/to vertical and horizontal positions on the transport trailer.

The transport trailer consists of a heavy industrial trailer dedicated to the transport of loaded transfer casks between the plant's fuel/reactor building and the storage facility. The trailer is designed to ride as low to the ground as possible to minimize the transfer cask height during transport and transfer operations, and it is equipped with a hydraulic ram for pushing the loaded canister into the concrete module, as well as for the retrieval of the canister, if necessary. A specially designed grapple device located on the bottom of canister is used for the coupling with the hydraulic ram.

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Life Management of Spent Fuel Storage

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Argentina has two nuclear Power Plants in operation, Atucha I (CNA I) and Embalse (CNE), and one under construction, Atucha II. All of them are based on natural uranium as fuel with heavy water moderator and coolant. Atucha I (370 Mwe) and Atucha II (750 Mwe) are almost unique in their type, both are Pressure Vessel type. The Embalse Nuclear Power Plant (600 Mwe) is a typical CANDU plant.

Since September 1994 the organisation of the Argentine nuclear activities, confirmed in April 1997, is as follows:

- "Nucleoeléctrica Argentina Sociedad Anónima" (NASA), which operates the two NPPs (Atucha 1 and Embalse).
- "Autoridad Regulatoria Nuclear" (ARN)- National Regulatory Board of Nuclear Activities.
- "Comisión Nacional de Energía Atómica". (CNEA) - National Atomic Energy Commission.

Within this scheme, one of the main activities to be undertaken by CNEA is to provide technological assistance to NASA in problems concerning NPP operation and on aging management of the NPPs.

The Atucha fuel elements are 6 meters long and have 36 rods with zircaloy-4 cladding. On the other hand, the fuel of the CANDU reactor has 36 rods which are 0.50 meter long. Storage facilities for these fuels are located in the places where the reactors are situated. Atucha I has two pool houses where 9200 spent fuel elements are accumulated so far. (11426 Mg U). From the middle of the year 2000 the whole reactor core is made up of fuel with 0,85% enrichment and therefore, the consumption of fuel will be of 210 fuel elements per year. In the wet storage facility of Embalse there are 37900 fuel elements and in the dry storage 35020 fuel elements. For dry storage there are 80 silos with 9 canisters with 60 fuel elements each. (1447 Mg U).

Upon closure of the Atucha Nuclear Power Plant one option is to transfer the spent fuel from wet to dry storage and another option is to continue with the operation of the pools and transfer only a number of selected spent fuel elements to dry storage. In the case that all of the spent fuel will be transferred to dry storage the schedule is the following:

Considering that the nuclear power station Atucha I will be closing in 2015, we apply the following rules:

- To continue with the wet storage of the spent fuel in the existent pools, until the year 2030.

- To begin the construction of a dry storage facility in the year 2011 and have it in operation in the year 2015 when the country will start transferring gradually all the spent fuel elements from the pools.

Therefore, it is extremely important to work on a Life Management Programme for Long Term Operation (LTO) of the Spent Fuel Storage and on the determination of the optimum technology for dry storage of Atucha I spent fuel elements.

In this paper we will describe the application of an Aging Management Methodology to Spent Fuel Storage. At first we will make a screening of the installations and determine the critical Systems, Structures an Components (SSCs) and then we will discuss the principal aging mechanisms and the stressors. Furthermore we will analyse the different technologies, considering the characteristics of the Atucha fuel elements and the possibility to transfer them from the pools to the dry storage facility. For the selection of the technology we kept in mind, the characteristic of the Atucha fuel elements, i.e. their length and their burnuo that will be different according to the time that is considered. The construction of the spent fuel storage facility should also be carried out in modules, allowing to add sectors according to the necessities. The conditions for the transfer from the pools to the dry. Storage facility should also be considered.

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Management of Spent Fuel from Reactor of Bangladesh

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Bangladesh has been operating a 3 MW TRIGA MARK II research reactor since 1986. The reactor was installed in the campus of the Atomic Energy Research Establishment (AERE) at Savar, which is located at about 40 km northwest of Dhaka. It is one of the main nuclear research facilities in the country. The reactor uses TRIGA LEU fuel with uranium content of 20% by weight. The enrichment level of the fuel is 19.7%. The reactor has so far been operated for 4731 hours with a total cumulative burnup (BU) of 8363 MWh (348 MWd). The main areas of use are: training of man-power for nuclear power plant applications, radioisotope (RI) production, neutron activation analysis, neutron radiography and neutron scattering. Radioisotopes produced to date are: I-131, Sc-46 and Tc-99m.

The Bangladesh Atomic Energy Commission (BAEC) has been pursuing a project to establish a nuclear power plant (NPP) of 600 MW PWR type in the western zone of the country since the mid 60's. Now, with regard to the safe management, storage and disposal of radioactive waste arising from operation of the research reactor (RR) and also from the proposed NPP expected to be constructed in the future, BAEC has drawn-up short and long-term plans and programs.

For addressing safety of radiation sources, protection of man & the environment and in compliance with the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management [1], the national NSRC Act (1993) and the NSRC Regulation (1997) [2] are in operation; while the Radioactive Waste and Spent Fuel management Rule-2005 [3] is awaiting approval of the Ministry.

It is observed that the current spent fuel management policy in different countries can be divided into three broad groups:

Group-A: Countries that have been following the once-through fuel cycle and are focusing their attention on storage followed by disposal of the spent fuel (e.g. Canada, Sweden and the USA);

Group-B: Countries that have selected the reprocessing option and are actively operating or constructing reprocessing plants or have contracts for reprocessing abroad and/or are returning some or all of their fuel to the country of origin (e.g. France, Japan, the Russian Federation and the UK); and

Group-C: Countries that are still evaluating their spent fuel management programs (e.g. the Republic of Korea, Lithuania, Mexico).

Bangladesh is a peace-loving country with a strong commitment towards nuclear nonproliferation. Accordingly, it has signed several multilateral and bilateral agreements, protocols, treaties, etc. prevailing in the International Nuclear Non-proliferation regime. Bangladesh has also signed a Nuclear Cooperation Agreement with the USA on 17 September 1981, which facilitated export of nuclear technology from the USA to Bangladesh. The research reactor was procured under the provisions of this agreement. The tenure of the Agreement has recently been extended up to 2012. Taking all these into account, it is thought that the policy of Group-A together with a part of Group-B involving the returning of some or all of the spent fuels to the country of origin, would be the suitable option for Bangladesh in terms of the management of its spent nuclear fuel from the RR & future NPP [4].

Presently, the TRIGA facility has three Fuel Storage Pits located in the floor of the reactor hall. The pits are made of stainless steel pipes of a diameter 25.4 cm (10 inches) and of depth 457.2 cm (15 feet). Each of the pits is provided with a lock on its stainless steel cover plate to limit access to the pit and also an mild steel cover plate (with lifting hook) that fits flush with the floor. The storage pits were designed to hold 19 TRIGA fuel elements in each pit. For storing the fuel elements into these pits, suitable storage racks would be needed. At present the BAEC reactor facility does not have any rack of such kind. However, efforts have been undertaken to design and develop the storage racks compatible with the storage pits mentioned above and also with the handling and lifting facilities available in the reactor hall. Besides these pits, there are 3 submerged fuel storage racks located along the inner wall of the reactor tank at a depth of about 610 cm (20 feet). Each of the racks is capable of holding 10 fuel elements (total capacity $3 \times 10 = 30$). The purpose of these racks is to provide temporary storage for the fuel-moderator or the graphite dummy elements before their transfer to the fuel storage pits mentioned above. The BAEC reactor facility also needs to have a spent fuel transfer cask for transferring irradiated fuel from the reactor pool to the spent fuel storage pits. BAEC has taken measures to design and develop such cask with the assistance of the IAEA under a TC project being implemented now (2005-2006 cycle) in the reactor facility. Presently, there does not exist any spent fuel element in the reactor facility. However, with the recently undertaken RI production enhancement program, it is expected that the reactor will start generate spent fuels from the year 2012. It is worthwhile to mention that a Central Radioactive Waste Processing and Storage Facility (CWPSE) has been constructed near the research reactor facility in AERE. The activities of this facility include: collection, handling, segregation, characterization, classification, treatment, conditioning, storage and disposal of all kinds of radioactive wastes generated from nuclear installations, and from application of radioactive materials in medicine, industry, research, agriculture, education, etc.

BAEC has been working as the competent authority for the Nuclear Safety and Radiation Control in Bangladesh. The legal basis is the ‘Nuclear Safety and Radiation Control (NSRC) Act, 1993 and the Nuclear Safety and Radiation Control Rules, 1997’ which incorporate the requirements of the International Basic Safety Standards. The Act-1993, Rules-1997, Statutory Regulatory Orders of 1996, 2000 and the Rules-2005 cover all aspects of radiation, waste and transport safety. BAEC is speeding-up the process of becoming a party to the Joint Convention. The paper presents the current strategy for the safe management and disposal of radioactive waste, including Spent Sealed Radiation Sources (SRS) [5,6], spent fuel, the national policy for the back end of the fuel cycle [7,8] and anticipated future trends.

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Spent Fuel Management in Belgium

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In average, about 120 tU are unloaded each year from the Belgian NPP, one half from the Doel power station and the other half from the Tihange one.

Four units are in operation on the Doel site where dry storage in heavy metallic dual-purpose containers is organized. At the end of December 2005, fifty-four containers were in the dry containers storage building put into service in 1995. The capacity of the containers in comprised between 24 and 37 places. Five types of containers are presently in storage, depending on the unit(s) where the spent fuel elements come from and depending on their burnup. Three pressure sensors permanently monitor the helium pressure in the void between the metallic gaskets of the primary lid. A signal is triggered, should the pressure fall under a preset alarm level.

The material used to build the baskets until now makes use of boronated extruded aluminium of which boron was enriched with B-10. The next generation of baskets will implement a powder based technology allowing higher natural boron content incorporated in an aluminium matrix, without B-10 enrichment.

Each cask loading gives rise to a thorough gamma and neutron mapping in order to prove compliance with transport regulations and to add, if necessary, removable additional polyethylene shielding plates in dry storage conditions.

The operating license of the storage facility requires that the dry containers shall be qualified for spent fuel transport. Consequently the validity of the approval certificates of the different cask models is periodically extended with regard either to their content or only to their time limits.

Three units are in operation on the Tihange site. After a deactivation period comprised between 2 and 5 years, the spent fuel elements are transferred to a centralized storage pond, located on the same site and able to accommodate 3,700 fuel assemblies. The pond is in operation since 1997. A shuttle, partly filled with water and able to contain up to 12 elements, transfers the fuel elements from the deactivation ponds to the storage pond. At the end of 2005, about 1,500 fuel assemblies had been transferred to the centralized pond.

Until now no problem regarding either the tightness of the gaskets in the dry storage containers or a detrimental evolution of spent fuel in the wet storage has been encountered.

Following from a debate in Parliament led in 1993, reprocessing and direct disposal are put on an equal footing : reprocessing contracts have been concluded in the years 1976-1978 for a quantity of 670 tU. The resulting conditioned waste is presently repatriated.

A study aimed at evaluating the feasibility of conditioning the spent fuel in a domestic facility was developed in the aftermath of a resolution decreed by the Parliament in 1993. A limited R&D program was also conducted in view of validating the most sensitive steps in the process. The study was focused on technical, economic, safeguards, environmental and waste production aspects.

In parallel ONDRAF/NIRAS, the National Agency for Radioactive Waste and Fissile Material, extended its R&D program on the disposal of medium- and high-level waste to the specific aspects of the spent fuel.

Management of Spent Fuel at Kozloduy NPP

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Kozloduy Nuclear Power Plant (NPP) has been constructed and commissioned in three stages.

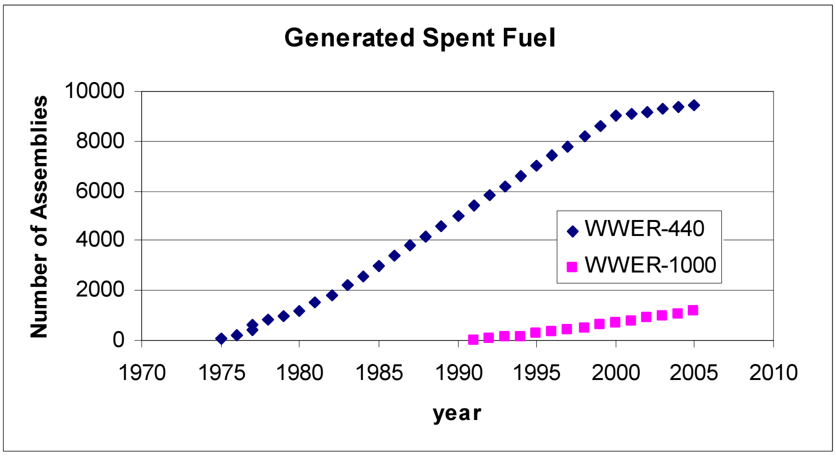
The first part of the plant includes units 1 and 2, which were commissioned in 1974 and 1975 respectively. These are the standard type WWER -440/B-230s. Following a decision of the Bulgarian Government, the units were closed for decommissioning in December 2002.

The second part of the plant consists of units 3 and 4, which were commissioned in 1980 and 1982. They are upgraded WWER -440/B-230 units. As per decision of the Bulgarian Government, these units will be finally shut down for decommissioning in December 2006.

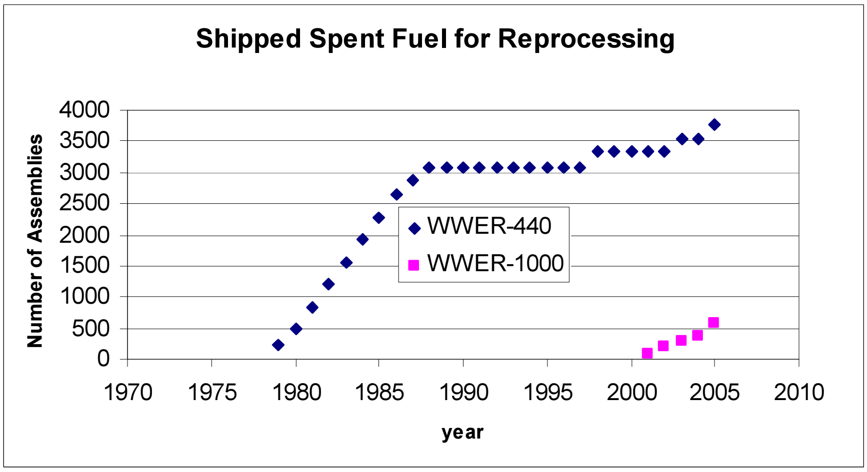
The third part comprises units 5 and 6, which were commissioned in 1988 and 1993. These are WWER-1000/ B-320 units.

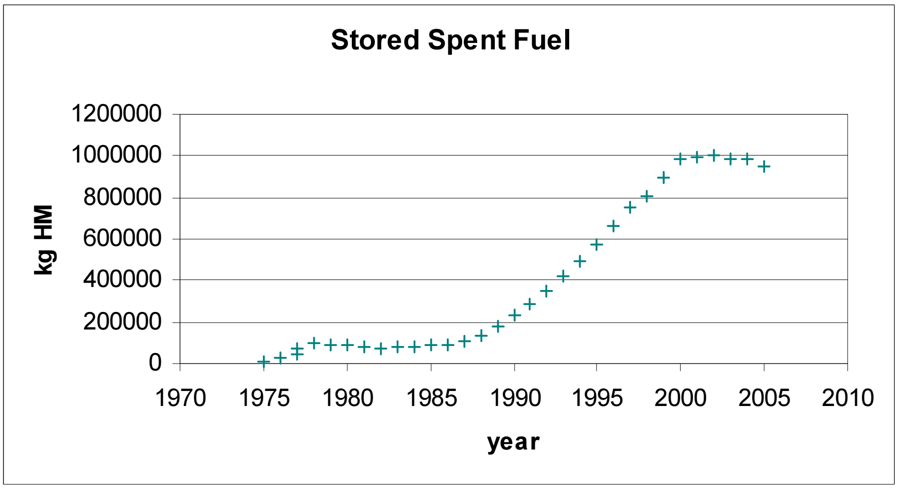
Kozloduy NPP has generated more than 400 billion kilowatt-hours of electricity from the beginning of its operation to the end of 2005.

1654 tHM (metric tonnes of heavy metal) of spent nuclear fuel (SNF) assemblies have been generated as a result of the operation of the units. This amount is allocated in 9470 SNF assemblies from WWER-440s and 1203 SNF assemblies from the WWER-1000 units.



The original project envisaged that the SNF should be shipped back to the manufacturing country after a three-year storage period. Following this concept, by the end of the 1980s, the SNF from WWER-440s had been transported back on regular basis. The balanced SNF amount routinely stored at Kozloduy NPP site was about 200 tHM. The subsequent 10-year break in SNF shipments increased on-site SNF storage up to 1000 tHM. Thereafter, by the end of 2005, 700 tHM distributed in 3774 WWER-440 SNF assemblies and 586 WWER-1000 SNF assemblies were shipped. Following the resumption of SNF shipment for reprocessing, the amount on-site at Kozloduy NPP has been stabilized at a level of about 950 tHM.





A spent nuclear fuel storage facility was built and commissioned at Kozloduy NPP in 1990. It is an ‘underwater’ type of storage facility and consists of four pools having a total capacity of 4920 assemblies. These assemblies are placed in baskets. The number of baskets allowed is 168. The facility is capable of storing both WWER-4400 and WWER-1000 spent nuclear fuel assemblies.

Construction of a ‘dry’ SNF storage facility has been planned. The spent nuclear fuel will be kept in casks placed within a building with natural ventilation. This storage facility will be constructed in two stages. The first stage will afford storage of 2800 WWER-440 SNF assemblies. The second stage will provide storage of 5200 WWER-440 SNF assemblies and 2500 WWER-1000 SNF assemblies.

In this way, Kozloduy NPP management would like to keep open various options for spent nuclear fuel management: reduction of the SNF amount on the site (by transport for reprocessing under favourable market conditions) or long-term safe storage of SNF.

Safety Philosophy at Ontario Power Generation Used Fuel Dry Storage Facilities

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Ontario Power Generation (OPG) operates the Pickering Waste Management Facility (PWMF) and Western Waste Management Facility (WWMF) where OPG has been storing 10-year or older used fuel in Dry Storage Containers (DSCs) since 1996 and 2003 respectively. OPG currently has 5300 tonnes of spent fuel in dry storage. A construction licence for a third facility, the Darlington Waste Management Facility (DWMF), was obtained in August 2004 and the facility is expected to start operations in late 2007.

The used fuel dry storage safety philosophy embodies the defence-in-depth approach to keep radionuclide emissions and gamma radiation dose rates within the regulatory limits and As Low As Reasonable Achievable (ALARA).

The defense-in-depth approach is accomplished with multiple barriers between the used fuel and the public during each stage of the used fuel dry storage process. Each barrier independently provides a measure of safety toward preventing the release of radioactive materials as well as effective neutron and gamma shielding:

- the uranium dioxide matrix effectively contains the radionuclides present in the at-least-ten-year-cooled used fuel (either under wet or dry storage conditions), except for the free fractional inventory of tritium (in vapour form) and krypton-85 (which is a gas);
- the fuel cladding additionally contains the free fractional inventory of tritium and krypton-85 that would otherwise be available for release;
- a further layer of defence is provided during loading the DSCs by the water in the Irradiated Fuel Bay (with the exception of noble gases, the bay water traps fission products that may be released from the fuel cladding) and also provides neutron and gamma shielding;
- the DSC Transfer Clamp with the elastomeric seal are designed to retain a slightly sub-atmospheric pressure in the DSC cavity prior to on-site transfer;
- the DSCs are designed to prevent or reduce to an acceptably low level the escape of fission products in the unlikely event a fuel failure occurs under normal or accident conditions and to protect the fuel from extensive damage from missiles or falling objects during abnormal or accident conditions;
- the reinforced concrete and steel construction of the DSCs provides neutron and gamma shielding throughout the used fuel dry storage process;

- the processing and storage buildings also provide an additional barrier to further reduce gamma dose rates to the public during processing and storage of the DSCs.

The focus is on the prevention of fuel damage during the entire used fuel dry storage operation, to ensure that fission products remain contained within the fuel elements. If an abnormal event or accident occurs, then the used fuel dry storage process provides sufficient mitigation and accommodation measures to ensure sufficient barriers remain intact so the radioactive releases remain below the Canadian Nuclear Safety Commission (CNSC) regulatory requirements.

Safety assessment of each used fuel dry storage facility and the associated systems and operations is required to support each request for regulatory approval to construct and operate the facility. The objective of the safety assessment is to assess the dose consequences to the public and the workers under normal operation and postulated credible accident scenarios and to confirm that dose rates are below the regulatory dose limits for the workers and members of the public.

When assessing malfunctions and accidents, postulated external and internal events are considered. Consideration is also given to the design basis accidents of the specific nuclear generating station that could affect the used fuel dry storage operations as the waste sites are in close proximity to the nuclear operating stations. The probability of occurrence for each initiating event considered is calculated and for those events deemed credible (i.e. frequency $> 10^{-7}$ events per year), a bounding fuel failure consequence is predicted. Given the chemical characteristics of the radionuclides in used fuel, the design of the CANDU fuel and the conditions inside the DSC, then in the event that a used fuel bundle should become damaged during used fuel dry storage operations, the only significant radionuclide species that are volatile are krypton-85 and tritium. Release of these radionuclides is considered in calculating public and worker doses.

For the DWMF safety assessment, the worst case scenario assumes 30% fuel failure within a DSC as well as failure of the DSC transfer clamp seal allowing the release of 30% of the Krypton-85 and tritium available. The consequences to the public from this release were concluded to be less than 0.2% of the regulatory limit.

Preparative Works for the Spent Fuel Storage Facility at the Temelin NPP in Czech Republic*

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The Temelin Nuclear Power Plant (NPP) has operated two VVER-1000 nuclear power reactors since June 2002 and April 2003. The capacity of the spent fuel storage pools in the reactor units enables the operation of Temelin NPP until 2014.

Storage considerations begin with the document called "Concept of the disposal of radioactive waste and nuclear spent fuel", as agreed by the Czech government in May 2002. This document stipulates basic principles of the nuclear fuel cycle back-end strategy:

- Spent fuel shall be stored in dry storage facilities, in storage casks or in dual-purpose casks (transport and storage).
- Priority is set on placing the storage facilities at the nuclear power plants. Skalka (situated about 160 km from the Temelin area) will serve as the backup site (underground storage with horizontal access shaft) to at-reactor sites.
- Contemporaneously with the preparations for the deep repository, possibilities of nuclear spent fuel recycling and adopting new technologies aimed at decreasing the volume and toxicity of the nuclear spent fuel will be pursued.
- The deep repository shall be put into operation by 2065.

The technical approach for the spent fuel storage facility (SFSF) in Temelin is very similar to spent fuel storage at the Dukovany NPP site, the first part of which has been in operation since 1995. A dry storage facility concept with dual-purpose storage/transport casks was adopted. This concept is the most flexible and enables, in the case of urgency, transport of the spent fuel immediately. This approach makes transshipment unnecessary and is more acceptable to the public.

The difference between the Dukovany and Temelin concepts consists in the storage building arrangements. A double section arrangement of the storage area is being considered for the Temelin facility, where each storage section has its own crane. The taller receiving and service portion of the building is also provided with its own crane.

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* paticka
† paticka u Coufala
‡ paticka u Brzobohateho

The main reasons for these differences on the part of the Temelin SFSF lie in limited site space and higher requirements for stability and robustness of the building against external impacts.

The preparative works for the 1370 t (metric tonnes) SFSF at Temelin NPP (based on 30 years of operation) was started in 2002 by feasibility studies for various storage locations. The EIA (environmental impact assessment) process including collection of remarks and objections in Austria and Germany took more than 2 years. In order to meet EIA requirements, an independent study evaluating potential consequences of a big transport aircraft crash upon the storage building was elaborated in 2004. Together with the EIA process, a procedure took place in 2005 in accordance with article No. 37 of the Euratom Treaty as well as activities aiming at the site approval to be issued by the State Office for Nuclear Safety. A tender procedure to identify a contractor for the casks in question proceeds at the present time.

According to the project plan, the site permit is supposed to be obtained in 2006. Approval of construction is planned for 2009 and the storage facility will be put into trial operation in 2014.

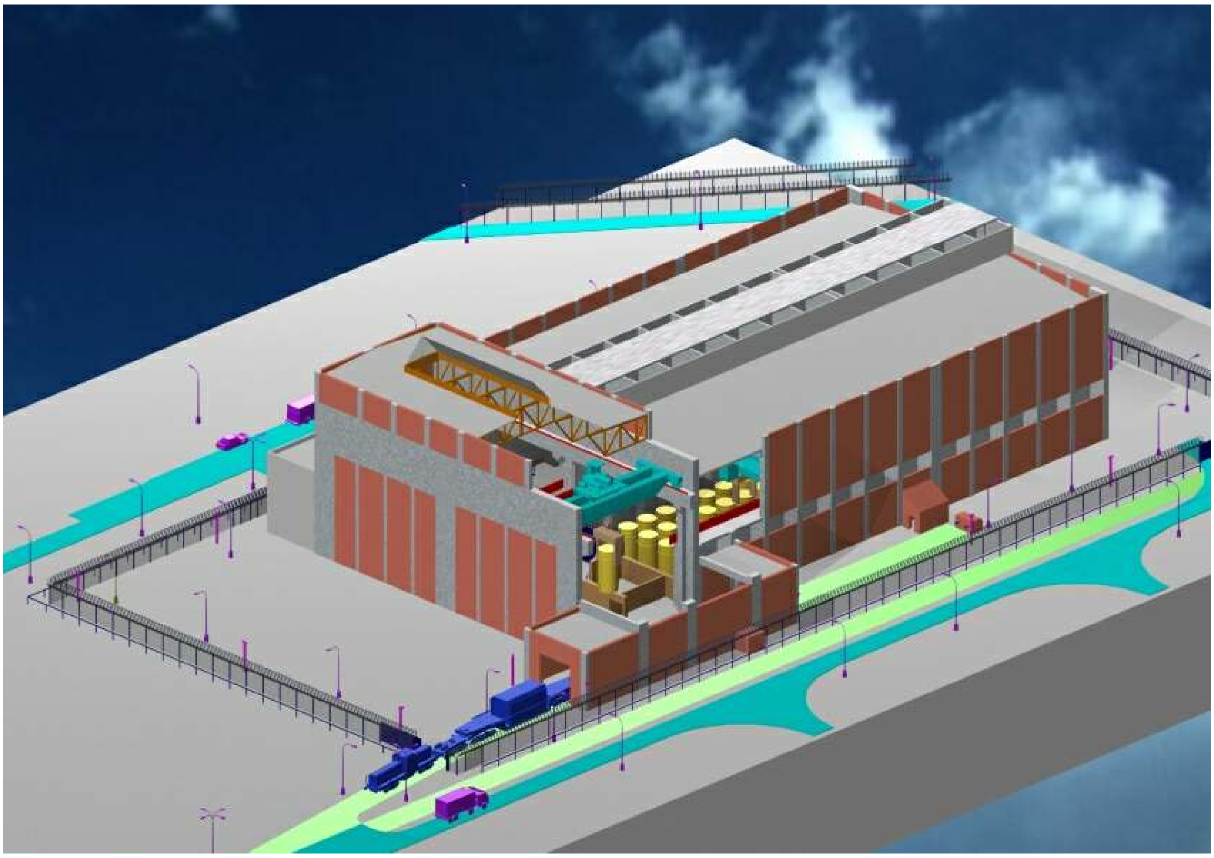

FIG. 1. SFSF at Temelin NPP

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FEM Transient Temperature Computer Analyses of CASTOR 440/84

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This paper describes code development processes started at the Nuclear Research Institute (NRI) Rez plc under sponsorship of CEZ, a. s., including documentation of model validation and preparation of a practical engineering tool for the utility. Testing and implementation of the product occurred at CEZ headquarters and NPP Dukovany, accompanied by evaluation by the State Office for Nuclear Safety of the Czech Republic for compliance with legal requirements (VDS 30).

The original two-dimensional (2D) and three-dimensional (3D) thermo-physical steady state computer models for CASTOR containers filled by VVER 440 fuel were developed at the NRI Rez facility during recent years. The main goal was to prepare a tool for realistic simulation of thermal fields in CASTOR containers to be able to evaluate the maximum temperature for fully loaded containers with fuel assemblies with various thermal power and burnup levels. The idea was to evaluate margins relative to the limiting allowed temperature.

The programme was adapted to the transient domain later, to simulate the most critical industrial operations during CASTOR loading and handling operations. Controlled values included temperature peaks and ranges as well as time durations for the manipulations.

The original 2D transient finite element model (FEM) developed at the NRI was adapted to the COSMOS/M programme, achieving successful results when validated against industry measurements as well as in comparisons between both codes. An example of results is shown in Fig. 1 showing a 2D temperature colour profile in a cross-section of the container.

Practical implementation at CEZ headquarters and at the Dukovany NPP was accomplished this year with on-site acceptance tests to check calculations against all relevant procedures and requirements.

In summary, CEZ, a. s., and NRI are able to make industrially-oriented, practical, FEM-based CASTOR thermal evaluations for homogenous and heterogonous CASTOR loadings with reasonable accuracy.

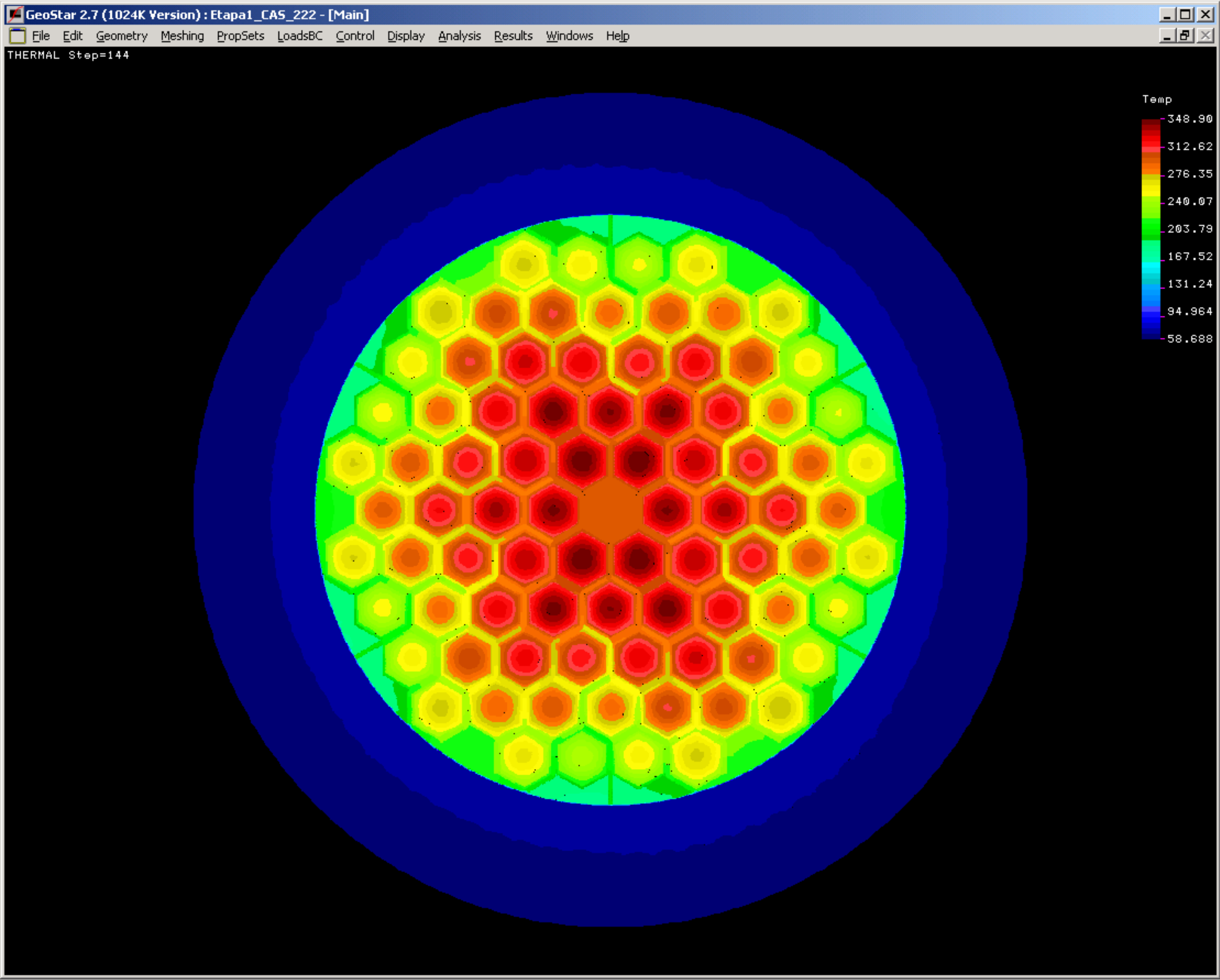

Fig. 1.Example of 2D temperature profile (cross-section of the container; time step 144; max temperature)

Licensing of CASTOR© 440/84M Cask for Transport and Storage of Spent Fuel

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Dry cask storage technology for the temporary placement of spent fuel from power reactors has been used in the Czech Republic since December 1995, when trial operation of the Interim Spent Fuel Storage Facility at the Dukovany (ISFSF Dukovany) nuclear power plant (NPP) site started. This storage technology is based on the use of CASTOR© 440/84 casks, each of which accommodates 84 WWER-440 fuel assemblies.

As the storage capacity of ISFSF Dukovany is limited to 600 tHM (metric tonnes heavy metal) in 60 CASTOR© 440/84 casks and as it will be fully exploited in mid 2006, activities related to the design, construction and operation of a new Spent Fuel Storage Facility (SFSF) at the same site (SFSF Dukovany) were launched in the late 1990s. The new SFSF Dukovany will use at least at the beginning of its operation a modified CASTOR© 440/84M cask, which is based on the design of the currently used CASTOR© 440/84 cask. However, several important modifications can be identified in the design of the new cask and are related to:

- improved neutron shielding properties,
- modified trunnion construction,
- optional use of a third welded lid, and
- a new fuel basket design.

The licensing procedure (officially called "design approval") for the CASTOR© 440/84M cask started on 23 July 2003, when an application for licensing was received by the national regulatory body of the Czech Republic – State Office for Nuclear Safety (SÚJB). The licensing procedure lasted about two years and included the preparation of independent reviews [1] on:

- cask inventory and criticality calculations,
- thermal calculations,
- shielding analyses and radiation safety,
- the confinement system, and
- mechanical analyses of the cask and its components.

This procedure concluded with issuance of the cask licence for railroad transport and storage on 12 July 2005.

CASTOR© 440/84M casks can be loaded in three basic configurations:

- homogeneously with 84 WWER-440 fuel assemblies with burnup up to 45 GWd/tU (gigawatt days per metric tonne uranium) and thermal output less than or equal to 290 W (watts),

- heterogeneously, when central positions hold up to 36 fuel assemblies with burnup up to 50 GWd/tU and remaining positions are occupied with spent fuel with burnup up to 45 GWd/tU. Thermal output of each assembly is less than or equal to 290 W,

- heterogeneously depending on the fuel assemblies inventory and neutron and gamma flux, when the thermal output of up to 6 centrally located fuel assemblies can reach 340 W each. The thermal output of remaining assemblies is less than or equal to 290 W each.

The thermal calculations [2] were performed not only for standard transport and storage conditions when the surface temperature shall not exceed 85 °C, but also for transient conditions during drying of the cask. The cladding temperature limit was set to 350 °C and, for different total thermal outputs of the loaded cask and He content in the helium-vapour mixture, the time when the cladding temperature reaches the temperature limit was calculated. The results of transient temperature calculations were used in operational instructions for cask drying.

Shielding assessment is based on the inventory of so called reference spent fuel, which has the most conservative properties from the point of view of gamma and neutron radiation. Axial distribution of radiation fields was taken into account together with the activity of fuel assembly construction material not containing fuel pellets. The calculations were performed by the MCNP computer code and the results compared with limiting values according to IAEA regulations [3].

The confinement system was assessed taking into account release of fission gases and volatile radionuclides as 3H, 85Kr, 129I, 134Cs and 137Cs. For normal and accidental situations the release of radioactive material is calculated and the compliance of release rates with IAEA TS-R-1 values is provided.

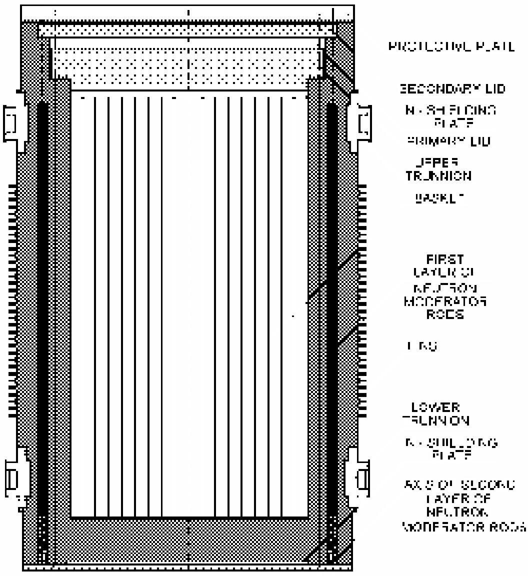

FIG. 1. Storage configuration of CASTOR© 440/84 M cask

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License Procedure for Spent Fuel Storage Facilities in the Czech Republic from the Competent Authority Point of View

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The Policy, approved by the Czech Government on 15 May 2002 (Government Resolution No. 487/2002), is the fundamental document defining a strategy of the State and its agencies in spent fuel and radioactive waste management through 2025.

The main principles of the Policy for the spent fuel management are:

- Spent fuel management is provided by nuclear power plants (NPPs) authorized for operation in the Czech Republic. Spent fuel shall be stored in dry storage facilities at the NPPs, in storage only casks or in transport and storage casks.
- Possibilities for spent fuel reprocessing are monitored and assessed, as well as the use of new technologies leading to the reduction of spent fuel volume and toxicity. A deep repository shall be put into operation in 2065.
- The costs of activities associated with disposal of spent fuel are paid from the nuclear account, a financial source created by generators of spent fuel in agreement with the Atomic Act and established government Order. The Ministry of Finance manages the nuclear account, as a part of the state financial assets and liabilities. This assures that the costs of disposition for wastes generated now will not be transferred to future generations.
- The general public is kept informed about the Policy and about its fulfillment.

The license procedure for spent fuel storage facilities is governed by three acts:

- Act No. 100/2001 Coll., on assessment of impacts on the environment,
- Construction Act (No. 50/1976 Coll.), and
- Act No. 18/1997 Coll. (Atomic Act)

The procedure consists in practice of four stages:

- Assessment of impacts of the planned construction on the Environmental Impact Assessment (EIA) process,
- Siting decision,

- Construction permit, and
- Operations permit.

Interim spent fuel storage facility at Dukovany (ISFSF)

The first preparatory works started in May 1991. In October 1992, a public hearing took place to discuss the environmental impact of the ISFSF based on an EIA study. In December 1992, after evaluation of the results of the public hearing, a positive judgement from the Ministry of the Environment was issued recommending site approval, which was issued in August 1993. In November 1993, following the evaluation of a Preliminary Safety Report, the State Office for Nuclear Safety issued the agreement for issuing a construction permit for dry type storage. Final approval of the construction permit was issued in June 1994. On 23 January 1997, the State Office for Nuclear Safety issued approval for 10 years of ISFSF operation.

Spent fuel storage facility (SFSF) at Dukovany

Preparation for the SFSF at Dukovany NPP started in 1997. The EIA process took 2 years. The Czech electricity supply company, CEZ, a. s., received the site permit in 2000. The State Office for Nuclear Safety issued permission for starting construction in November 2002, resulting in construction approval issued in 2003. Duration of the construction work was 24 months. Approval of the State Office for Nuclear Safety for trial operation is expected during 2006, while the initial 10-year operation period of the ISFSF is planned to begin in late 2007 or 2008.

Spent fuel storage facility (SFSF) at Temelin

The first steps was started in 2002 by conducting feasibility studies for various SFSF locations. In order to meet the EIA process requirements (including collection of remarks and objections from Austria and Germany), an independent study evaluating potential possibilities and consequences of a big transport aircraft crash upon the storage building was developed in 2004. Together with the EIA process, a procedure according to article No. 37 of the Euratom Treaty and activities aiming at the site approval to be issued by the State Office for Nuclear Safety took place in 2005. Issuance of the site permit is expected in 2006, with construction approval expected in 2009 and trial operations planned for 2014.

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SKODA VPVR/M Cask for Spent Nuclear Fuel from Research Reactors

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The LVR-15 research reactor at Nuclear Research Institute Rez plc was designed in 1957. The original fuel type was EK-10 with 10% enrichment and of Russian origin. Subsequently, in 1974, the inner parts of the reactor were reconstructed for the utilizing of a highly enriched uranium (HEU) fuel, namely, the Russian IRT-2M fuel assemblies (FA) with 80% ^{235}U enrichment. From 1989 to 1996, the reactor was rebuilt to be operated at higher power, up to 15 megawatts (MW). The type of FA was retained but the enrichment was changed to 36 % with the use of the IRT-2M fuel type. The transformation took several reactor cycles using gradually different mixed core configurations. Since 1998, the reactor has been operated only with the IRT-2M (36%) FA.

The policy for spent fuel management in NRI is based on using three facilities for spent fuel assemblies from the LVR-15 reactor. The first one, the current at-reactor (AR) facility, is a pond in the reactor hall next door to the reactor. The second one is the away from reactor (AFR1) pond in the reactor auxiliary building. The third facility, (AFR2), is a pond at the institute's high-level waste storage (HLWS) facility.

During the time period of the former Soviet block, several different types of research reactors were designed and operated, mostly utilizing HEU of Soviet origin. There have been seventeen countries with research reactors operating on Soviet- or Russian-supplied fuel – Belarus, Bulgaria, China, Czech Republic, DPRK, Egypt, Germany, Hungary, Kazakhstan, Latvia, Libya, Poland, Romania, Serbia and Montenegro, Ukraine, Uzbekistan and Vietnam. FA enrichment varies from 2% to 90% ^{235}U.

In 2004, the U.S. and Russian Federation signed an agreement concerning the repatriation of Russian-origin high-enriched uranium (HEU) research reactor fuel to Russia. The primary goal of the Russian Research Reactor Fuel Return (RRRFR) program is to eliminate highly enriched uranium (HEU) stockpiles and persuade eligible countries to convert their research reactors from HEU to low enriched uranium (LEU). Under the RRRFR program, Russia has agreed to take back spent and fresh fuel from research reactors so long as the reactor operators agree to convert the reactors to operate on LEU or shut down. So far, eight shipments have repatriated Russian-origin fresh HEU fuel under the RRRFR program, from Serbia, Romania, Bulgaria, Libya, Uzbekistan, the Czech Republic, and Latvia.

The SKODA VPVR/M cask had to fulfill safety requirements mandated by legislation regarding shielding quality, subcriticality and thermal cooling. Sources of gamma rays, neutron sources, heat sources and isotope inventory were determined by the program

ORIGEN. The revised version of ORIGEN incorporates updates of the reactor models, cross sections, fission product yields, decay data, and decay photon data, as well as the source code. Shielding calculations of the SKODA VPVR/M cask were provided by the program DORT in one-dimensional (1D) and/or two-dimensional (2D) geometries. A revised multigroup cross-section library (BUGLE-96, based on ENDF/B-VI Release 3) used in these calculations was produced for light water reactor shielding and reactor pressure vessel dosimetry applications. The cross-section processing methodology is the same as that used for producing BUGLE-93 and is consistent with ANSI/ANS 6.1.2. According to the legislation requirements for the storage and transport casks for the spent fuel, the subcriticality of the system has to meet the requirement of $k_{eff} \lesssim 0.95$. The MCNP program for system criticality calculations was used in the mode KCODE with the DLC-200 pointwise cross-section data library. All calculations were done for the fuel with maximum multiplication capability, i.e. fresh fuel.

The Škoda VPVR/M cask of the B(U)F type, identification designation CZ/048/B(U)F–96, meets the applicable International Atomic Energy Agency regulations [1], as well as applicable international transport regulations which refer to these IAEA regulations.

The Škoda VPVR/M cask, identification designation CZ/048/B(U)F–96, is intended for road and rail transport, inland waterways and sea transport of 36 undamaged fuel assemblies (FA) with spent nuclear fuel (SNF) from research nuclear reactors or 36 stainless containers with damaged FA and/or with fuel elements from damaged FA and/or with parts of fuel elements from damaged FA. The cask can also be used to store:

- 36 undamaged FA
- 36 stainless containers with damaged FA
- 36 stainless containers with fuel elements or parts of damaged FA

The cask manufacturer is ŠKODA JS a. s.

General nominal parameters of the Škoda VPVR/M cask:

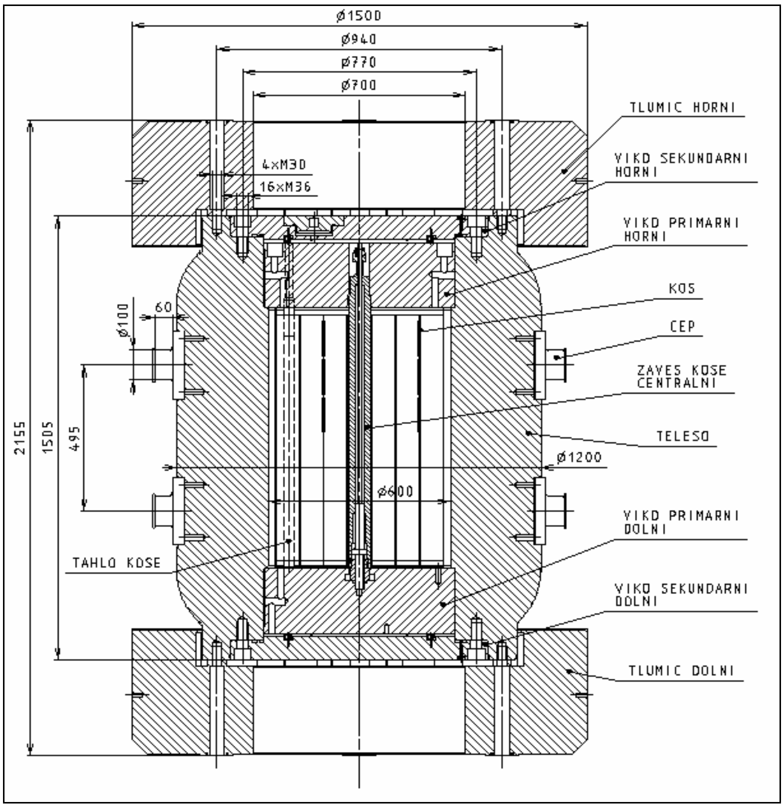
Outer cask body diameter	1 200 mm
Height without shock absorbers	1 505 mm
Height with shock absorbers	2 155 mm
Cask cylindrical wall thickness	300 mm
Maximum weight of loaded cask	
- without shock absorbers	11 150 kg
- with shock absorbers	12 390 kg
Maximum weight of loaded FA with SNF	450 kg.

Cask radioactive contents admissible for transport

a) Types of LA (loaded assembly) with SNF which can be loaded into cask	EK-10, S-36, VVR-M, VVR-M2, VVR-M5, VVR-M7, VVR-(S)M, IRT-2M, IRT-3M, TVR-S
b) Number of LA in cask	max. 36
c) Total activity of all LA in cask	max. $3.93.10^{15}$ Bq
d) Residual thermal power of one LA in cask	max. 37.5 W
e) Total LA residual thermal power in cask	max. 450 W
f) Initial nominal enrichment of fresh nuclear fuel with isotope ^{235}U	max. 90% by weight
g) Weight of ^{235}U in every LA in cask	max. 500 g

Illustration of Škoda VPVR/M Cask

Identification Designation CZ/048/B(U)F-96



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The System of Storage and Capsulation of Spent Fuel in the Egyptian Research Reactor (ET-RR-1)

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In Egypt, the first research reactor (ET-RR-1) has been used for a long time in conducting research work in the fields of reactor physics, heat transfer and thermohydraulics. Also, it is used in the production of radioisotopes for medical purposes and chemical researches. Due to the operation of the reactor for about 40 years, the need for larger and modern spent fuel storage arises [1]. Defects appearing in the fuel elements motivate an investigation of the reactor fuel and the design of a new system of encapsulation of the defective fuel elements for long-term storage.

The new spent fuel storage facility is a structure containing the storage and a so-called receiving and fuel handling area arranged in a separate building. Fuel storage is in a stainless steel tank filled with water and located in a concrete pit under the ground level. The tank is covered by slabs made of steel and concrete. Aluminium storage tubes are fixed in the tank for vertical storage of the fuel assemblies in a rectangular matrix arrangement. The capacity of the storage tank is up to 176 fuel assemblies [2].

In the design of the storage, both thermal and physical aspects are taken into consideration. The temperature, conductivity and the pH value of the water are measured. The water supply and air-ventilation of the reactor are used also for the storage. The storage ventilation depression as well as the water level is always maintained at the design values for the safety of the personnel and the facility. Also, the water is filtered regularly using two mechanical filters, one of them is operating and the other is spare [3].

The encapsulation system is designed to prevent the leakage of fission fragments from the defective assemblies to the water of the storage facility. The defective fuel assembly is encapsulated in a tightly closed aluminium tube filled with nitrogen containing 5% of helium as an inert gas to prevent the corrosion of fuel rods. The encapsulation tube is provided with a non-return valve to avoid escape of nitrogen. The encapsulation system includes two casks, one for the fuel transport and the other for encapsulation process. The system contains a drying unit, nitrogen charging unit and a welding mechanism to weld a special cap containing the non-return valve. The leakage of nitrogen gas is tested by an apparatus which is very sensitive to helium gas. The encapsulated assembly is then returned to the storage facility for long term storage.

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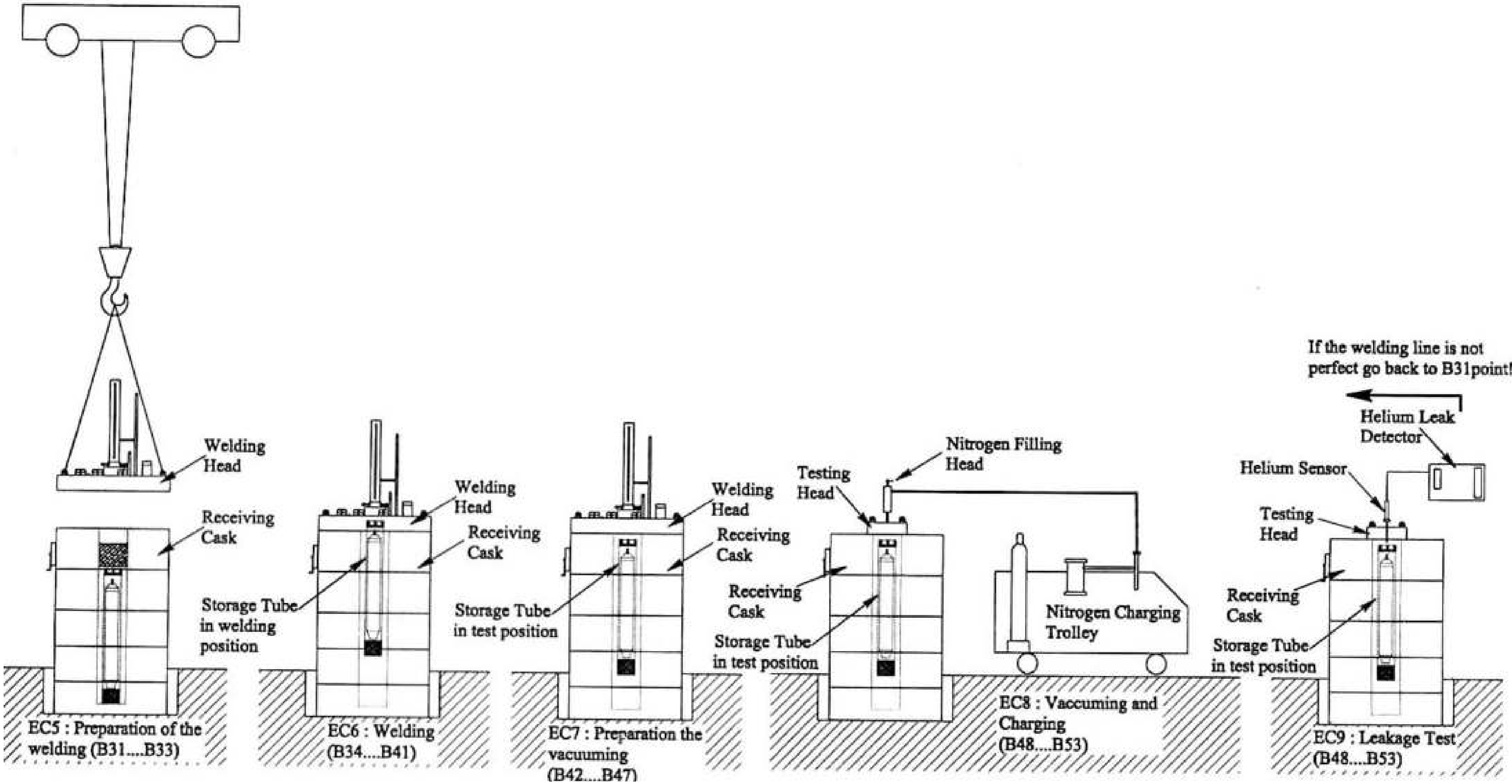

Flowchart of the Spent Fuel Encapsulation Procedure

Failure of Closure Lid Welds for the Nuclear Materials Waste Packages

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The long-term structural integrity of waste packages is required to assure the safe storage and transportation of spent nuclear fuel and high-level waste.

Cask storage is one of the primary means for storing spent fuel. It is also recommended to use transportable storage casks from an economical point of view.

Transportation packaging is divided into five categories:

(a) Low Specific Activity (LSA), (b) Type A (Fissile), (c) Type B (Normal Form), (d) Type B (Special Form), and (e) Type B (Fissile). Dry spent fuel storage systems are grouped as either metal shielded type or concrete shielded type systems.

This paper summarizes evaluations performed for lid welds of one of the transportation package designs. Due to the potential for high residual stress caused by welding and the possibility of water contacting portions of the outer weld surfaces, stress corrosion cracking could occur. Consequently, the magnitude and though-wall distribution of weld residual stress in the final un-annealed closure welds is an important consideration and was the focus of this study. The waste-package design comprises concentric inner and outer cylindrical shells, each with a closure lid welded to its respective shell. The welds of the inner and outer closure lids are close to one another. The inner and outer shells are fabricated using a thermally enhanced fit-up process. The Non-linear Finite Element Method is used to evaluate the effect of a shrink fit, the proximity of the two welds, and the residual stress distribution due to the weld configuration. The possibility of crack generation due to the resultant stress distribution is discussed. Results of this study can be used to optimize the design of the associated welds and increase their long-term reliability by minimizing the potential for stress corrosion cracking.

Capacity Extension Plans for TVO-KPA-STORE in Finland

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A clarification on plans to extend spent fuel storage capacity at Olkiluoto (KPA-store) was made in February 2002. Since then the decision was made to construct a new plant unit, Olkiluoto 3 (OL3), which is planned to start its operation in year 2009. The spent fuel of the new pressurized water reactor plant will be stored in the KPA-store extension part. OL3 will be an European Pressurized Water Reactor (EPR) design with a net power of 1600 MWe and an expected life time of 60 years. Also, the first and second Olkiluoto units have extended lifetime of 60 years.

Spent fuel disposal will start in 2020. However, the need for spent fuel storage will increase since the second Olkiluoto plant unit will permanently shut down in 2040 (Fig. 1). Extension of KPA-store is not necessary if no new plant is constructed at the Olkiluoto site. However, the provision for further extensions is not excluded.

It looks reasonable to extend the KPA-store with four additional pools. Three pools would be provided with fuel racks for OL3 spent fuel (Fig. 2 and Fig. 3). The fourth extension pool would be reserved for emergency unload purposes.

The investment costs for the KPA-store extension are about 29 Million Euros with the dense fuel racks included. The storage extension should be available in year 2012.

Sensitivity analyses show that if a new plant unit is constructed in coming years at the Olkiluoto site, then construction of a common new intermediate storage for the spent fuel is worth considering.

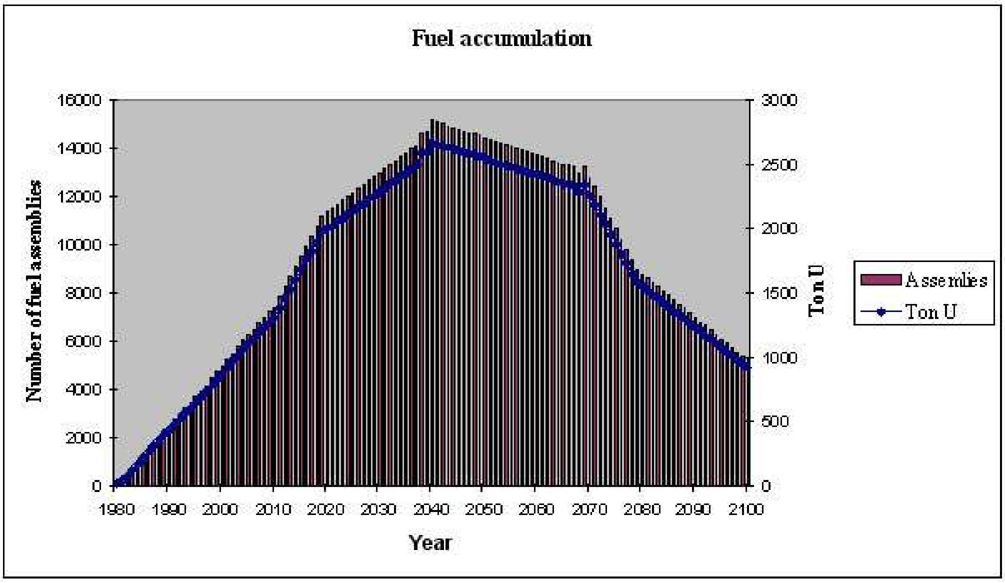

FIG. 1. Equivalent fuel accumulation of Olkiluoto plant.

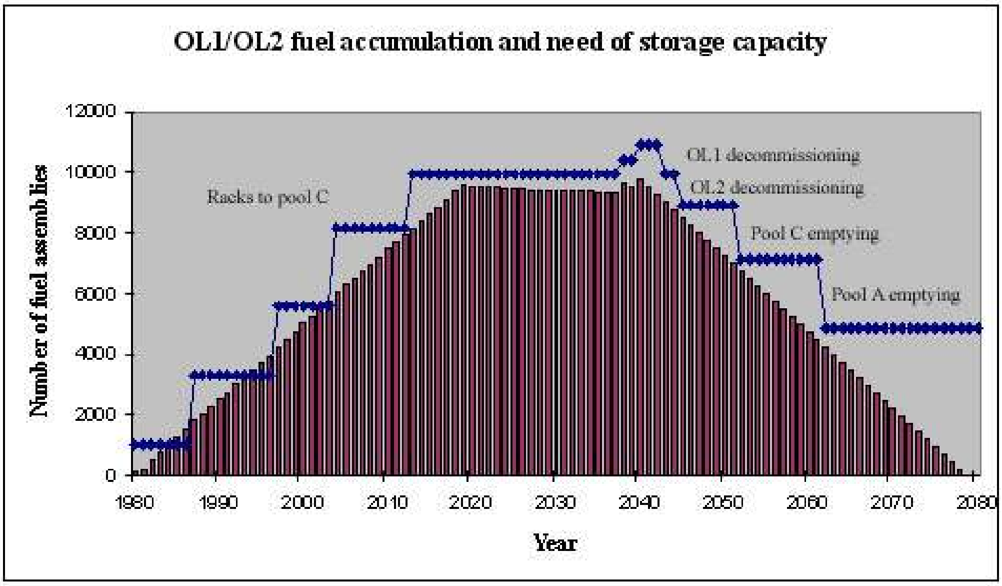

FIG. 2. Capacity extension for Olkiluoto 1 and Olkiluoto 2 units.

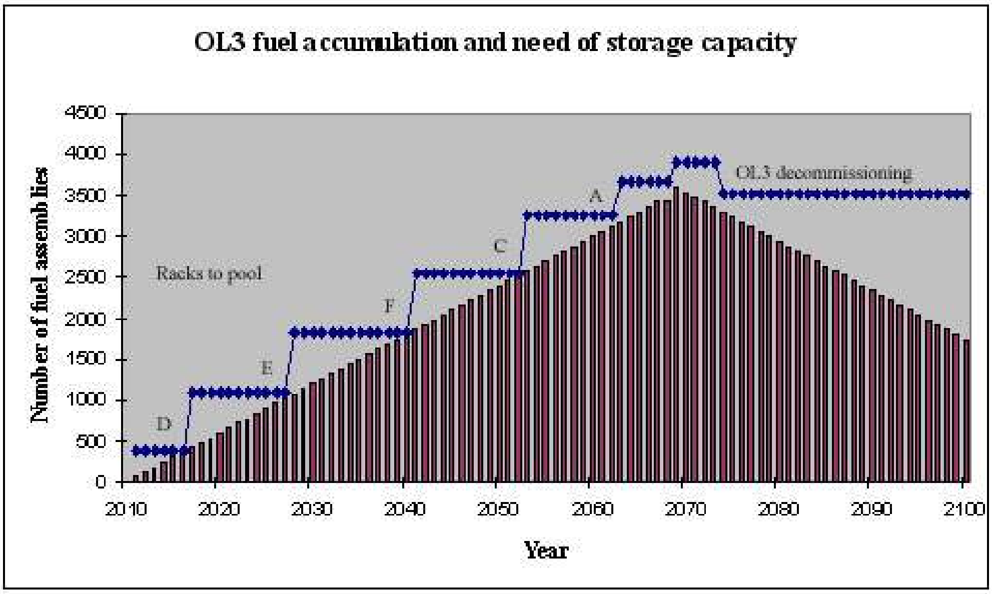

FIG. 3. Capacity extension for Olkiluoto 3 unit.

R&D for Transport and Storage of Spent Fuel, Anticipation of the Needs Linked with Evolutionary Designs

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There is a continuous improvement in fuel design associated with the increase of reactor performance. This is notably the case for evolutionary reactors such as the European Pressurized Water Reactor (EPR) but also for existing reactors. Competitiveness relies also upon optimal fuel utilization, including the quick removal of fuel assemblies from reactor cores. At the same time, these fuel designs show enhanced safety features.

These evolutions lead to new challenges. Management of spent fuel requires more performance and innovative solutions. Spent fuels are no longer the same, and now include higher enrichments and high burn up fuels (over 60 000 MWd/tHM) with high heat load and high radiation levels. Management of spent MOX fuel is also a challenging matter. To optimise competitiveness of the nuclear option, anticipation of research and development (R&D) needs is necessary. This is why Cogema Logistics has been implementing a strong R&D program to support new flexible and safe solutions for transport and storage of spent nuclear fuels.

The current R&D program is focusing on the following items:

- New basket designs and materials with high boron content for sub-criticality (metal matrix composites);
- High performance thermal transfer systems;
- New neutron shielding materials resistant to more severe conditions;
- New shock absorbing covers to keep the G-loads low in all conditions.

The Spent Fuel Storage Solution in Armenia

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TRANSNUCLEAR Inc.'s NUHOMS® system (TRANSNUCLEAR Inc. is a subsidiary of AREVA Group and of TN International) is a totally passive installation which provides shielding, safe confinement and long-term interim storage of spent fuel assemblies. With close to 300 NUHOMS® canisters currently loaded in North America, the NUHOMS® solution has provided answers to various requirements and constraints: adaptation to PWR/BWR spent fuel, high seismic criteria, increase of burnup, increase of heat load, diminution of exposure rates, etc. Though the qualities of the NUHOMS® system have been acknowledged in the US for years, the technology has crossed the Atlantic Ocean, notably with the implementation of NUHOMS® solutions in Armenia.

In 1996, the Armenian Government adopted the NUHOMS® technology for its Armenian Nuclear Power Plant (ANPP), VVER 440, located at Medsamor. The related contract included storage material procurement, and engineering services for the design, construction and operation of the storage system. The storage installation currently accommodates 616 spent fuels.

In 2005, the Armenian Nuclear Power Plant needed additional storage capacity. As the NUHOMS® technology gave full satisfaction during the first stage, it was naturally chosen for this extension. Moreover, the phased construction of the additional storage capacity facilitates the management of ANPP by economizing and distributing the cost for fuel storage over the time span when storage is actually required. Therefore a new contract was signed.

We have taken the benefit of the initial operational experience, and we assume that it could be relevant to share this feedback with the nuclear community on the occasion of a major conference. Hence this paper, that focuses on the following topics:

- the NUHOMS® technology (overview);
- adaptation of this technology to the Medsamor site's requirements (e.g. VVER fuel, sets of several modules);
- feedback from the initial operations;
- extension of the storage capacity.

The Metsamor project involves technical skills from Armenian and French sides and enables local suppliers to play a major role. In a nutshell, this partnership can be seen as a perfect example of mutual understanding and joint effort to a win-win agreement.

Spent Fuel Transport and Storage System for NOK Nuclear Power Plants

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Both NOK nuclear power plants in Switzerland - LEIBSTADT (KKL) BWR nuclear power plant and BEZNAU (KKB) PWR nuclear power plant - have opted to ship spent fuel to a central facility called ZWILAG for interim storage. In the mid-nineties, COGEMA LOGISTICS was contracted by KKL for the supply of the TN™52L and TN™97L transport and storage casks for BWR fuel types. In 2003, KKL also ordered from COGEMA LOGISTICS the supply of six TN™24 BHL transport and storage casks. This paper shows how all the three cask designs have responded to KKL requirements to ship and store BWR spent fuel. In addition, it highlights the already significant operational feedback of the TN™52L and TN™97L casks by the KKL and ZWILAG operators.

In 2004, NOK also ordered three TN™24 GB transport and storage casks for PWR fuel types. These casks are presently under manufacturing.

TN™52L

One TN™52L unit was delivered to KKL in 2000. This TN™52L cask has been used as a transport cask for the shipment of KKL spent fuel to COGEMA La Hague plant. After that, it has been used for the dry storage of KKL spent fuel at ZWILAG. The paper will present the wide range of operations made with the TN™52L cask: loading, routine transport, unloading, maintenance, transport to ZWILAG and storage at ZWILAG. The experience gained from these operations will be presented.

TN™97L

Nine TN™97L units were ordered by KKL. They have all been loaded and successfully stored at ZWILAG. The TN™97L transport and storage cask has the highest payload ever designed world wide for BWR spent fuel. The paper will detail the operational feedback with this "giant", from the cask loading at KKL to its transport and storage at ZWILAG.

TN™24 BHL

KKL has recently opted for the TN™24BH cask design. The TN™24BH was first designed for the fuel elements of the Swiss nuclear power plant of Mühleberg. By taking advantage of TN™52L and TN™97L experience, the TN™24BHL (L for Leibstadt) will fulfil KKL future needs of shipping spent fuel with increasing burn up and enrichment to ZWILAG, while keeping a high payload.

TN^{TM}24GB

For some of its fuel assemblies, the nuclear power plant of Beznau (KKB) has opted for the TN^{TM}24G cask design. The TN^{TM}24G was first designed for the fuel elements of the Swiss nuclear power plant of Gösgen. This paper will show why the TN^{TM}24GB (B for Beznau) fulfils NOK needs: casks available in 2008 with a high payload in spent fuel cooled for long durations.

Onsite Storage Facilities for Spent Fuel from Nuclear Power Plants in Germany

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With the agreement of 14 June 2000 between the Federal Government and the power utilities (ratified on 11 June 2001) the German power industry respected the Federal Government's decision to phase out the commercial nuclear electricity generation. The standard life time of nuclear power plants was limited to 32 years from the date of commissioning. The Act on the Structured Phase-out of the Utilisation of Nuclear Energy for the Commercial Generation of Electricity entered into force on 22 April 2002. The key points of this agreement and the respective act relevant to spent fuel management in Germany are the following:

- Reprocessing will be discontinued and replaced by direct disposal of spent fuel elements.
- The delivery of irradiated fuel elements to La Hague and Sellafield for reprocessing will be terminated by the middle of 2005.

With regard to direct disposal of spent fuel, a remaining period of several decades still needs to be bridged, depending on the availability of a disposal facility and the length of time required for a decrease of the decay heat until disposal. The Federal Government's concept envisages that for this purpose, spent fuel should be placed in containers in storage at the nuclear power plant sites where they are generated. After that they shall be conditioned and disposed of.

By the end of 2003, nuclear licenses had been granted to the decentralised storage facilities for spent fuel assemblies at twelve nuclear power plant sites - Brunsbuettel, Brokdorf, Kruemmel, Unterweser, Grohnde, Emsland, Biblis, Neckarwestheim, Philippsburg, Grafenrheinfeld, Isar and Gundremmingen. The storage facilities are either under construction or already complete (Emsland). It was necessary to erect four temporary storage facilities at four power plant sites as a buffer for six years in maximum until the commissioning of onsite storage facilities will be completed.

They are designed as dry storage facilities in which transport and storage containers loaded with spent fuel assemblies are emplaced. Various types of storage facility design have been licensed. Cooling is procided by passive air convection which removes the heat from the containers without any active technical systems. The leak-proof and accident-resistant containers ensure safe enclosure as well as the necessary degree of radiation shielding and criticality safety during both normal operation and in the case of incidents. The heat transfer from the containers into the environment is enhanced by means of cooling fins. Protection against external impacts, such earthquakes, explosions and aircraft crashes, is ensured by the thick walls of the containers. It was demonstrated and confirmed in the licensing procedure that the containers are suitable for at least 40 years of storage; the licences limit the storage period correspondingly. Due to the planned date for the availability of a disposal site around the year 2030, the question of prolonging the storage period is immaterial.

Basically, two design options were applied for the storage facilities. In the first option the storage building consists of two parts being separated by a wall in the middle of the building. The wall thickness is approximately 85 cm, and the roof thickness approximately 55 cm. In the other concept, the storage buildings have no separating wall. The wall thickness is approximately 120 cm, and the roof thickness approximately 130 cm. Each of the individual storage facilities has a capacity of between 80 and 192 storage positions for suitable transport and storage containers. The storage facility at Neckarwestheim is a special case, where the containers will be stored in two tunnels lined with gunite. This special underground solution was developed to accommodate the specific situation of the site.

As a transitional solution until the on-site storage facilities are complete, an in order to avoid any disposal shortfalls, four nuclear power plants have applied for temporary storage facilities. Theses installations have a capacity of up to 28 storage positions with a prefabricated mobile concrete enclosure for each container. The intention is that the container will be transferred to the respective on-site storage facility for a limited period of time, approximately up to 6 years. The containers, combined with the concrete enclosures, ensure compliance with the admissible dose limits stipulated by the Radiation Protection Ordinance. For the Biblis, Neckarwestheim, Philippsburg and Kruemmel nuclear power plants the licences for the temporary storage facilities have already been granted, and 63 containers have already been emplaced up to December 31, 2005.

The legal basis for the licensing of decentralized storage facilities is given in § 6 of the Atomic Energy Act. All facilities under the regime of § 6 Atomic Energy Act have also to undergo an environmental impact assessment (EIA) when applied for after March 15, 1999. Additionally, a license pursuant to planning and building law according to the respective federal state building regulations is required for the construction of the storage building.

The utilities as generators of spent fuel and radioactive waste are responsible for the development, construction and operation of storage facilities. The responsibility

- for the licensing of nuclear fuel storage rests with the Federal Office for Radiation Protection (BfS);

- for the licensing of construction of a storage building, independent of the type (storage building, storage tunnel, storage area and concrete encasements), rests with the respective federal state building authority; and

- for the supervision during operation rests with the authority of the respective federal state executing the atomic regulatory framework.

Pursuant to § 6 Atomic Energy Act a licence shall be granted if there is a need for such storage and if the "precaution against damages according to the state-of-the-art" as the most extensive complex is verified.

The following subjects have to be regarded, examined and verified:

- Safe enclosure of the radioactive inventory;
- Sufficient shielding;
- Subcriticality; and
- Sufficient dissipation of decay heat.

Aircraft crashes and - following 11 September 2001 - attacks with a hijacked large passenger plane are to be examined as well. Before September 11, 2001, only the crash of smaller military jets has been put into consideration.

Trends in Dry Spent Fuel Storage and Transportation in Germany and the United States of America

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As part of AREVA's effort to provide nuclear fuel solutions, worldwide, for the entire fuel cycle, the company supports an ongoing, global effort to track trends in spent nuclear fuel storage and transportation. Due to the progressing implementation of dry storage in Germany and the U.S., this paper reports in particular on country-specific licensing trends and related databases. In France, dry storage of spent fuel is still in a political decision process.

Germany's revised energy act, dated 2002, stipulates prohibition of reprocessing and restriction of nuclear waste disposal to a final repository. To avoid large-scale transportation, nuclear plant operators shall construct on-site storage facilities for dry cask storage, to keep spent fuel assemblies until a final repository is available. At present, all construction applications are granted and most of the storage buildings have been erected. However, the process of public involvement has not been completed yet.

The U.S. still uses a "once-through" nuclear fuel cycle. Spent fuel is stored at the reactor site. A license application for the national spent fuel repository at Yucca Mountain is still in preparation. There is active research on reprocessing, advanced reactors, and advanced fuel cycles to reduce the amount and radiotoxicity of wastes. A centralized private spent fuel storage facility has been licensed but is not yet under construction. Because of these constraints and the limitations of fuel pool capacity, the use of at-reactor dry fuel storage is increasing.

The safety functions of a dry cask storage system, such as subcriticality during and retrievability after storage, impose requirements to avoid degradation of the fuel assembly. The safety functions are achieved by maintaining the integrity of the fuel assembly structure and preventing systematic cladding failures. These are basic and common requirements from which country-specific rules were derived.

In Germany, criteria for the dry storage of fuel assemblies were promulgated by the Reactor Safety Commission in 2002. It is stated that mechanical integrity of the fuel assembly structure must be maintained during handling, storage, transport for final storage, and discharge operations, systematic cladding failures have to be avoided by limiting stress and strain of the material under consideration. Defective fuel rods need special treatment and/or confinement.

The present engineering approach to ensure cladding integrity is to impose limits of 1% plastic strain and 120 MPa tangential (hoop) stress. These values limit thermal creep degradation and hydride reorientation under dry storage conditions. They are valid for all commercial Zr-base materials up to burnups of about 70 MWd/kgHM and corresponding hydrogen concentrations of several hundred ppm.

For the handling of an individual damaged fuel rod, a special capsule has been developed. The capsule can be drained and stored in a canister with the dimensions and handling properties of a fuel assembly. The canisters can be stored in the spent fuel pool and in dry casks. Additionally, a wide spectrum of reconstitution techniques is available for structural and fuel rod damage.

In the U.S., guidance from the Nuclear Regulatory Commission (NRC) has been motivated primarily by regulations designed to promote operational safety when the fuel is removed from storage. NRC staff guidance specifies that the cladding of spent fuel be kept at 400°C or lower. Nowadays, because U.S. utilities are achieving even higher burnups and employing advanced cladding alloys such as M5®, the U.S. NRC is seeking additional information, for a wider range of alloys and burnups, on 1) mechanical properties of irradiated cladding and structural components, 2) hydride reorientation, 3) damage to fuel assemblies, and 4) fuel pellet oxidation. These data are needed for both storage and transportation issues.

The trend to higher discharge burnups has required extension of the database for the cladding integrity assessment. In the 1980s, rod internal pressures and cladding hydrogen levels were lower than those of more recent fuel, and cladding behavior could be assessed by burst tests. Higher hydrogen levels, which revealed significant degradation under burst conditions, led to more realistic creep tests at lower stress levels and storage temperatures of 300-400°C. The most realistic way to assess the long-term behavior is to perform creep tests at stresses and temperatures that reflect the cask storage conditions more closely. To this end, short gas filled rods may be used to simulate the cladding behavior in the central region of the storage cask. AREVA NP is performing such tests on its commercial PWR and BWR cladding materials such as Extra low tin (ELS) Duplex/Zry-4, M5® and Zry-2/LTP. These tests are geared to better understand the degradation limits and to make provisions for more demanding dry storage conditions. It is important to note that these materials will have a wide range of hydrogen concentrations. Whereas the Duplex ELS-type may show hydrogen concentrations of several hundreds of ppm, M5® will remain below 100 ppm up to the highest burnups achievable. This means that, at the beginning of dry storage, where temperatures of up to about 400°C can be reached, hydrogen is dissolved and will only be precipitated at lower temperatures and, hence, lower stresses. As a result, hydrogen reorientation will largely be avoided.

Safety Analysis of C30 Cask for Advanced Fuel

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At Paks NPP, Hungary, the C30 cask is used for the transport of the spent fuel assemblies from the at-reactor wet service pool to the interim storage facility. Occasionally, it is used for transport of assemblies between units. The C30 cask originally was licensed for fuel with no enrichment zoning, having nominal enrichment 3.6 %, and for the following loading parameters:

Max. number of assemblies	30
Total heat production	<15 kW
Average burnup of the cask load	<33 GWd/tU
Burnup of a single assembly	<40 GWd/tU
Initial enrichment	<3.7 %
Cooling time	>2.5 years

Due to the introduction of new, advanced fuel assemblies, increasing reactor power, increasing fuel burnup and the need to lower cooling time, the NPP plans to apply for a licence modification to allow the extension of the parameter range described above. The desired range of the parameters includes 50 GWd/tU burnup of a single assembly and 0.5 years of cooling time for the fuel. The change in fuel design for the different fuel types includes enrichment zoning, use of gadolinium, increase of average enrichment and increase of lattice pitch of the fuel pins.

To get a new licence for such essential changes makes it necessary to perform completely new safety analyses for subcriticality and radiation shielding. These analyses have already been partly done and partly will be completed in the near future. Both types of analyses are based on the results of thermohydraulic calculations performed previously [1].

The subcriticality requirements are met by the C30 cask loaded with the new fuel assemblies for normal and accidental conditions based on the fresh fuel assumption. The results of the thermohydraulic analysis were strongly utilized in this analysis.

The influence of the extended parameters on the neutron and photon dose rates is examined for normal and accidental conditions using the constraints from the thermal physical analysis. The maximum acceptable number of loaded assemblies with extended parameters is investigated. Loading strategies are suggested for different values of burnup and cooling time to meet the radiation protection safety requirements.

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Potential Benefits of Burnup Credit in Hungary

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According the present Hungarian practice, the criticality safety analysis of a transport/storage device is based on the fresh fuel assumption, and the subcriticality limit should be ensured by a conservative safety margin, covering all kind of uncertainties. However, it was realized, that using the planned new, advanced fuel types, the presently used transport/storage facilities can be used only with reduced capacity in some cases, if burnup credit is not used in subcriticality analysis.

Two sample cases, where application of burnup credit could avoid the capacity decrease are the compact storage pool at Paks NPP and the TK-6 transport cask. If the new, Russian designed fuel assemblies with enrichment zoning and increased lattice pitch (average enrichment 3.82 %, p=1.23 cm) are loaded into the compact storage pool, an absorber assembly should be placed at every 12th position to ensure the required level of subcriticality. The maximal capacity of TK-6 is 30 fuel assemblies, however using the fresh fuel assumption, in the case of optimal moderation the criticality safety criteria can be met if only 24 fuel assemblies are loaded into the cask, or 27 assemblies + 3 absorber assemblies are loaded.

A crude estimation suggests that these constraints can be removed taking into account the real uranium and plutonium content corresponding to the relatively low burnup of 10 and 25 MWd/kgU. These figures illustrate that a significant benefit can be taken from the use of burnup credit.

For accurate determination of the burnup required to meet the subcriticality criteria including the uncertainties in these facilities, a series of investigations should be performed. The most essential points should be clarified for a safe use of these benefits of burnup credit. The largest sources of the uncertainties are associated with errors due to the nuclear data, the errors in isotopic composition calculations and the uncertainty due to the influence of the axial burnup distribution (end effect).

The influence of the end effect on the compact storage pool was studied using approximately 200 axial burnup profiles. These profiles were taken from fuel cycle calculations performed for "real life" operational histories by the KARATE core design code. The maximal value of the end effect was about 3 % if actinides + fission products were used and about 1 % if only actinides were taken into account.

For the estimation of the uncertainties arising from the error of the nuclear data, about 130 MOX experiment were selected and analysed from the ICSBEP handbook. At a first glance, the calculations suggest, that the uncertainties from this source are about about 1-1.5 %. However, the representativity of these experiments for burnup credit applications is questionable. So this topic needs further examinations.

The error due to the composition calculations can be determined by a comparison of calculations with post-irradiation experimental data. For VVER type of fuel, only a few of such measurements are accessible with well documented irradiation history, and these data are available publicly only since the very last year. (They were performed in Dimitrovgrad.) Their comparison with calculations and the analysis of the results are planned in the near future.

Licensing Framework of Storage for Spent Fuel Installation in Indonesia

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Indonesia has operated three research reactors i.e. TRIGA 2000 reactor at Bandung, Kartini reactor at Yogyakarta and Multi Purpose Reactor-GA Siwabesy (MPR-GAS) at Serpong. Besides Indonesia has also operated some nuclear installations such as radioisotope production installation, radio-metallurgy installation, and transfer channel-interim storage for spent fuel installation (TC-ISFSFI). While being operated for research activities, reactors produce spent fuel that has the potential for radiation hazard.

The utilities shall collect spent fuel temporarily throughout the life time of reactor. To control the spent fuel storage according to requirements (Act No. 10 year 1997 article 24.2), the National Nuclear Energy Agency (BATAN) has constructed the TC-ISFSFI, managed by the Center for Development of Research Reactor Technology (CDRRT). TC-ISFSFI is intended to collect spent fuel and other irradiated materials produced by MPR-GAS, the radioisotope production installation, the radio-metallurgy installation and other institutions.

The control of TC-ISFSFI is conducted by the regulatory body (BAPETEN). It is implemented by establishing regulations, carrying out licensing and performing inspections. The TC-ISFSFI shall be subjected to licensing. According to the Act (No. 10 year 1997 article 17), the construction and operation of nuclear reactors and other nuclear installations as well as the decommissioning of nuclear reactors shall be subjected to the licensing. Implementation of the Act, especially the licensing regulations, includes the Chairman Decree of Bapeten (CD) No. 06 year 1999 on the Licensing Construction and Operation of Nuclear Reactor.

In performing a review of the licensing requirements, BAPETEN uses government regulations and the other regulations on nuclear and radiation safety such as:

- Government Regulation (GR) No. 64 year 2000 on Licensing of Nuclear Energy Utilization,
- Government Regulation No. 27 year 2002 on Radioactive Waste Management,
- CD No. 01/1999 on radiation safety,
- CD No. 06 year 1999 on the Licensing Construction and Operation of Nuclear Reactor,

- GR 63 year 2000 on Safety and Health against the utilization of Ionizing Radiation,
- GR No. 27/2002 on radioactive waste management.

CDRRT-BATAN has applied for a licence by submitting licensing documents to BAPETEN (at that time the Atomic Energy Control Bureau –BATAN). BAPETEN carried out the safety evaluations based on the regulation above. The licensing documents submitted include: safety analysis report; the quality assurance programme, related procedures (i.e. environmental monitoring, radioactive monitoring procedures, etc.), the emergency preparedness programme, Design Information Questionnaire (DIQ) and physical protection programme.

Bapeten has carried out the safety evaluation based on the available guides and regulations. BAPETEN has gained considerable experience from previous related activities. BAPETEN has evaluated the Safety Analysis Report proposed by CDRRT and carried out inspections to check the safety of the TC-ISFSFI activities.

This paper addresses present regulations related to spent fuel storage, including licensing system regulations.

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Fast Evaluation of 235U/238U Ratio in Nuclear Spent Fuel Safeguarded by Thermal Ionization Mass Spectrometry by Using Refractory Metal Oxides

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Thermal ionization mass spectrometry (TIMS) has long been the pre-eminent analytical method for measuring uranium isotopic composition in nuclear spent fuel safeguard. Modern mass spectrometers have improved with very stable electronics, vacuum systems, and software, such a way that other improvement were restricted to the sample application chemistry. In most of the cases, uranium samples were applied onto rhenium filaments as a neat uranyl nitrate form, after prolonged purification procedure. Solvent extraction, ion exchange resin, and electrodeposition are among the most common purification procedures adopted for nuclear materials safeguards. Impurities can contribute a lot of instability to the ion beam arising from the physical or chemical nature of the deposited sample, hence it is a pure surface phenomenon. Their presence as metal oxides form will alter the nature and composition of the loaded sample layer, remembering the fact that most of the fission products will form oxides of lower melting point than that of the parent uranium oxide. It will lead to a nonhomogenous evaporation of the metal ions from the filament surface, and as a consequence, to nonhomogenous ion production [1, 2].

Some studies showed that the effect of metal ions, halides, and other fission products revealed no considerable effect on the precision and accuracy. Others claimed that the use of borax, tantalum oxide (Ta_2O_5) activator improve single filament techniques, samples loaded from phosphoric acid with small quantity of suspended graphite were claimed to be useful in improving the measurement. It is not possible to obtain nuclear spent fuel without the presence of considerable amount of impurities. Although many articles in the literature conclude that small amount of impurities contribute no significant effects on the precision and the accuracy of the mass spectrometric measurements of uranium isotopic ratio in nuclear spent fuel safeguards, it requires much longer time to achieve pilot signal stability. A stable pilot signal is essential to get internal and external precise and accurate data acquisition [3-5].

The addition of refractory metal oxides forming solutions in larger proportions than that of uranium samples will dominate the evaporation phenomena without influencing the measurements of masses. A suitable refractory metal oxides forming solutions, such as that of chromium (M. pt 2265°C), cerium (M. pt 2600°C), magnesium (M. pt 2800°C), and thorium (M. pt 3300°C), was investigated.

It was found that the time to reach steady pilot signal for uranium ions is less than ten minutes. The internal and external precision of the measurements were of the same magnitude to that of the neat uranium solution and sometimes it was better. At the same time, the accuracy of the analysis is similar to that of the neat uranium solution.

Investigation of the effect of such metal forming solutions on improving the $^{235}U/^{238}U$ ratio analysis without prior purification, was carried out. From these results, it is was found that the addition of refractory magnesium oxide forming solution to simulated sample solutions of nuclear spent fuel will improve the both the accuracy and precision, as well as shorten the time required to obtain stable pilot signal for uranium. It was found that the time required to reach stable pilot signal was shorter (proportions from 2:1 to 1:4). In case of high level impurities in the simulated process solution, it was found that it is not practical to analyze it without prior purification by solvent extraction. The present technique represents a good alternative, and it improves the quality of the results as well, and will probably reduce the mass fractionation effects. From these results, it was concluded that the addition of refractory metal oxides forming solutions could be advantageous to the analysis of uranium isotopic ratio, particularly in the case of magnesium, in getting better accuracy, precision, and shorter time. The accuracy and precision of varying proportions of metal oxide forming solutions to that of uranium was studied.

As far as we are aware, there is no work performed in the past, in which refractory oxides is added to improve the uranium isotope ratio determination by thermal ionization mass spectrometry for nuclear spent fuel safeguards.

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Demonstration Tests Program Using Full-Scale Metal Cask and Concrete Cask

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To contribute to safety standards for concrete casks, CRIEPI successfully completed a four-year study program in March 2004 of demonstration testing for interim storage of spent fuel, mainly related to concrete cask storage technology. The results obtained in demonstration tests were reflected in Japanese safety guidelines for concrete casks issued by NISA/METI (Nuclear and Industrial Safety Agency, Ministry of Economy and Trade Industry) in June 2004. Moreover, a new research program for the verification testing of cask integrity under long-term dry storage conditions has been started. The schedule for these programs is shown in Table.1. This paper summarizes these research programs.

Concrete cask performance test

Two types of full-scale concrete casks, reinforced-concrete type and concrete filled steel type, and two types of multi-purpose canisters (MPC) to store 21 high burn-up spent PWR fuel assemblies were designed and fabricated by Japanese manufacturers for the demonstration tests. High corrosion-resistant material is used for each MPC's body material. The drop test yard and thermal test facility were installed in the Akagi test center of CRIEPI, located about 130km north of the center of Tokyo.

To verify the thermal characteristics of the concrete cask, the heat removal tests under normal and accidental conditions with two types of concrete cask were executed in vertical and horizontal orientations. In the normal condition, heat power was controlled from 10kw to 22.6kw. In the accidental condition, 50% and 100% of the inlet ducts were blocked. Temperature distribution of the container and MPCs, flow velocity at inlet and outlet ducts, heat balance, and crack width on the cask surface were measured during tests [1]. To verify the applicability of numerical methods, the thermo-hydraulic numerical analyses with the PHEONICS code were also executed.

To verify the structural integrity of MPCs during handling in the storage facility, two drop tests were executed (1m height in the horizontal orientation and 6m height in the vertical orientation) onto the unyielding target with the two full-scale MPCs. Acceleration and strain at various points were measured. Helium leak tests were also performed before and after the drop test to confirm the integrity of leak-tightness of the MPC against the impact. The welded lid structure was cut and investigated for welding quality with a microscope after the drop test [2].

Moreover, to clarify the tipping-over characteristics of concrete cask in the freestanding condition, seismic tests with the full-scale concrete cask have been executed using the tri-axial table test facility, clarifying the seismic response of the components.

Containment Performance Test of Metal Cask

A new research program of the verification test of cask integrity under long-term dry storage conditions was started in April 2004. Some important issues related to metal cask are as follows.

a. Drop test without impact limiters under accident conditions

In an interim storage facility of spent fuel, metal casks will be handled without impact limiters. Although there have been a lot of tests and analyses reported for evaluation of drop tests of metal casks, no quantitative measurement has ever been made for any instantaneous leakage through metal gaskets during the drop tests. In this study, leak tests were performed using a full scale cask without impact limiters simulating a drop or a tipping-over accident onto a concrete floor in a storage facility. Instantaneous leak rate was quantitatively measured at the drop tests. The amount of leakage was insignificant [3].

b. Metal cask containment performance during long-term storage

The confinement structure of the metal cask is designed to have a highly reliable multi-barrier system using metallic gaskets instead of the conventional rubber gaskets. Therefore, it is very important to clarify the influence of the stress relaxation of the gaskets on the containment performance of the metallic gaskets for long-term usage. Long-term containment of metal gaskets in the cask lid structure of full-scale models has been measured for more than 14 years under a constant temperature of 160°C. The results indicate that containment will be maintained for more than approximately 50 years, taking account of decay heat of spent nuclear fuel.

* These activities have been being executed under contract with NISA/METI.

Table.1 Schedule of demonstration programs for storage cask

Program Item		2000	2001	2002	2003	2004	2005-2008
Concrete Cask Performance Test		*Phase1*					*Phase2*
a. Basic Design							
b. Fabrication of full-scale concrete cask							
c .Demonstration tests	*Heat removal*						
	MPC Drop						
	Seismic						
e. Safety Analysis							
Containment Performance Test of Metal Cask		*Phase1*					*Phase2*
a. Drop Test Without Impact Limiter							
b. Long-term Sealability Test of Lid Structure							

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High Burnup Fuel Cladding Tube Property Test for Evaluation of Spent Fuel Integrity in Interim Dry Storage

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The Japan Nuclear Energy Safety Organization (JNES) has been planning and performing tests in order to evaluate fuel integrity in interim dry storage. By 2003, thermal creep testing and irradiation hardening recovery testing had been performed using irradiated cladding tubes and the creep equation for boiling water reactor (BWR 50GWd/t type) and pressurized water reactor (PWR 48GWd/t type) fuel cladding had been established [1, 2]. Since 2004, JNES has been performing the tests using irradiated cladding tubes, in which the effects of hydride reorientation are investigated for up to 55GWd/t type BWR and PWR fuel cladding. This paper reports on the whole JNES test plan as well as results of hydride effects evaluation testing obtained from 2004-2005.

JNES tests were composed of hydride effect evaluation testing, irradiation hardening recovery testing, and creep testing. In each test, both BWR and PWR irradiated cladding tubes are used as test material. In hydride effect evaluation testing, hydride reorientation testing has been performed in order to evaluate the correlation between hydride reorientation behavior and conditions such as hoop stress, temperature, and cooling rate for BWR and PWR 55 GWd/t type cladding. Mechanical property testing is also performed as a part of hydride effect evaluation testing, in which mechanical properties of cladding with radial hydride after hydride reorientation testing are evaluated. Irradiation hardening recovery testing is performed in order to evaluate the irradiation hardening recovery behaviour during interim dry storage. As for creep behaviour, JNES had already established the creep equation based mainly on secondary creep rate evaluation for 50GWd/t type BWR and 48GWd/t type PWR fuel cladding tube until 2003 [2]. The creep rupture behaviour had been evaluated for the same type of cladding tubes. The creep property of both BWR and PWR 55GWd/t type fuel cladding tubes will be evaluated by 2006. The test schedule is shown in table 1.

Up to 2005, JNES hydride effect evaluation testing has been performed using both 5-cycle irradiated Zry-2 cladding for 50GWd/t type BWR fuel and 3-cycle irradiated Zry-4 cladding for 48GWd/t type PWR fuel. In hydride reorientation testing, hoop stress in cladding tube specimens was loaded by inner pressure. The specimen temperature was held for 30-60 minutes at the test temperature in the furnace to dissolve the hydride in the cladding, then the specimen temperature was decreased to around room temperature to precipitate the hydride. The degree of hydride reorientation was evaluated from metallography of the specimen after testing. In the test for BWR fuel cladding, hoop stress conditions were from 0 to 100MPa and the temperature conditions were from 250 to 400 deg.C. In the test for PWR fuel cladding, stress conditions from 0 to 130MPa and the temperature conditions were from 250 to 340 deg.C. The effect of cooling rate was also tested. The test matrix for 3-cycle irradiated Zry-4 cladding is shown in table 2 as an example. In addition, the mechanical property test, which is composed of a ring compression test and a longitudinal tensile test, was also performed after hydride reorientation testing in order to evaluate the circumferential and longitudinal mechanical property of hydride reorientation treated specimens. The crosshead speed in the ring compression test was about 2mm/min and the strain speed in the longitudinal tensile test was about 8.3x10^{-5} /s. In 2006, deformation speed dependency of mechanical properties for hydride reoriented material will be tested.

The results of hydride reorientation testing obtained during 2004-2005 clarified that the degree of hydride reorientation depends on the hoop stress and temperature. 3-cycle irradiated Zry-4 cladding for 48GWd/t type PWR fuel showed the increase of hydride reorientation at 300 deg.C under the hoop stress condition above 100MPa. The difference of hydride reorientation behaviour between Zry-2 for BWR and Zry-4 for PWR was indicated from the test results for each cladding material.

Table. 1. JNES Test Schedule

		2000	2001	2002	2003	2004	2005	2006
Survey and Planning								
Creep Test	>Creep test	PWR48GWd/t, BWR50GWd/t					PWR55GWd/t, BWR55GWd/t	
	>Creep rupture test	PWR48GWd/t, BWR50GWd/t						
Hydride Effects Evaluation Test	>Hydride reorientation test >Mechanical property test						PWR48GWd/t,55GWd/t BWR50GWd/t,55GWd/t	
Irradiation Hardening Recovery Test		PWR48GWd/t, BWR50GWd/t 330deg.C-420deg.C					<330deg.C	

Table. 2. Hydride Reorientation Test Matrix for 3cycle irradiated Zry-4 cladding

Temp. (deg.C)	Hydrogen Cont. (ppm)	Cooling Speed (deg.C/h)	Hoop Stress (MPa) 0	85	100	115	130
340	~140	30	-	-	A	-	A
300	~70-90		A	A	A	-	A
275	~50-70		A	-	A	A	A
250	~50		-	-	A	-	A
300	~180		-	-	-	A	-
300	~180-210	3	A	-	-	A	-
		0.6	-	-	-	A B	-

A: Hoop stress was kept constant.

B: Hoop stress decreased with temperature.

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Drop Accident Analyses of Dry Metal Cask without Impact Limiter and Evaluation of Leak Rate

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In Japan, utilities are preparing to initiate an independent interim storage facility (ISF) for spent fuel at Mutusi-city in Aomori prefecture in 2010. In the ISF, dual purpose metal casks which are used for both transportation and storage will be adopted, because no direct handling of spent fuel is necessary at the ISF, thereby reducing risks. The metal cask will be handled without impact limiters in the ISF. Therefore, supposing a hypothesis cask drop accident without the limiter, cask drop tests using an actual size simulated cask [1] were analyzed and the leak characteristics from the flange with the metal gasket were investigated. The tests were conducted without the limiter, and the conditions were a horizontal drop and rotational impact with the supporting point at a trunnion.

Before the calculation of this cask drop event, based on examination of results obtained from small scale tests for seal performance of flange with aged metal gasket [2], a correlation curve between total sliding movement of lid and leak rate was obtained as follows:

$Q=10^{(2.2X-9.5)}$ for $0.1 < X < 0.5$

$=10^{(0.45X-7.3)}$ for $1.5 < X$

Where

Q: leak rate (Pa$*$ m^3/s)

X: total sliding movement of Lid (mm)

The relation between the total sliding movement of the lid and the leak rate obtained from the cask dropping tests without the impact limiter was compared with the correlation. Considering the leak rate increase due to aging of the gasket which is assumed to be ranging from 100 to 1000, the result from the cask drop tests agreed to the correlation with a 95 % confidence level.

Then, a general non-linear dynamic simulation computer code, LS-DYNA was used in the calculation of the cask drop tests. In the calculation, a half of the cask, considering axial symmetry, and a concrete floor were modelled. The calculation for the horizontal drop test was initiated just before a trunnion impacts the floor. For the rotational impact test, the calculation was initiated just before the edge of the outer flange impacting the floor. The impacting velocity of the cask was calculated assuming a free drop from the original position for both horizontal drop and rotational impact. Although the drop height was 1 m in both horizontal drop and rotational impact tests, the drop height was changed to 1.5 m and 2.0 m as a parameter in the calculation.

The increase of the leak rate is not only due to the increase of the total sliding movement of the lid but also suspected to be caused by plastic deformation of flange or bolts. The applicability of the correlation curve between the total sliding movement of the lid and leak rate postulates that structural parts composing the leak tightness of the lid maintain the integrity without plastic deformation even after the event. Using this correlation and the total sliding movement of the lid, the leak rate could be evaluated. In the evaluation, the difference in hoop diameter, and effects of aging and elapsed time between the event and the measurement should be corrected. Figure 1 shows data obtained from the small scale tests for seal performance of lid with aged metal gasket, the correlation curve with 95 % confidence level and the calculated total sliding movement. For the simulated cask used for the test, the clearance between the lid and the cask body is small and the total sliding movement is limited. Therefore, increase of the leak rate due to large total sliding movement is difficult to occur. In the rotational impact event, the top of the cask near the seal collides against the floor. In the calculation for the 2 m rotational impact case, plastic deformation was observed around the top of the cask but not near the flange. The present calculations were done under room temperature conditions. The leak rate estimation methodology is applicable to the actual cask drop accident, if plastic deformation does not occur near the flange.

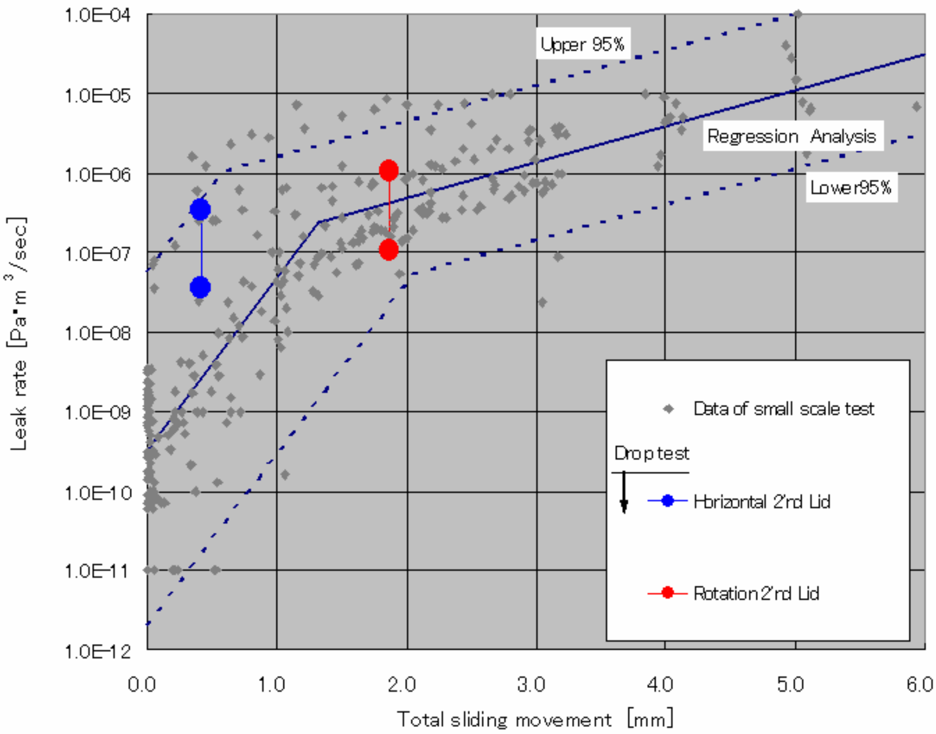

FIG. 1. Correlation between leak rate and total sliding movement of lid

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Long-Term Management of Spent Nuclear Fuel in Korea: A Preliminary Evaluation of the Influence of High Burnup Fuel

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A technology related to a geological disposal as a long-term spent nuclear fuel management strategy has been under development at the Korea Atomic Energy Research Institute since 1997. In an attempt to pace with the worldwide trend, Korea has also pursued a longer cycle operation of nuclear reactors and the corresponding higher discharge burnup of spent fuel. If the discharge burnup of the spent fuel increases, however, the associate increase in radion levels, decay heat, and radionuclide inventory could affect: (i) the thermal behaviour in relation to the temperature in the buffer and host rock surrounding the disposal canister; (ii) the dose rate in the radiation shielding; (iii) the assurance of sub-criticality from the nuclear safety aspects. In this study, these impacts on the safety in geological disposal and the long-term management of spent fuel will be investigated.

First, the inventories, initial ^{235}U enrichment, discharge burnup, and specifications such as the dimensions, fuel rod array, and the weight of the spent fuel generated from the existing and planed nuclear power plants were investigated and projected to support the analysis of the impact of a high burnup fuel on the safety level in a geological disposal facility and the long-term management of spent fuel. As a result, the historical and projected inventory by the end of 2057 is expected to be 20,500 MTU and 14,800 MTU for PWR and CANDU spent fuel, respectively. The quantity of spent fuel with an initial ^{235}U enrichment of 4.5 wt.% and below was shown to be 96.5% in total. Average discharge burnup was revealed to be ~36 GWD/MTU and ~40 GWD/MTU for the period of 1994-1999 and 2000-2003, respectively. It is expected that the average burnup will be ~45 GWD/MTU at the end of 2000's. From the comprehensive study, it was concluded that the imaginary spent fuel with a 16×16 Korean Standard Fuel Assembly, a cross section of 21.4 cm×21.4 cm, a length of 453 cm, a mass of 672 kg, an initial ^{235}U enrichment of 4.5 wt.% and a discharge burnup of 55 GWD/MTU could cover almost all of the higher burnup fuels to be produced by 2057.

For the high burnup fuel described above, and the low burnup fuel selected previously, the source terms such as the decay heat, radiation intensity and spectrum, the individual radionuclide activity and relevant toxicity were calculated by using ORIGEN-ARP to obtain the quantified difference and subsequent effect from a long term safety point of view for the management of spent fuel. The low burnup fuel considered is a 17×17 Korean Optimized Fuel Assembly with an initial enrichment of 4.0 wt.% ^{235}U and a discharge burnup of 45 GWD/MTU. In the decay heat calculation of this study, the difference for each spent fuel element was quantified for the fission product and the actinide group for a considerably long time span. The predominant radionuclides causing the decay heat to increase was listed. The thermal impact caused by a higher thermal load was calculated and analyzed. Especially, a thermal analysis of a temperature profile in the bentonite buffer and the host rock was implemented by using the FLAD3D code. For the shielding calculation, a comparison was made for the weight to satisfy the design limit, to prevent a radiolysis and a subsequent corrosion at the surface of the canister accommodating the high burnup fuel versus the low burnup fuel. For the toxicity, the dominant radionuclides impacting on the long-term safety aspect were deduced for the fission product, activation product and actinide nuclides. For the criticality aspects, the initial ^{235}U enrichment is closely correlated with the discharge burnup, which means that a higher burnup needs a higher fissile enrichment. Therefore, the influence of a high burnup fuel with a high initial fissile enrichment on criticality safety is also an important concern in the long-term spent fuel management field. For the criticality impact caused by a higher initial fissile enrichment and discharge burnup, the SAS2 module in SCALE5 was applied to obtain the radionuclide concentration as a function of the enrichment and the pertinent burnup. By assuming the various low, medium, and high burnup fuels, this effect was investigated in detail. Both the case in which only an actinide burnup credit is considered and the case in which an actinide and fission product burnup credit is considered were analyzed further. The SCALE5 code package was used for all the criticality and shielding calculations.

Through the study previously listed, a long-term management strategy for a high burnup fuel was proposed. Additionally, the important impacts from a long-term safety point of view as the discharge burnup increases were summarized.

A Study on the Management of Spent Fuel Storage Capacity in South Korea

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The saturation of South Korea's at-reactor (AR) spent fuel storage pools will create a necessity for additional spent fuel storage capacity. Because the South Korean government has the plan to increase the number of nuclear power plants from 20 units (end of 2005) to 27 units by 2015, the increase of spent nuclear fuel generation will be accelerated. Because there is no clear national plan for spent nuclear fuel storage and disposal, the utility company (Korea Hydraulic Nuclear Power company) is planning to construct a spent fuel storage facility with 11 000 tHM capacity for Pressurised Water Reactor (PWR). This study is intended to predict the maximum allowable periods when the storage facility will be fully occupied with respect to the already fixed re-racking plan of spent fuel storage pools and construction of new nuclear power plants. The result from this study will be used as fundamental data for an efficient management of the spent fuel storage capacity up to the time of the availability of a future disposal facility.

To perform this, the amount of spent fuel generation was estimated through the nuclear power supply plan until the year 2015. After 2015, the amount of spent fuel generation was estimated by assuming three scenarios that predict the number of future nuclear power plants: the "discontinue scenario," the "maintenance scenario," and the "upgrade scenario." All of the three scenarios assume that spent fuel transfer between sites is not allowed, but that transfer between NPPs at the same site is allowed. Currently, average thermal efficiencies and capacity factors for South Korean commercial PWRs are around 35 % and 90%, repectively. These values are assumed to be sustained to estimate the spent fuel generation for this study. The "Discontinue scenario" is based upon the assumption that no new nuclear power plants will be built after 2015. The "Maintenance scenario" assumes the maintenance of the total generation capacity in 2015 by replacing permant shutdown plants with new ones after 2015, while the "Upgrade scenario" is based upon the assumption that the sustained development of nuclear power is continued with the nuclear power plants (NPPs) increase keeping up to the estimated amount of the total electricity consumption after 2015. In order to estimate the total electricity generation after 2015, this study uses a logistic curve fitting method, to estimate the per capita eletricity generation after 2015. Then, the total electricity generation can be calculated by multiplying the per capita electricity generation with the estimated population. Projections of the share of nuclear power may be predicted by the MESSAGE-V method. Nuclear energy will contribute to the total Korean energy generation by continuously supplying more than 40% of the electricity for this scenario. For each scenerio, current and extended burnup was considered separately. Current burnup is based upon the continued current discharged burnup level: 44 000 MWd/tHM for the Kori site and 52 000 MWd/tHM for the other sites. Extended burnup is based upon the nuclear fuel supply plan in South Korea. The extended burnup case assumes that the average burnup of spent PWR fuel will be increased to 55 000 MWd/tHM after 2010. Finally it is calculated that the storage facility should be operational by 2016. As shown in figure 1, for three scenarios with the current burnup case, it is estimated that the spent fuel storage facilities will be saturated by 2041.

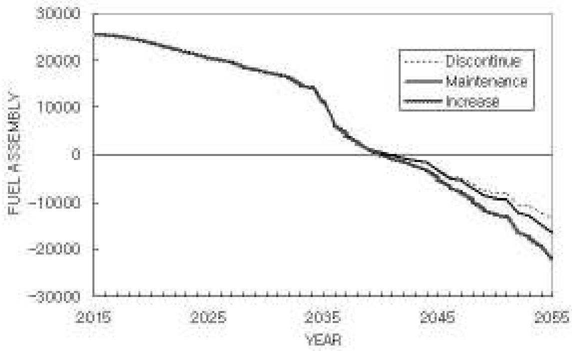

Figure 1.Storage facility: vacant capacity

For the "Discontinue scenario" and the "Maintenance scenario" with extended burnup, it is estimated that the spent fuel storage facilities will be saturated by 2049. For the "Upgrade scenario" with extended burnup, the saturation is expected by 2050. In the case that a disposal facility is not in operation in 2050 at the latest, it will be necessary to secure additional storage capacity. If a disposal facility is in operation at 2050 it is estimated for the current burnup case that additional capacities of about 5 100 tHM, 3 900 tHM, and 3 400 tHM for storing the spent nuclear fuel will be required for the "Upgrade", "Maintenance", and "Discontinue" scenario, respectively. In order to get a more meaningful result, it is recommended that a study on the practical storage facility capacity with respect to the risk accessment, public acceptance and further research should follow.

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Modelling of the RBMK-1500 Spent Nuclear Fuel Characteristics and Comparison with Available Experimental Data

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Estimation of the nuclide content and radiation characteristics of irradiated nuclear fuel is one of the most important questions related to SNF handling, storage and disposal. SNF characteristics for RBMK type reactors are almost absent in the scientific literature, while a great variety of publication with modelling and experimental results for BWR, PWR, WWER type reactors are available. The aim of this work was to create a calculation model for RBMK fuel using the SCALE computer code system and to compare calculation results with available experimental and numerical data. Such comparison can provide information about the suitability of computer codes for the estimation of the RBMK fuel characteristics.

A calculation model of the RBMK-1500 reactor channel containing fuel was developed using SAS2H sequences from the SCALE system. The basic assumptions for the modelling were the following: the RBMK-1500 fuel bundle (which consists of 18 fuel rods) inside the reactor channel was described as an element of 5 concentric cylinders; ^{235}U enrichments - 1.8%, 2.0%; burnup varies from 4 to 28 MWd/kgU; burnup shape of the fuel bundle was assumed as uniform.

Obtained calculation results are compared with available experimental and numerical data for RBMK fuel presented in referenced publications [1-5].

Experimental measurements of the activity ratios ((^{239}Pu+^{240}Pu)/^{238}Pu, ^{240}Pu/^{239}Pu, ^{134}Cs/^{137}Cs, and ^{144}Ce/^{137}Cs) of the samples polluted with Chernobyl NPP nuclear waste are presented in [1, 2, 3]. Computer modelling was used to obtain the burnup values that correspond to the experimentally measured activity ratios. Calculated burnup values were compared with values presented in the Chernobyl NPP loading cartogram.

Destructive analysis of the spent RBMK fuel is presented in [4]. Concentrations of the ^{238}Pu, ^{240}Pu, ^{242}Pu, ^{242}Cm, and ^{244}Cm in the RBMK fuel with different burnup were estimated using this method. Computer modelling was performed for the same fuel and for different burnup values. As an example, comparison of experimental and modelling results for ^{238}Pu and ^{242}Cm is presented in Fig. 1.

Calculations of the transuranium nuclides concentration for RBMK nuclear fuel with initial 1.8% ^{235}U enrichment using analytical equations are presented in [5]. For the same fuel, computer modelling was performed and obtained results were compared.

In the paper, detailed comparison of the available experimental and numerical data for RBMK fuel with numerically results obtained using SCALE system will be presented.

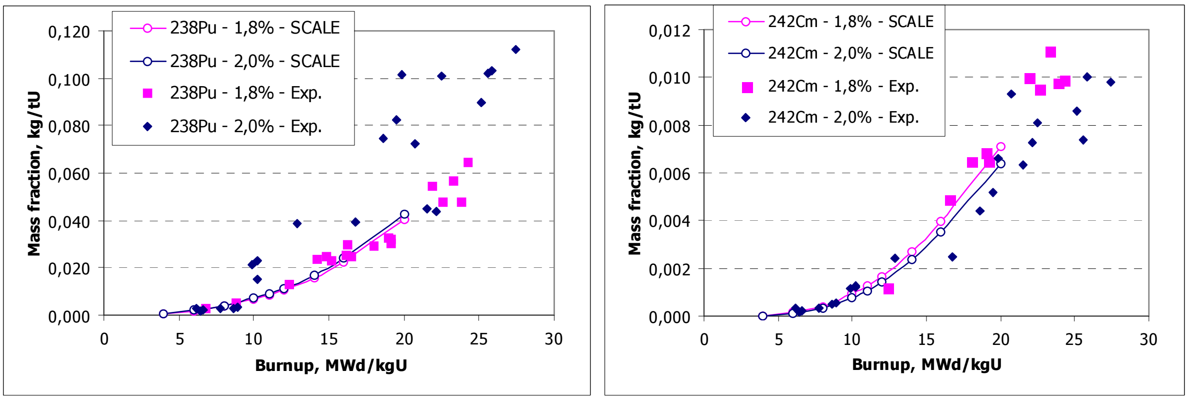

FIG. 1. Comparison of experimental and modelling results

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Feasibility of Plutonium Use in BWR Reactors, A way to Dispose of the Spent Fuel

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To assess the convenience of a closed fuel cycle, preliminary calculations have been done to evaluate which option will be the most attractive to follow from an economic point of view. Currently in Mexico, there is no defined policy for high level waste, so it is necessary to perform several studies to help define a possible strategy focused on the spent fuel. The calculations shown here indicate that from the economic point of view, recycling could be an expensive solution or at least more expensive than the once-through option.

1. Introduction

The BWR reactors of Laguna Verde Nuclear Power Plant have an electrical output of 654 MWe each, and the core contains 444 fuel assemblies. To reach the 18-month cycle currently established for operation, it is necessary to load around 112 fresh fuel assemblies (1/4 of the core, approximately) after each operation cycle, resulting in 112 spent fuel assemblies being discharged from the reactor.

The BWR fuel assembly (FA) contains approximately 180 Kg of heavy metal (uranium). After discharge and reprocessing, the amount recovered will be 94% uranium and 1% plutonium, which means 169.2 kg of uranium and 1.8 Kg of reactor grade plutonium.

If a once-through cycle is considered for both reactors, the amount of fuel assemblies through their entire life of operation will be 112 fuel assemblies/cycle multiplied by the number cycles minus one plus the initial load of the reactor. This produces 3244 assemblies for each reactor, resulting in a total of 6488 fuel assemblies or 1622 ton of high radioactive waste.

When recycling the spent fuel of both reactors, practically all the fuel discharged will be reprocessed except for the last four cycles (if the plant is planning to close and there is no license extension). This would result in 1448 UOX assemblies plus 612 MOX fuel assemblies as spent fuel from both reactors, or the equivalent to 515 ton of high radioactive waste. So, when using recycling, the amount of spent fuel is reduced to around 32% of the original amount produced without recycling.

Once the MOX fuel is loaded into one of the reactors (in fact the MOX can be loaded into the two reactors dividing the amount of MOX available), the amount of UOX fuel will be reduced, and less spent fuel will be available for reprocessing. Consequently, the plutonium production decreases and the amount obtained from the reprocessing will be enough for 27 MOX assemblies. Here, the load of MOX assemblies is reduced and the number of uranium assemblies loaded will be higher, increasing the uranium spent fuel discharged. Now the plutonium will be enough for the manufacturing of 28 MOX reaching the equilibrium point.

For the once-through cycle, we will have 6488 spent FA for direct disposal, while for the recycling option, we need to reprocess 4412 uranium spent fuel assemblies to get the necessary plutonium to manufacture 628 MOX fuel assemblies. However, as the core will be mixed, 5860 uranium fuel assemblies are necessary, resulting in 1448 UOX FA plus 628 MOX FA for disposal.

2. Economics

To assess the economics for each option, the parameters shown in the Table 1 were used. The costs for uranium and services correspond to the spot prices reported for UxC Consulting Company during the last week of September 2005 [1], and the enrichment and Burnup corresponds to technical data of LVNPP[2].

Table 1 Economic Parameters for Fuel

Enrichment	3.7%
Uranium cost	81. $/KgU
Conversion cost	11.5 $/KgU
Separative Work	114 /SWU
Kg HM/Fuel A.	180 Kg
Burnup	40 GWd/TU

The cost of one fuel assembly can be broken down as follows:

180 Kg U requires the next services (considering 0.3% of enrichment tails).

1782 Kg U3O8	122 431.2	
1511 Kg conversion	17 381.8	
856.7 Separative Work Units	97 655.5	
Fabrication	40 140.0	
Total	277 608.4	USD

This amount divided by the number of Kg/FA equals: 1542.27 USD/Kg Fuel

Direct disposal data:

To estimate the cost for direct disposal of spent fuel, the prices for Sweden reported in the OECD study [3] were taken as a basis to evaluate the costs in the calculations. So for transport and storage, a cost of 230 USD/Kg HM was assumed and 610 USD/Kg HM for encapsulation and final disposal.

Recycling option data:

The OECD study reported the PWR fuel cycle unit prices, using as a reference USD 1991. Those data assign a cost of 860 USD/Kg HM to the recycling option. Those prices include transport, reprocessing and waste disposal.

3. Results

The results obtained (after several calculations using electronic sheets developed for the fuel cycle costs, including the front and back end of the cycle) for once-through and partial-recycling options are shown in the Table 2. To calculate the reprocessing, disposition, and MOX manufacturing costs. The OECD study was used.

Table 2 Economic Evaluation of Fuel Cycle Option

		Once through		Recycling	
	Cost $/Kg HM	Number of FA	Cost $	Number of FA	Cost $
UOX F. A.	1542	6488	1800×10^6	5860	1626×10^6
MOX F. A	1600			628	181×10^6
Reprocessing	1140			4412	905.34×10^6
Fuel disposal	860	6488	1004×10^6	2076	321.37×10^6
Total Cost $			$\mathbf{2804 \times 10^6}$		$\mathbf{3033.7 \times 10^6}$

The methodology applied corresponds to constant money calculations, to make a direct comparison between once-through and recycling schemes.

However, the costs for final conditioning and disposition have uncertainties attached and the use of costs reported in the OECD study should be considered generic. Taking this into account, the costs for the recycling option can be higher [4].

4. Conclusions

The main result from this study, under the scenarios proposed, is the fact that even with the current higher costs of uranium, the recycling option is more expensive than the once through option. The results show that the recycling option is around 7.5% more expensive, and the reduction of high level waste will be approximately 68% which is significant. Another possible advantage will be that the spent fuel storage pools will be almost empty except at the end of life for the plant.

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Regulation of Spent Nuclear Fuel Management in Russia

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The legislative basis related to the handling of spent nuclear fuel during storage and reprocessing at the Fuel Cycle Facilities in Russian Federation is briefly discussed.

The Joint Convention on the Safety of Spent Nuclear Fuel Management and on the Safety of Radioactive waste Management was ratified by the Russian Federation on 04 November 2005.

During the last period of time the efforts on the development of the necessary legislative basis and the proposals on it's modernization related to storage and reprocessing of the spent nuclear fuel, including the spent fuel imported into the Russian Federation from abroad, were undertaken in Russia.

The main legislative acts regulating the handling of spent nuclear fuel in the Russian Federation are the following:

Federal law "On the Atomic Energy Using";

Federal law "On the Environment Protection";

Federal law "On the Radiation Protection of the Population";

Federal law "On the Special Ecological Programs for the Radiation Polluted Territories Rehabilitation".

The main Federal Norms and Rules related to the management of spent nuclear fuel at the nuclear fuel cycle facilities of Russian Federatian are following:

"Nuclear Fuel Cycle Facilities Safety. General Provisions" (NR-016-2000);

"Spent Nuclear Fuel Reprocessing Facilities. Safety Requirements" (NR-013-99);

"Spent Nuclear Fuel Dry Storage Facilities. Safety Requirements" (NR-035-02);

"Safety Requirements for Nuclear Fuel Storage and Transportation on the Atomic Energy Facilities" (PNAEG G-14-029-91).

The related industrial norms and rules on the nuclear and radiation safety also exist in the Russian Federation.

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During the last period of time the efforts on the development of the necessary legislative basis and proposals on the modernisation of the existing industrial basis related to storage and reprocessing of the spent nuclear fuel, including the spent fuel imported into the Russian Federation from abroad, were undertaken in Russia.

The Order of spent nuclear fuel import into the Russian Federation was established according to the main principles of the nuclear weapons nonproliferation, environmental protection, and the economical interests of the Russian Federation, and approved by the Decree of the Government of the Russian Federation named “On the Order for the Import of the Irradiated Nuclear Reactors Assemblies into the Russian Federation” dated 11 July 2003, No 418.

The import of irradiated assemblies into the Russian Federation is based on the annual limits approved by the Government of the Russian Federation taking into account the proposals prepared by the state body responsible for governing the use of atomic energy, and coordinated with the appropriate regulatory body in the field of the safety in the use of atomic energy, as well as with the local government authorities of the territories where the reprocessing and storage facilities are located.

In particular, according to the Order, the import of irradiated assemblies into the Russian Federation is possible only in the cases of the positive result of the state ecological review of the “joint project”, which should be prepared by the designated organizations and should be coordinated with the state body responsible for governing the use of atomic energy and with the appropriate regulatory body in the field of safety in the use of atomic energy. The designated organizations should have the necessary licenses to conduct such kind of activity.

The document “Statute on the development of the special ecological programs for the radiation polluted territories rehabilitation” was approved by the Government Decree dated 14 July 2000, No.421. These programs should be included into the ”joint project”.

The state ecological and other necessary state reviews of the appropriate project related to the spent nuclear fuel import into the Russian Federation from foreign countries for the purpose of storage and/or reprocessing should be implemented (conducted) according to the up-to-day legislative basis, and the common irradiation impact risk reduction. The enhancement of the ecological safety level as the result of the project realization (implementation) should be founded as well.

The main safety principles and regulatory requirements for storage and reprocessing of spent nuclear fuel at the fuel cycle facilities in Russian Federation are intended to be discribed, as well as the list of initial events for the design bases and beyond design bases accidents for the spent nuclear fuel storage facilities could be discussed also.

Regulatory Aspects of Construction of Dry Spent Nuclear Fuel Storage Facility in Russia

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Construction of the first spent fuel (SF) dry storage facility (XOT-2) in Russia began in 2004 on a site at "The Mining and Chemical Enterprise"(MCE). This storage facility is supposed to be used for temporary storage of SF from VVER-1000 and RBMK-1000 reactors [1].

The whole projected capacity of the storage facility is more than 33000 tons of uranium. Systems for SF storage have two physical barriers: hermetic capsules and hermetic storage "sockets". It is necessary to guarantee the following requirements for SF storage: storage duration is not less than 50 years, storage medium is nitrogen, storage temperature is from 300^0C to 350^0C, water content is less than 25 g/cm^3.

In accordance with regulations (Article 26 of the Federal Law "On the Use of Nuclear Energy"), no activity in the field of nuclear energy use is allowed without a licence for its execution. The Law defines several categories of facilities including the category "storage of SF"

The list of types of activities in the field of nuclear energy use that require licensing is established by the Government of the Russian Federation in the "Provisions on licensing of activity in the field of nuclear energy use". The Provisions include such types of activities as siting, construction, operation and decommissioning of SF storage. To obtain a licence, an applicant submits a set of documents to Rostechnadzor.

The most important part of the licensing procedure is expert review of the documents justifying the safety of the planned facility. The purpose of this review is to give an independent assessment about the safety of the type of activity proposed by the licence applicant, to assess proposed technical solutions related to safety assurance. The results of the assessment are then set forth in a technical document prepared by the Safety and Engineering Centre for nuclear and Radiation Safety (SEC NRC).

The review of the complete set of documents justifying the safety of XOT-2 construction was executed in three stages. It was done in this manner to give the applicant (MCE) time to prepare for construction of the XOT-2.

During the first stage of the review (January 2004), the principal questions that should be decided before the start of the XOT-2 construction were examined. Assessments were made for the following:

- conceptual provisions of safety,
- criteria and principles of safety for dry storage being used,
- the influence of external and internal effects of manmade and natural events on the reliability of the building structure,
- the building structure durability and reliability under the external and internal effects,
- quality assurance programs for construction of the XOT-2.

The main shortcomings and observations discovered by experts during the first stage of the review were addressed by the operating organisation according to terms coordinated with Rostekhnadzor before the licence for the construction of the dry SF storage was issued.

The projected decisions that influence safety of XOT-2 operation were examined during the second stage of the review (May 2004). It included assessments on:

- transportation technology operations with SF containers;
- systems for receiving SF;
- systems for preparing SF for storage;
- systems for protracted storage of SF;
- nuclear, radiation, technical and fire safety provisions;
- heat elimination system effectiveness;
- radioactive waste management systems;
- accounting for and control of nuclear materials and radioactive waste;
- physical protection of the facility;
- plans and procedures for decommissioning SF storage.

At the results of the reviews, about 140 observations and shortcomings were indicated that would be addressed by corrections in the project, technical documentation and the report on justification of safety.

The licence for construction of the dry SF storage was issued to the applicant in 2004 for the period of 10 years. The license made it a condition that the applicant is directed to remedy all observations and shortcomings that are contained in the reviews of the SEC NRS, in resolutions of the Glavgosexpertiza, in the State ecological review, and in the hygienic–epidemiological conclusion provided by FU “Medbioextrem” of the Ministry of Health of the Russian Federation.

At the end of 2004, the SEC NRS performed the third stage of the review, to assess actions planned by the applicant and design organizations to address all observations and shortcomings. The review of the complete set of documents justifying safety for XOT-2 construction was based on analyses of more than 70 documents of the Russian Federation and several IAEA documents.

A special regulatory document for SF storage safety related to design, siting, construction, operation and decommissioning is document НП-035-02 “Facility dry storage of spent fuel: Safety requirements”. Another special regulatory document is НП-016-2000 “General provisions of safety assurance facilities of nuclear energy use”. The latter document provides criteria, principles and general requirements for assurance of nuclear and radiation safety at facilities for nuclear energy use.

All reviews were made taking into consideration recommendations contained in the “Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management” [2].

At the present time and according to the licence terms, the applicant is working on: clarifications and elaboration of some project decisions (e.g. influence of an earthquake); control of hermetically-sealed physical barriers; justification of possible gas escape out of storage sockets; improvement of radioactive waste management and management (e.g. provisions for fuel spills).

Inspectors from the regional department of Rostekhnadzor in Zheleznogorsk supervise compliance with the licence terms, including construction work and amendments to address shortcomings.

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Waterproofing Shielding for Concrete Under Radiation Load

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An important aspect of safety during the transportation and storing of radioactive materials is to supply waterproofing of all concrete constructions having both direct contact with radiating substances and providing for strength, seismic shielding etc. This problem needs to be considered for all structures near water in the nuclear industry and concrete installations involved in the treatment and storing of radioactive materials. In this connection, the need to develop efficient techniques both for repair of existing structures and waterproofing of new structures for these applications is real.

Nowadays, various techniques of concrete waterproofing are widely applied in the world. However, under radiation conditions, many of these techniques can result in irreparable damage to the durability and reliability of a concrete construction rather than providing protection. For instance, damage can occur if waterproofing materials contain organic constituents, polymers etc. Applying new technology or materials to structural elements requires in-depth analysis and thorough testing. The price of an error may be too great.

A comparative analysis shows that one of the most promising types of waterproofing materials for radiation-loaded concrete constructions is “integral capillary systems”(ICS).

The tests on radiation, thermal and strength stability of ICS-treated concrete samples were completed at the Russian Federation Nuclear Center at Snezhinsk (RFNC-VNIITF). The main result is that applying ICS improves the waterproofing properties and protective strength properties of concrete in conditions of radiation. During the ministerial meeting for hydraulic structures safety (Moscow, Minatom, 17 June 2003), ICS was approved for use on nuclear industry structures. ICS has subsequently been successfully applied to many structures in Russia (cooling pools repair, treatment systems, different storage etc.).

This paper is devoted to describing the research strategy and tests, test results and planning of new tests and also to describing experience to date in applying ICS.

Topics on Gamma-Ray Control of the Compound Metal and Concrete Protection Quality

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A complex metal and concrete structure in the form of three layers of special steel with reinforced concrete filling between them has been developed as a containment wall for very durable casks for spent nuclear fuel shipment and storage. In compliance with the requirements of national standards and regulations applicable for nuclear power engineering as well as IAEA recommendations, these casks have to provide the rated protection against ionizing radiation and withstand emergency impacts while preserving integrity of tightness system and radiation protection.

The described metal and concrete casks have been designed and are intended to be used at durable spent fuel storage installations near reactors such as the Leningrad and Kursk RBMK reactors as well as other nuclear power plants (NPP) in the Russian Federation. The choice in favor of these casks as compared with the steel, cast iron and other domestic models was made in view of their lower price and frequently greater capacity. The first lot of the casks for NPP was produced by the works «Energotex» situated in the town of Kurchatov in immediate proximity to the Kursk NPP.

Production of a large lot of other standard-sized metal and concrete casks (86 casks already produced by April 2006) is expanding in view of mass utilization of submarines in the North and Pacific fleets. This increased production is also influenced by anticipated future fuel unloading of nuclear ice-breaker fleet reactors. These orders from the Murmansk Marine Enterprise and others are financially supported by the USA Department of Energy and the UK Department of Commerce and Industry.

The ability of these casks to meet radiation protection requirements is monitored using a specially developed gamma-control test bench following manufacture and dynamic strength assay. The bench is an electro-mechanical unit fitted with remote PC-based programmable control. It performs a synchronous scanning of the cask surface using a Co-60 radioactive source coupled with a detector.

In addition to the above discussion of metal and concrete casks and the test bench description, this presentation also addresses methods for evaluating these control results. Applicability and reliability of the chosen verification technique have been confirmed by selected data analyses for about a hundred casks of various types (NPP, submarines etc.).

Spent Fuel from NPP in Slovakia

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In Slovakia six VVER-440 reactor units are in operation. The oldest one (Unit 1, NPP Jaslovské Bohunice) has been in operation since 1978. The discharged spent fuel is stored for 3 or 4 years in the at-reactor pool (wet storage) and afterward is transported to the Interim Spent Fuel Storage (ISFS) facility in Jaslovké Bohunice.

Table.1 The capacity of the at-reactor pool

	Lower level		Upper level **	
	normal	defective	normal	defective
Bohunice 1	313	60	290	60 ***
Bohunice 2	314	60		
Bohunice 3	319	60	296	54 ***
Bohunice 4	318	60		
Mochovce 1	603 *	53	296	54 ***
Mochovce 2	603 *	52		

* compact racks

** upper level is for both reactors

*** positions for hermetic canisters only

The Interim Spent Fuel Storage capacity has increased from 5040 assemblies to 14 112 assemblies by using new baskets and a new arrangement. The ISFS is a water pool of Soviet design. The original basket (T-12) has a capacity of 30 assemblies and 56 baskets are located in the pool. The new compact basket KZ-48 has a capacity of 48 assemblies and 98 baskets are located in the pool. The process of reloading assemblies from T-12 to KZ-48 is in progress.

While the discharged fuel from the initial cycles was transported to the Soviet Union, subsequently discharged spent fuel remains in Slovakia.

Unit 1 in Jaslovske Bohunice will close at December 31st 2006. This paper presents an overview of discharged spent fuel from its operation (number of assemblies, burnup, nuclide composition, decay heat, activity).

Management of Spent Fuel in the Slovak Republic

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The skills in handling spent fuel have been collected in Slovakia for more than 30 years. During this time period a well established spent fuel management system was created.

The Slovak Government established the basic policy of spent fuel management in several resolutions. In 2000 the Slovak Government adopted the power policy of the Slovak Republic that is also related to the concept of fuel cycle back-end. In 2001, the Slovak Government in his Resolution No. 5/2001 accepted “The proposal on the schedule of economical and material solution of the spent fuel management and decommissioning process of nuclear facilities” and decided to submit the “Policy of decommissioning of nuclear facilities and management of spent fuel evaluated according to the act on environmental impact assessment” for a discussion on governmental level by the end of 2007.

The state supervision on nuclear safety of spent fuel management is performed by the UJD. The legislative framework in the Slovak Republic is based on acts and regulations. Acts are at the highest legislative level. Based on general requirements described in the acts, the regulations describe more detailed requirements. Several guides were issued by UJD. Unlike the acts and regulations, guidesare not binding for operators. Act No. 541/2004 Coll. on Peaceful Use of Nuclear Energy is the main legislative norm.

In Slovakia there are six nuclear power units in operation. These units generate about 500 spent fuel assemblies (approximately 60 ton of heavy metal) per year. Temporary storage of the spent fuel after its unloading from the reactor core is carried out at the at-reactor spent fuel storage pools. The spent fuel is stored in a rack and cooled by borinated water. After at least 2.5 years of storage in the at-reactor pools, the spent fuel is removed to the Interim Spent Fuel Storage Facility (ISFSF). ISFSF was commissioned in 1988. During 1997-2000, it was subject to a reconstruction and seismic upgrade. The original capacity was increased from 5040 to 14112 spent fuel assemblies. The spent fuel will be stored there for at least 50 years. A long-term goal within the concept of spent fuel management is the construction of a geological repository for the disposal of spent fuel in the Slovak Republic.

The export of spent fuel from Slovakia started in the seventies. Fuel assemblies were transported to the former USSR. After the commissioning of the ISFSF, spent fuel is transported from the at-reactor storage pools to the ISFSF for intermediate storage. In 2004 UJD approved the transport container C-30 for transport of forty-eight spent fuel assemblies. The transport capacity of the containers was increased, and so the number of transports could be reduced.

UJD steers various research tasks under the Research & Development programme. The Division of Nuclear Materials in cooperation with the Nuclear Power Plants Research Institute executes a task regarding the application of burnup credit in criticality calculations for WWER-440 spent fuel assemblies. The task will be perform from 2005 through 2007. In 2004 UJD started in co-operation with Department of Nuclear Chemistry of Komensky University in Bratislava a new scientific programme. The programme is targeted to the comparison of calculated isotopic compositions of WWER-440 spent fuel with isotopic compositions received by chemical analysis and measurements. The programme should be finished at the furthest in 2007.

In 2005 the operator of the ISFSF began to install an inspection stand. The stand is intended to be used for the dismantling of leaky assemblies. Besides, the stand will be used for various measurements of the spent fuel. The installation will be finished in spring 2006.

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The South African Experience in Spent Fuel Management at the Koeberg Nuclear Power Station - Perspectives from the National Nuclear Regulator (NNR)

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Units 1 and 2 at the Koeberg Nuclear Power Station (KNPS), near Cape Town in South Africa, were put into commercial service in 1984 and 1985 respectively. Initially it was intended that spent fuel assemblies would be stored in the spent fuel pools at the power station for a maximum period of four years whereafter they would be transferred to a dry storage facility which would be created at a safe site, unidentified at the time.

By 1986 an appropriate dry storage facility system had not yet been identified, and as the exhaustion of the installed capacity of the pools (285 cells per pool) was approaching, the initial racks were replaced in 1988 with high density racking with a capacity of 728 cells per pool.

In 1990 an order for four Castor type X/28 F licenced for dual transport/storage casks was placed by Eskom (the South African Electricity Utility and Operator of the KNPS), in anticipation that transportation from the Koeberg spent fuel pools, to the remote dry storage site would take place. The casks could also function as a contingency storage capacity for 112 spent fuel assemblies.

In 1995 Eskom began a feasibility study to establish an optimum storage facility. Ultimately two storage options were identified:

- Dry storage in casks; and
- High density storage in the spent fuel pool (spent fuel pool reracking) - Super high density racks were considered to accommodate the storage requirements at Koeberg for 40 years.

Eskom finally concluded that wet storage was the most viable option to consider.

However due to delays in the spent fuel pool reracking project Eskom had to develop a contingency plan, which involved the use of the dry storage casks referred to above. These casks could provide enough storage space to delay the need for high density storage racks by one fuel cycle on each unit. The safety case, presented by Eskom for the use of the casks as an interim spent fuel storage measure for three years, was approved by the NNR in February 2000 and inspections were carried out to verify compliance with conditions for the various NNR hold points.

Subsequently all four casks were loaded and are stored, in an appropriate building, on the Koeberg site.

In parallel Eskom applied to the NNR for permission to commence reracking of the spent fuel pools. This would accommodate all the spent fuel that will accrue during the remaining lifetime of the plant.

Following its in depth review of the safety case presented by Eskom, the NNR gave approval to Eskom to commence reracking operations of the spent fuel pool of the Koeberg reactors, which was satisfactorily completed on both reactor units.

The paper will provide an overview of the regulatory review process for both the dry storage casks and the re-racking of the spent fuel pools at the KNPS, focussing on major issues identified, and resolution thereof in terms of additional safety studies and modifications to the plant.

As an exmple the main aspects of the NNR review of the safety case of the re-racking of the spent fuel pools (submitted by Eskom) were as follows:

- Criticality safety;
- Spent Fuel Pool Cooling;
- Structural analysis;
- Quality control; and
- Installation phase.

The Role of Spent Fuel Storage in Multinational Approaches to the Backend of the Fuel Cycle

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In many of the countries with nuclear power, storage of spent fuel is not, or is no longer, a major problem. This positive situation often results ironically from the unsuccessful attempts of national waste management programmes to move ahead with disposal projects. Delays on the repository front have compelled some countries to increase their storage capacities, either by re-racking pools at reactors or by constructing new storage facilities (e.g. Germany, Switzerland, Hungary). In any case, many programmes planned for long periods of interim storage to allow the spent fuel to cool sufficiently, or simply to postpone the expensive task of implementing disposal. Examples of the former include Sweden, Finland, and Japan; the latter approach is illustrated by the Netherlands and Slovenia.

There are, however, some prominent exceptions; in those countries that urgently need expanded storage capacities, the reasons are usually political or societal rather than technical. The USA has manoeuvred itself into a corner by trying to implement an aggressive disposal strategy at Yucca Mountain, while centralised storage schemes have been blocked by law (at Yucca Mountain) or by opponents (in Utah). In Japan, there have been problems in gaining public acceptance at potential centralised storage sites. This problem is even greater in Taiwan.

Accordingly, support for multinational storage concepts has come in the past mainly from these countries with such problems. There have been USA projects (e.g. from the NPT) and Japanese proposals (e.g. from Suzuki) for international storage of spent fuel for decades in Russian facilities. The Russian authorities have also supported plans to launch commercial interim storage schemes. In the Russian case, one of the major drawbacks of the proposal from the point of view of potential customers is that the final disposition of the spent fuel (or the high level wastes that could result from reprocessing this fuel) is not clearly defined. If the fuel or the residues are to be returned to the customer after some time, then there is a much reduced incentive to use such a service, since the need for expensive deep disposal is, at best, postponed.

Multinational storage of spent fuel is therefore not in itself a key issue. Multinational disposal, on the other hand, is a topic which has become increasingly prominent over the past several years. Numerous countries with small nuclear programmes would welcome multinational disposal projects that allowed them to profit from the potentially large economies of scale in repository implementation. In a few countries, the possibility of hosting such a multinational repository has been discussed (most clearly in Russia but also in China, Australia and recently in the USA). The increased concern about international nuclear terrorism has led to greater readiness to consider such options. If, or when, multinational disposal becomes a practical option, then the optimization of interim storage strategies for the fuel to be disposed of becomes an important task.

There are several feasible options for storing the spent fuel that would go to a multinational repository: at the production reactors, at centralised national or multinational stores, or at the site of the final disposal. The choice is also dependent on other factors such as the siting of fuel encapsulation facilities and the number and geographical distribution of available repositories. A study on optimization would have to consider the entire spent fuel management system, from production, through storage and transport, to disposal. The system attributes to be considered are broad; they include nuclear and conventional safety, security aspects, economics and, of course, the crucial societal issues affecting all nuclear matters.

A very modest start at examining such issues has been made in the scope of the SAPIERR project that has run for the past 2 years with support from the EC FP6 research programme. In this project, organizations from 14 European countries with interests in the concept of shared repositories collaborated to define a possible inventory for such a facility. The options for implementation (not yet including identification of potential sites) were examined and first considerations of the influence on storage and transport were carried out. Currently, a proposal for a follow up project, SAPIERR-2 is being reviewed. This two year study could look in more detail at the influence of storage strategies on the disposal project and vice-versa, taking into account all of the attributes mentioned above. Even in a national waste management strategy, the storage and disposal requirements are intimately linked and intelligent system planning can ease technical and societal problems. In a multinational context, the pallet of options is much wider, but the challenges – especially in the political and societal areas – are much greater and systematic studies are correspondingly even more crucial.

Selection Criteria for Spent Fuel Storage Technologies

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Fissile fuel material contained in cladding or encapsulation material that has been irradiated in a power reactor is considered spent fuel. There are several types of spent fuel, including pressurized water reactor (PWR) fuel, boiling water reactor (BWR) fuel, mixed oxide (MOX) fuel, Canada Deuterium uranium (CANDU) fuel, other pressurized heavy water reactor (PHWR) fuels, high temperature reactor (HTR) fuel and advanced gas cooled reactor (AGR) fuel [1]. There are two principal alternatives for managing spent fuel:

- the direct disposal route: spent fuel, conditioned after a sufficient decay period, is directly disposed of without the separation of fissile components; and
- the reprocessing route: spent fuel is reprocessed, and high level waste (HLW) containing mostly fission products and a small proportion of the actinides is disposed of after proper conditioning [2].

The initial phase for spent nuclear fuel is storage under wet conditions in reactor pools after discharge from the reactor cores. This cooling phase is necessary for further handling of the spent fuel. After this initial phase sooner or later spent fuel needs to be transferred to another storage facility. This situation occurs for two reasons: first there is not enough capacity in the reactor pool; and second, the reactor has to be decommissioned. Because of these reasons, away from reactor storage (AFR) technology has to be taken into consideration at the beginning. This paper presents the main factors for selection of storage technologies. These factors are important for determination of basic requirements (storage space and storage time).

The main factors begin with the spent fuel characteristics. These are: fuel type and dimensions, enrichment ratio, rods, cooling time after discharge, cladding and other materials, fuel integrity and (short and long term) production amounts. Lifetime of the storage facility is another factor to be determined before the design stage. A reasonable lifetime is more than 100 years depending on requirements. Generally, storage periods of 30-80 years is selected for a new storage facility. Storage periods are chosen according to expected dates for a disposal facility or to postpone a decision. The storage facility lifetime also presents several advantages. It provides time to postpone decisions and for selection of disposal options. The main selection process is carried out between two storage types, wet and dry storage technologies. Dry storage technologies are preferred for cost saving due to modularity and less effort in operation and maintenance. Safety criteria influencing the selection of storage types include: long-term subcriticality, radiation doses, containment barriers, safe retrievability of fuel, risk analysis. A review of spent fuel storage facilities implemented during the last 10 years show that the storage in a dry environment is becoming more common [3]. In addition to the safety criteria, existence of national regulations and licensability are critical factors. Site selection has to be done at the beginning, and depends on not only technical factors but social factors as well. The main technical factors are: earthquake, climatology, hydrology, transport conditions and calculated radiation dose to the public. The time needed to implement a new storage facility covers several stages including design, licensing, construction and training of operators.

Public acceptance is another factor which may cause significant delays in the implementation process. Processing rates have to be determined according to the methods for loading and storing amount of spent fuel per year. Generally, dry storage facilities are independent from the reactor operations. But in the case of a failure, interface with reactor operations will be necessary. Multipurpose casks or vaults may solve these problems. All these factors affect the cost of the storage facility. The unit cost of an AFR storage facility may differ significantly among the various storage options. Costs can be presented by cost per storage unit or per fuel assembly in storage. It also depends on the required number of storage units. Main factors affecting selection criteria are presented in Figure 1.

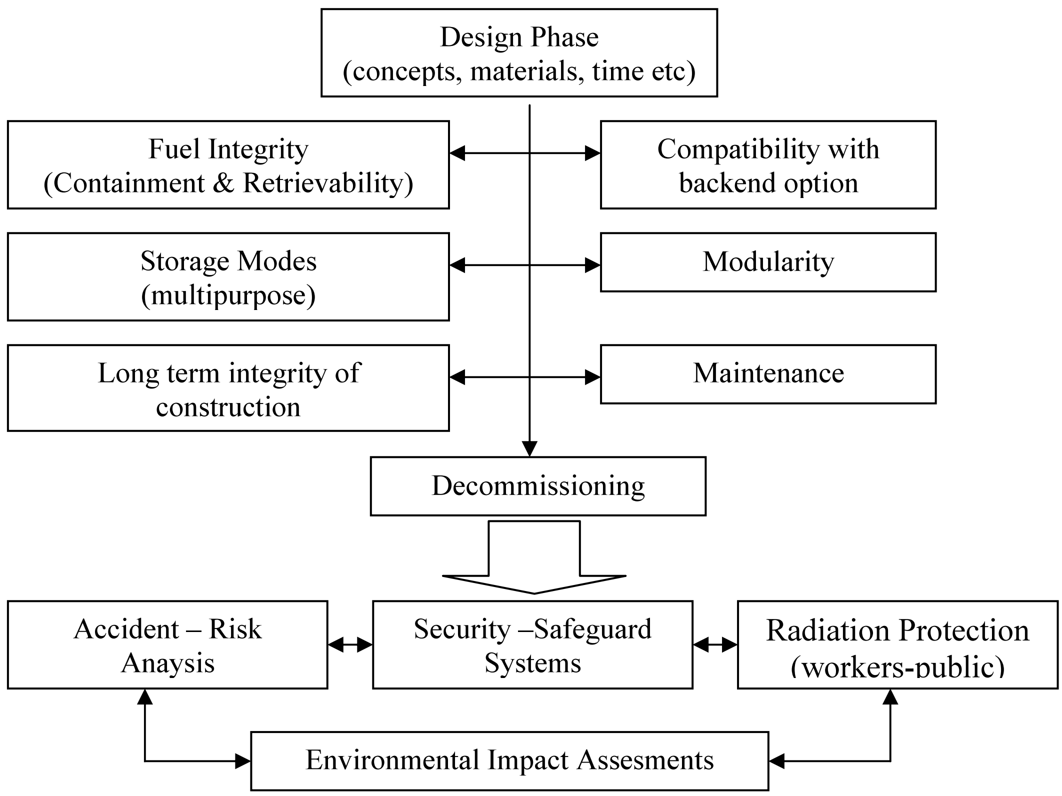

FIG 1. Technical evaluation criteria for selection of spent fuel storage technologies.

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Licensing of Zaporizhya NPP Spent Fuel Storage Facility. Implementation and Results

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The interim away-from-reactor spent nuclear fuel storage facility (ISFS) at the Zaporizhya Nuclear Power Plant is the very first facility of this type in the Ukraine. The ISFS is a storage facility for spent nuclear fuel from WWER-1000 reactors. It is located on the site of the Zaporizhya Nuclear Power Plant (NPP). ISFS consists of a storage pad with concrete storage casks placed on the pad. The storage cask design used for this WWER-1000 spent fuel is the VSC-24 storage cask used in the USA. The ISFS at the Zaporizhya NPP was created and is operated according to the regulation system in force in the the Ukraine.

The activity for creating ISFS was begun in 1993 when the situation with spent fuel storage in the Ukraine was very complicated. The need for cask adaptation to accommodate WWER-1000 nuclear fuel as well as Ukrainian rules and regulations made time durations for creating and licensing of ISFS lengthy. Accidents with VSC-24 casks which occurred in the USA also necessitated consideration of appropriate factors in the licensing process.

The development of the regulatory system in the field of nuclear energy utilization was being carried out in parallel with ISFS design development. In 1995, the Law "On the Use of Nuclear Energy and Radiation Safety" was passed. In particular, the Law sets requirements and conditions for receiving licences for appropriate kinds of activity connected with life cycle stages of nuclear facilities. Spent nuclear fuel storage facilities are recognized as nuclear facilities. Later on, other laws regulating relationships in the field of nuclear energy utilization and regulations containing organizational and technical requirements for nuclear facilities were developed and passed.

During more than 6 years of activity, the ISFS design was drastically updated. In accordance with the conditions of the ISFS commissioning licence, the very first cask was loaded and located on the storage pad in August 2001. During the commissioning period, a number of modifications of ISFS were also implemented following acceptance of their safety justifications.

One of the most significant updates of the ISFS design was the development of methodology for practical implementation of burnup credit (BUC). The initial methodology updated according to comments of the regulatory authority was recognized as acceptable and approved for use. The methodology met the requirements of rules and regulations in force and was used in the Ukraine for the first time.

Since 2004, ISFS has operated according to the conditions of Zaporizhya NPP operational licence. The licence contains the conditions of operation for Zaporizhya NPP units as well as Zaporizhya NPP spent fuel storage facility.

I. Pechera

This paper descibes the ISFS licensing history taking into account the regulatory assessments of importance for safety design modifications. The basic results of the licensing process, permitted conditions of operation and prospects for safety-related improvements are described. The licensing system for spent fuel storage facilities in the Ukraine, and the system of rules and regulations currently in force for such facilities are included in this description.

Management Criteria for Damaged Spent Fuel in Long-Term Storage

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1. Classification of Defects

- Damaged Spent Nuclear Fuel (DSNF): technological history
  - ◊ Operational parameters (burnup, cooling time, defects registration time, defects detection system)
- Mechanical defects classifier
  - ◊ Fuel assembly: defects of structural elements, breakdown of geometrical integrity, visually detectable defects of fuel cladding
- Cladding condition
  - ◊ Degree of cladding non-integrity (leakage), possibilities of identifying leaks, monitoring of the cladding condition in storage

2. Defects identification in storage

- Visual inspection to identify mechanical defects
  - ◊ The system for remote survey of fuel condition, ensuring quality of measurement and reliability of results
- Cladding condition
  - ◊ Application of technologically justified methods of control, development of characterization criteria

3. Design of facilities for DSNF retrieval, packing

- Designing for DSNF retrieval from storage.
  - ◊ Selection of retrieval options (“hot” cell, underwater method, cutting and packing of fuel assemblies)
- Site selection for DSNF packing activities
  - ◊ Studying of options for local DSNF packing activities
- DSNF drying
  - ◊ Ensuring water removal from outside of DSNF, removal of residual water from under the fuel cladding, as well as removal of chemically fixed water (in depositions)
- Organizational and technical actions for the management of DSNF spills
  - ◊ Ensuring collection, packing, storing of spills in compliance with safety requirements

4. Selection of container types for long-term storage of DSNF

- Ensuring two barriers of leaktighteness when storing DSNF

- Ensuring inert atmosphere in storage
  - ◊ Assessment of isotopic composition of the storage environment taking into account the release of gaseous materials from under the fuel cladding
- Reducing the impact of corrosion during storage
  - ◊ Taking into account the effects of the storage environment on corrosion processes for fuel cladding, structural materials of the fuel assembly and the elements of the storage system
- Excluding the possibility of exceeding critical pressure values during depressurization of fuel rods
- Design criteria for mechanical loadings
- Surface quality
  - ◊ Reduction of fixed contamination when packaging and storing DSNF
- Chemical compatibility
  - ◊ Ensuring non-interaction of storage system components and DSNF elements (isotopes)
- Operational reliability
  - ◊ Assessing processes taking place in storage structures and structural materials for containers and DSNF during normal operation and design accidents, including corrosion, creep, fatigue, shrinkage, ageing, changes caused by radiation, and other possible processes
- Costs
- Supplier stability
  - ◊ Ensuring the required productivity during DSNF management

5. Ensuring safety during transportation and storage of DSNF

- Organizing a list of events for DSNF storage conditions
- Fulfilling criticality requirements
- Ensuring that the requirements for radiation safety during normal operation, for emergency situations, design and beyond-design accidents are met
- Ensuring compliance with maximal temperature limits for fuel cladding during storage
- Ensuring fire- and explosion safety
- Monitoring fuel conditions and the storage environment

New Approach to Nuclear Safety Assessment of VVER-440 Reactor Pool

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The reactor pool (RP) for the existing fuel types of VVER-440 reactors meets nuclear safety conditions only under normal operations when it is completely filled with borated water. The analysis of nuclear safety, taking into account the conservative conditions required by regulatory documents, leads to violations of nuclear safety criteria.

The analysis has been carried out according to the up-to-date requirements provided in regulatory documents without giving credit to burnup, absorbers dissolved in water and removable absorbers. In addition, it was assumed in the analysis that the system is filled with a water-air mixture of the optimal density which corresponds to the maximum neutron multiplication factor. This is in full compliance with the regulatory documents in force as well as the existing international practice (optimal moderation condition).

The following results have been obtained under these conditions:

- In all cases RP racks without hermetic canisters are more dangerous (due to a lesser amount of steel which plays a role as neutron absorber);
- The criticality analysis of RP compartments without hermetic canisters results in Keff = 0.5161±0.0006 for normal operational conditions and aboric acid concentration of app. 2800 ppm;
- The maximum Keff for RP racks under an optimum air-water density of 0.25 g/ccm exceeds Keff = 1.3394± 0.0007 (unborated water is credited);
- The credit of boric acid dissolved in the RP water does not correct the situation substantially. Even at C_B = 2800 ppm we have Keff = 1.1752±0.0004 (air-water density of 0.117 g/ccm);
- In the analysis with simultaneous credit for boric acid (C_B > 1400 ppm) and fuel burnup, the system reaches Keff < 0.95 at the level of 50 MW day/kg . Under such conditions it is necessary to divide the RP into individual zones – for spent and low burnup (or fresh) fuel. Sparse assembly loading or an alternate loading with high burnup fuel, or an addition of an absorber in the material of the rack elements could be ways for achieving required level of criticality safety for a zone with low burnup fuel.

Most of the criticality calculations, whose results are presented in this report, have been obtained with the SCALE code pakage.

Several verification and confirmation calculations were made with the MCNP programmme.

The isotopic content of the spent nuclear fuel was calculated with the German cell code NESSEL (PHYBER programme complex). This code has been used for the last seven years to analyze the fuel loading of Ukrainian NPPs for the preparation of neutron-physical constants for VVER fuel depending on burnup.

Damage in Spent Nuclear Fuel Defined by Properties and Requirements

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The properties of light water reactor (LWR) fuel rods and assemblies are altered in service due to irradiation. Some of these alterations render the fuel unsuitable for emplacement in casks used for storage or transportation without special handling. Title 10 (Energy) of the U.S. Code of Federal Regulations Part 72 (storage) and Part 71 (transportation) establish direct requirements for the behavior expected of spent fuel. In particular, retrievability and prevention of gross breaches are required in storage and no reconfiguration of the fuel is allowed during normal transport. In addition, in the process of meeting other regulations related to criticality, shielding, and containment, the cask designers may need to place additional requirements on the behavior of the fuel. The definition of damaged fuel might be based on the ability of the fuel to perform in a manner such that the direct regulatory requirements and the onus placed on the fuel by the cask designer are met. Fuels that have alterations that do not permit it to perform its required safety function, without special handling, should be regarded as damaged. Since the requirements placed on the fuel may vary during phases of the fuel cycle, the potential exists for independent definitions to co-exist for interim dry storage, transport, and final disposal in a geologic repository.

The United States Nuclear Regulatory Commission's (USNRC) Spent Fuel Program Office (SFPO) has provided guidance in defining damaged fuel in Interim Staff Guidance (ISG) -1. This guidance is similar to that being developed by the American National Standards Institute (ANSI). Neither of these documents provides the logic behind the definition of damaged fuel. This paper will discuss the requirements placed on the fuel for dry interim storage and transportation and the ways that these requirements drive the definition of damaged spent fuel. Examples will be given illustrating the methodology.

Regulators Experiences in Licensing and Inspection of Dry Cask Storage Facilities

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All operating nuclear power reactors in the United States (US) are storing spent fuel in NRC licensed on-site spent fuel pools (SFPs). Most reactors were not designed to store, in these pools, the full amount of spent fuel generated during the life of plant operation. Utilities originally planned for spent fuel to remain in the SFPs for a few years after discharge from the reactor core and then to be sent to a reprocessing facility. However, the US Government declared a moratorium on reprocessing in 1977. Although the ban was later lifted, reprocessing has not been pursued as a feasible option. Consequently, utilities expanded the storage capacity of SFPs by the use of high-density storage racks. Eventually, utilities needed additional storage capacity. In response to these needs, NRC provided a regulatory alternative for interim spent fuel storage in dry cask storage systems. For spent fuel management, both pool storage and dry storage are safe methods, but there are significant differences. Pool storage requires a greater operational vigilance on the part of the nuclear power plant to maintain the performance of electrical and mechanical systems using pumps, piping and instrumentation. Dry storage technology uses passive cooling systems with robust cask designs requiring minimal operational vigilance.

The United States Nuclear Regulatory Commission (NRC), through the combination of a rigorous licensing and inspection program, ensures the safety and security of dry cask storage. NRC authorizes the storage of spent fuel at an independent spent fuel storage installation (ISFSI) under two licensing options: site-specific licensing and general licensing. In July 1986, the NRC issued the first site-specific license to the Surry Nuclear Power Plant in Virginia, authorizing the interim storage of spent fuel in a dry storage cask configuration. Today, there are over 30 ISFSIs currently licensed by the NRC with over 700 loaded dry casks. Current projections identify over 50 ISFSIs by the year 2010. No releases of spent fuel dry storage cask contents or other significant safety problems from the storage systems in use today have been reported. This paper discusses the NRC licensing and inspection experiences.

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The Management of Spent Advanced Gas-Cooled Reactor (AGR) Fuel

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In 1998 the approach to be taken by British Nuclear Fuels (BNFL) in discharging it's lifetime contracts for managing spent Advanced Gas-Cooled Reactor (AGR) fuel from British Energy's reactors was out-lined [1]. In the first instance this was to use existing facilities and storage techniques, i.e. wet storage, but recognised that if the initial option became untenable on safety grounds the alternative of constructing a purpose built facility would be taken. The decision would then be one of either to go down the existing wet storage technology route or to a develop a variation of the Scottish Nuclear Limited (SNL) dry store concept.

Since 1998 the operating climate has evolved. Changes which have impacted on the approach being taken include:

- Declaration of Magnox nuclear power plant lifetimes (23rd May 2000); now up to 2010 [2]. This has impacted on the availability of existing storage facilities for conversion to AGR storage only from storage in support of reprocessing. Originally the Magnox generation programme would have been completed prior to AGR storage only fuel being received to site; which provided a lead time for conversion of the site's Fuel Handling Plant.

- The restructuring of British Energy plc [3] completed 14th January 2005. For fuel loaded to reactor post restructuring, British Energy effectively lease the fuel and title for the fuel transfers to British Nuclear Group Sellafield Ltd. This changes the manner in which the spent fuel can be managed, previously this was influenced by the provisions in 'old' service contracts.

- Formation of British Nuclear Group in April 2004 in readiness for the opening of UK nuclear sites to be completed; in-line with the provisions made in the Energy Act 2004.

- Formation of the Nuclear Decommissioning Authority (NDA) in April 2005, under the Energy Act 2004, to take responsibility for the UK's nuclear legacy. The NDA approach [4] with respect to AGR spent fuel management will be to establish a national review for determining the longer term options for spent fuel. In the mean time the current strategy for managing AGR spent fuel [5], i.e. use of Thorp Receipt & Storage, is a 'bridging' solution to maintain power generation activities pending the out-come of NDA review.

- Changes in AGR station operating life [6], the projected Thorp end date (currently 2010/11) and site wide plant performance [4] all impact upon the required storage capability with time; at 1st April 2004 [7] the long-term capability need to provide storage for 3500tU. This requires flexibility in the end solution to accommodate all potential life time spent fuel scenarios.

Mean while the underpining engineering and spent fuel longevity studies have also evolved in-line with the changes in the operating environment. In the near term the engineering studies are focused upon developing the Thorp Receipt & Storage asset as a 'bridging' solution to maintain AGR fuel receipts in-line with NDA requirements.

These studies have established the existing plant limitiations which influence key enablers such as the potential storage capability, plant longevity and the logistics of transfering from one storage environment to another. When taking into to consideration that TR&S is a production facility and a seamless transfer from one storage mode to another will be required then the extent of modifications that can be accommodated is minimal. For example, increasing the storage capability for AGR fuel is influenced by the existing plant set-up, floor loading, the 'Defence in depth' bolt and lock grid positioning system and the mechanical lifting equipment. This has lead to optimised storage system design proposals that are a variation to either the existing AGR storage skip & container and LWR Multi-element Bottle & rack designs. Currently, a 7x7x3 storage array design based upon an LWR rack is the preferred option.

In terms of the spent fuel longevity studies these have focused on meeting the long term storage requirement that the primary containment barrier (fuel cladding) should have sufficient mechanical integrity after 80 years storage to enable the fuel to be removed safely for further conditioning. To date these studies have been focused on the wet storage angle as the dry storage aspects were mostly established by the Scottish Nuclear Limited dry storage project. Activities in this area have established the general corrosion rate of the cladding in the corrosion inhibitor sodium hydroxide, the evalaution of alternative corrosion inhibitors and the development of an on-line corrosion probe.

To supplement the project studies aimed at underpining the 'bridging' solution, the strategic options for the longer term are being evalauted as part of the overall site strategy. The studies in the spent fuel area have re-evaluated the existing wet storage position, but have also been broadened to evaluating new build options and in particular the multitude of dry storage technologies.

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VNIPIET Projects and Design Related to Management of Spent Fuel from Nuclear Power Reactors

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The nuclear energy strategy in Russia is based on proven water cooled advanced VVER-1000 + reactors as well as on fast neutron reactors. Closing of the nuclear fuel cycle with reprocessing and recycling of both uranium and plutonium is a key factor. The accumulation of plutonium is planned to be implemented in BN-type fast neutron reactors.

The share of nuclear electricity generation is planned to be increased from the present 16 % to up to 25 % by the year 2030.

Large scale constructions of fast neutron breeder reactors are needed in the long term perspective in order to solve the problem of nuclear fuel supply.

Commissioning of the BN-800 fast breeder reactor at the Belojarsk NPP will provide a possibility to develop a complete fuel cycle infrastructure, including MOX-fuel manufacturing, irradiated fuel reprocessing and waste isolation.

At a later stage a demonstration reactor with a BN-1800 fast breeder and the respective nuclear fuel cycle facilities will be built.

Finally by mid-century both systems advanced VVER+ and BN-1800 reactors will be in operation on commercial scale. The whole nuclear infrastructure will be based on the closed nuclear fuel cycle with improved technology in the “front end” and in the “back end” of the fuel cycle.

At present nuclear energy is being used in many countries, but only few of them have obtained sophisticated nuclear fuel cycle technology and have built the necessary infrastructure.

The implementation of any new nuclear energy programme is impossible unless spent nuclear fuel and radioactive waste are properly managed and wastes are finally disposed of.

Russia belongs to the group of countries that established sophisticated nuclear fuel cycle technologies and the related infrastructure necessary for the safe management of spent fuel.

The Russian closed nuclear fuel cycle concept is implemented at present at the “Mayak” facilities by reprocessing of spent nuclear fuel from VVER-440 reactors jointly with spent nuclear fuel from research reactors.

For many years the VNIPIET institute was involved in projects and design work related to the safe and reliable management of spent nuclear fuel at all post-reactor stages including water-cooled storage ponds (at reactor, away from reactor) and regional spent nuclear fuel storage facilities. Handling and transport systems for spent nuclear fuel as well as radiochemical reprocessing and radioactive waste conditioning facilities were developed.

The operation of these facilities for many years demonstrated safety and reliability of equipment and technology developed by VNIPIET and its counter part research organizations.

Recently some new projects were developed:

- Capacity extension of the existing storage facilities for spent nuclear fuel at several NPP's;
- Capacity extension of centralized water-cooled storage facilities at MCC;
- Developing a new centralized dry storage facility with a capacity of 38 000 t at the MCC site.

Chemical reprocessing is a key stage of spent fuel management for closed nuclear fuel cycle system.

VNIPIET jointly with some research organizations has made a number of feasibility studies devoted to improvements of existing reprocessing technology and development of new technologies to meet requirements of safety, waste minimization and non-proliferation.

Fractioning of fission products and new systems for HLW immobilization were considered.

In the future a step by step development ofa new project for the RT-2 reprocessing plant is planned. This complex will also include waste immobilization and disposal facilities.

Handling and transportation systems are also important for spent fuel management. Experience in the development and operation of transportation equipment and containers for spent nuclear fuelwill also be presented.

The possibility of using Russian nuclear fuel cycle infrastructure for solving global problems of spent fuel management will be discussed.

Existing research and design potential, test facilities, and valuable operational experience can be used as a solid basis for the creation of a new advanced spent nuclear fuel infrastructure, urgently needed for large scale nuclear energy programmes.

Application of Burnup Credit for the Storage of Spent Fuel at the Zaporizhya NPP

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The interim away-from-reactor spent nuclear fuel storage (ISFS) facility at the Zaporizhya Nuclear Power Plant (NPP) is the very first facility of this type in the Ukraine. The ISFS facility is for storage of spent nuclear fuel from WWER-1000 reactors. It is located on the site of the Zaporizhya NPP. Dry storage concrete casks are used at the ISFS facility. The prototype design for these storage casks is the VSC-24 storage cask.

In accordance with the conditions of the ISFS facility commissioning licence, the very first cask was loaded and located on the storage pad in August 2001. During the commissioning period, a number of ISFS facility updates and modifications were also implemented. Related safety justifications were recognized as acceptable. One of the most significant updates of the ISFS facility design was the development of methodology for the practical usage of burnup credit. It was recognized as acceptable and permitted for usage. The actual development and implementation of burnup credit methodology in Ukraine was carried out for the operation of the ISFS facility.

The burnup credit methodology implemented is described in the paper. The paper gives details on calculations and measurements tools, safety margins, verification and validation of models and equipment usage.