SAFETY AND LICENSING OF SPENT FUEL STORAGE AND TRANSPORTATION

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REGULATORY ASPECTS OF SPENT FUEL STORAGE AT PAKS MVDS FACILITY

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Abstract

The paper intend to show the regulatory oversight activities in connection with the Spent Fuel Interim Storage Facilty (SFISF). Licensing procedures, inspections and evaluation of the safety performance are introduced. To illustrate the safe operation of the facility, occupational and public exposure data and radioactive releases are summarized. The first Periodic Safety Review has been implemented sucessfully in 2008. Having revised the Final Safety Assessment Report of the SFISF the operational license of the entry hall and of chambers 1–16 has been extended by the HAEA. The new operational license is valid until the next Periodic Safety Assessment Review (November 30, 2018).

1. DESCRIPTION OF THE FACILITY

Interim dry storage of the spent fuel assemblies that are generated during the operation of the Paks NPP is provided in a Modular Vault Dry Storage (MVDS) facility. The Spent Fuel Interim Storage Facility (SFISF) started its operation in 1997 and provides for at least 50 years of interim storage for VVER-440 fuel assemblies in a contained and shielded arrangement. The fuel assemblies are stored vertically in individual fuel storage tubes, the matrix of storage tubes being housed within a concrete vault module that provides shielding. To minimize corrosion and degradation processes the assemblies are stored in an inert nitrogen environment inside the tubes. The SFISF can be divided functionally into three major structural units. The layout of the facility can be seen on Fig. 1.

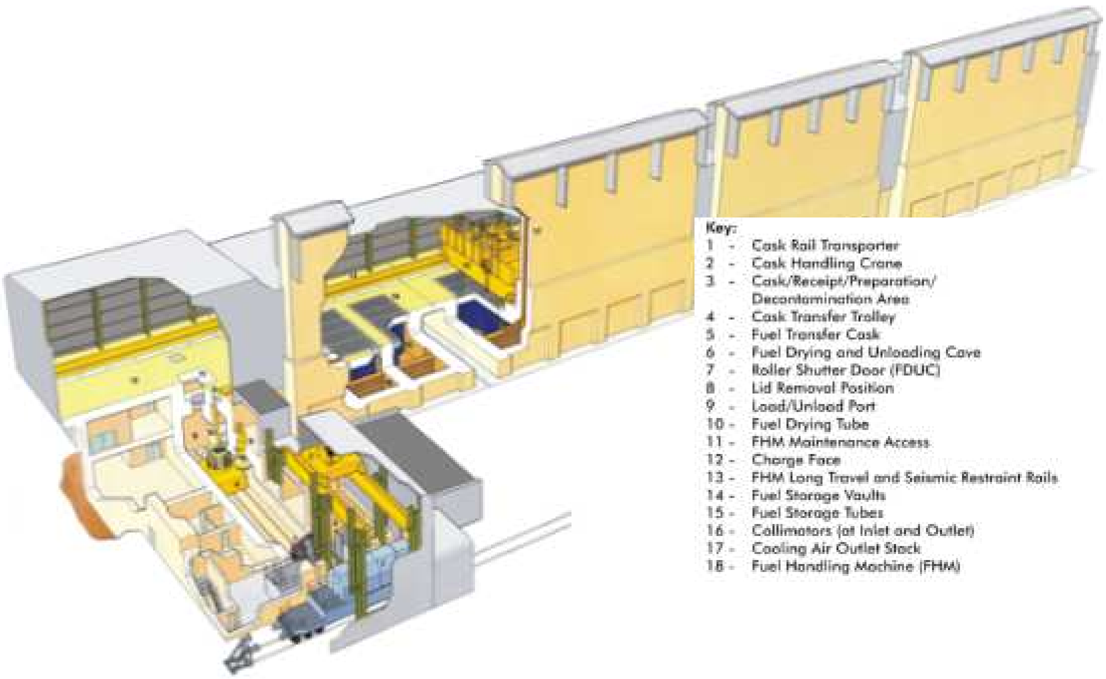

FIG 1. The layout of the Modular Vault Dry Storage facility.

The first structural unit is the vault module where the spent fuel assemblies are stored in the tubes. These vault modules include a minimum of three or maximum five chambers depending on the geometrical arrangement. The second major structural unit is known as the charge hall where the fuel handling machine travels during the fuel handling operations. The third major unit is the Transfer Cask Reception Building (TCRB) in which the reception, preparation, unloading and loading of the transfer cask takes place. The fuel handling system and other auxiliary systems are installed in the TCRB.

The first license to operate the SFISF was issued to the operator of the Paks NPP as the owner and licensee of the storage facility. According to the modified Act on Atomic Energy that became effective parallel with the original commissioning of the storage facility, a new agency called Public Agency for Radioactive Waste Management (PURAM) was established in 1998. PURAM was designated to carry out the multilevel tasks in the field of the radioactive waste management including the interim storage of the spent fuel. The NPP staff operates the SFISI in a contractual arrangement with the PURAM.

According to the Act on Atomic Energy, the relevant authority regarding nuclear facilities and thus regarding the SFM facilities is the Hungarian Atomic Energy Authority (HAEA).

2. REGULATORY OVERSIGHT OF THE FACILITY

2.1. Licensing procedures

The basic principles of the licensing procedure for the SFISF are analogous to those of all other nuclear facilities. A nuclear safety license should be obtained from the HAEA for all periods during the lifecycle of the facility (siting, construction, commissioning, operation; within and beyond the planned life-time, respectively), decommissioning (final shutdown and dismantling). Moreover, separate licenses must be obtained for all changes of construction to a given facility or modifications to its components/constructions should they belong to safety classes. In addition to this, the HAEA grants building and occupancy licenses for buildings and structures.

Within the licensing procedures, the specific aspects are dealt with by the special authorities designated by law. The HAEA has to take into consideration the additional requirements (stipulations and conditions) of these specialized authorities. Licenses are valid for a given period of time, and may be extended upon request of the licensee if all requirements are met.
In 2009 HAEA issued 192 decisions, out of which 168 are resolutions and 24 are procedural orders. 153 of the decisions were connected to the Paks NPP, 15 to the SFISF, and 14 to the two research reactors.

The most important decisions for the SFISF were the following ones:

- Renewal of the operation license for modules 1–16;
- Approval of corrective actions (according to the comprehensive inspections made in 2008);
- Approval of modified OLCs;
- Approval of Standard for Environment Monitoring;
- Significant modifications (e.g. a new camera system for safeguard purposes).

2.2. Inspection

The HAEA is entitled to perform inspections both with advance notice, or without notice should the latter be considered justified. In addition to this, the authority performs comprehensive inspections in every third year. These inspections cover the following areas:

- Realization of actions decided after the Periodic Safety Review (PSR);
- Emergency preparedness;
- Operation of radiation protection equipment, including certification and calibration;
- Maintenance, ageing management;
- Management of contractors;
- Documentation management, system and current status of internal rules;
- Feedback of operating experience and R&D results;
- Execution of technical reviews, status of service manuals;
- Categorization of safety classified measurements.

The authority summarizes the deficiencies and deviations requiring improvement measures in an itemized and summary report. The licensee is to submit an action plan to schedule the required corrective actions. The last comprehensive inspection for the SFISF has been performed in 2008.

2.3. Evaluation of safety performance

The authority operates a reporting system. Reports prepared for the regulatory body are detailed so as to enable independent review, evaluation and assessment of operating activities, and any noteworthy events that may have taken place. The investigation and assessment of any events affecting safety that have occurred during operation and the identification of the causes and the taking of corrective actions and measures in order to prevent their repeated occurrence is primarily the task of the licensee.

The HAEA evaluates annually the safety performance of all licensees based on the results of a Safety Performance Indicator System. The aim of this evaluation is the regulatory assessment of the activities and safety performance of a licensee, and thus monitoring and assessing the safety indicators of the operation as well as identifying probable safety gaps in a timely manner.

The licensee PURAM submits Quarterly Reports and Annual Reports to the HAEA. The detailed contents of these regular reports are prescribed in the operation license. Some examples from the annual reports can be seen in Tables 1–3:

- Table 1 shows the airborne radioactive releases into the environment in 2009;
- Table 2 summarizes the exposure data for the personnel in the past decade;
- Table 3 summarizes the annual exposures (1998–2009) of the public assessed from the airborne and liquid releases, in comparison with the authorized dose constraint (10 μSv/a effective dose for the member of the critical group).

TABLE 1. AIRBORNE RELEASES OF SFISF IN 2009 (DATA ABOVE THE DETECTION LIMITS ARE IN BOLD)

Nuclide	Annual release, $R_{i\ airborne}$ [Bq]				Release limit, $EL_{i,\ airborne}$ [Bq/a]	Release limit criterion, $\sum_{i,airborne} \frac{R_{i,\ irborne}}{EL_{i,\ airborne}}$
	Outlet stack	Charge face	Fuel handl. machine	Total		
^{3}H	6.12E+07	—	—	6.12E+07	0E+14	2.92E-07
^{14}C	7.36E+06	—	—	7.36E+06	0E+11	2.83E-05
^{51}Cr	2.16E+05	3.41E+05	1.09E+03	5.57E+05	0E+13	3.72E-08
^{54}Mn	2.23E+04	3.48E+04	1.10E+02	5.72E+04	0E+11	5.20E-07
^{58}Co	2.33E+04	3.56E+04	1.12E+02	5.90E+04	0E+11	3.28E-07
^{59}Fe	4.73E+04	7.35E+04	2.32E+02	1.21E+05	0E+11	6.73E-07
^{60}Co	3.28E+04	4.91E+04	1.55E+02	8.20E+04	0E+09	1.05E-05
^{95}Zr	4.07E+04	6.24E+04	1.97E+02	1.03E+05	0E+10	1.29E-06
^{95}Nb	2.40E+04	3.68E+04	1.19E+02	6.10E+04	0E+11	4.69E-07
^{106}Ru	2.14E+05	3.20E+05	1.01E+03	5.35E+05	0E+1	4.46E-05
^{110m}Ag	2.24E+04	3.38E+04	1.07E+02	5.63E+04	0E+09	7.93E-06
^{125}Sb	6.80E+04	1.03E+05	3.21E+02	1.71E+05	0E+10	2.38E-06
^{134}Cs	2.41E+04	3.64E+04	1.14E+02	6.06E+04	0E+09	7.48E-06
^{136}Cs	2.73E+04	4.49E+04	1.45E+02	7.23E+04	0E+10	7.94E-07
^{137}Cs	2.46E+04	3.69E+04	1.17E+02	6.17E+04	0E+09	6.78E-06
^{141}Ce	3.81E+04	5.97E+04	1.90E+02	9.79E+04	0E+11	4.66E-07
^{144}Ce	1.53E+05	2.32E+05	7.27E+02	3.86E+05	0E+10	2.41E-05
^{89}Sr	7.70E+02	—	—	7.70E+02	0E+11	4.05E-09
^{90}Sr	7.37E+02	—	—	7.37E+02	0E+09	1.32E-07
^{234}U	8.68E+01	2.69E+02	4.19E-01	3.56E+02	0E+09	1.62E-07
^{238}U	7.92E+01	2.54E+02	3.78E-01	3.34E+02	0E+09	1.39E-07
^{238}Pu	5.93E+00	8.69E+00	2.06E-02	1.46E+01	0E+08	1.56E-08
$^{239/240}Pu$	2.92E+01	1.45E+02	1.51E-01	1.74E+02	0E+09	9.68E-08
^{241}Am	1.82E+01	6.53E+01	9.62E-02	8.36E+01	0E+09	8.36E-08
^{242}Cm	5.93E+00	8.69E+00	2.06E-02	1.46E+01	0E+09	2.32E-09
^{244}Cm	8.76E+00	2.55E+01	3.44E-02	3.43E+01	0E+09	2.64E-08
Total:						1.38E-04

TABLE 2. RADIATION EXPOSURE OF WORKERS IN SFISF DURING THE PERIOD OF 2000–2009

Year	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009
Number of loaded fuel assemblies	500	750	420	480	270	500	480	360	480	480
Highest individual dose [mSv]	0.071	0.035	0.582	0.849	0.168	0.298	0.198	0.203	0.220	0.260
Collective dose [man*mSv])	2.1	2.1	8.6	6.4	3.7	6.3	7.6	6.6	9.6	8.3

TABLE 3. ASSESSED PUBLIC EXPOSURES FOR THE PERIOD OF 1998-2009, IN COMPARISON WITH THE AUTHORIZED DOSE CONSTRAINT 10 MSV/A

Year	Annual effective dose for the member of the public					
	Liquid releases		Airborne releases		Total	
	[nSv/a]	[%]	[nSv/a]	[%]	[nSv/a]	[%]
1998	0.11	0.0011	0.12	0.0012	0.23	0.0023
1999	0.04	0.0004	0.06	0.0006	0.10	0.0010
2000	0.17	0.0017	0.1	0.0010	0.27	0.0027
2001	0.10	0.001	0.11	0.0011	0.21	0.0021
2002	0.14	0.0014	0.08	0.0008	0.22	0.0022
2003	0.08	0.0008	0.09	0.0009	0.17	0.0017
2004	0.05	0.0005	0.07	0.0007	0.12	0.0012
2005	0.13	0.0013	0.20	0.0020	0.33	0.0033
2006	0.539	0.00539	0.437	0.00437	0.976	0.00976
2007	0.276	0.00276	0.683	0.00683	0.96	0.0096
2008	0.926	0.00926	1.84	0.0184	2.77	0.0277
2009	0.902	0.00902	1.38	0.0138	2.28	0.0228

2.4. Unified operation license for the SFISF facility

In 2008, the extension and commissioning of the SFISF by the 12–16 storage chambers came to completion. The HAEA decided on issuing a uniform operating license for the formerly built 1–11 and the new 12–16 chambers constructed during the extension. The PURAM elaborated the uniform Final Safety Analysis Report (FSAR) which, besides being applicable to support the operating license of the second phase, was also applicable to support the uniform operating license. The HAEA compared the FSAR with the Periodic Safety Report (2007) of the first eleven chambers. During the evaluation of the uniform FSAR, the authority concluded that the document basically met the content and formal requirements. The safe operation of the facility is justified; however taking account of the improvement actions and supplements prescribed, the first renewal of the operating license of chamber 1-16 was valid until only December 31, 2009, since the HAEA in its decision closing the PSR specified improvement actions and determined modifications and supplementations in the operating license for the FSAR. In the first quarter of 2009 the PURAM elaborated the modified FSAR and a new operation license has been issued (valid until November 30, 2018). According to the legal provisions, the special authorities took part in the licensing procedures. The environmental authority required monitoring of the radioactive releases in the facility's environment, the police contributed with security requirements, while the fire protection authority has also set specific requirements regarding fire safety.

REFERENCES

[1] HUNGARIAN GOVERNMENT, Governmental Decree 89/2005. (V. 5.) on the nuclear safety requirements of nuclear facilities and the related regulatory activities. Annex No. 6. Nuclear safety code for spent fuel interim storage facilities (2005).

[2] PUBLIC LIMITED COMPANY FOR RADIOACTIVE WASTE MANAGEMENT, Annual Reports of SFISF (1998-2009), PURAM (2010).

ACTIVITIES RELATED TO SAFETY REGULATIONS OF SPENT FUEL INTERIM STORAGE AT JAPAN NUCLEAR ENERGY SAFETY ORGANIZATION

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Abstract

Major research activities in safety regulation of spent fuel interim storage at JNES are presented. In Japan, the first license application was approved by the government in May 2010 and the design and construction method will be submitted to the regulatory authority NISA soon. A commencement of its operation is expected at December 2012. In its plan, dual purpose metal casks for storage and transport will be stored in a concrete building for about 50 years, and then they will be transported to a spent fuel reprocessing facility. When they will be shipped out after the storage, no visual inspection for cask internals will be scheduled. Major reason of no visual inspection is to avoid any radiation exposure from contingent incident during opening the casks lid. JNES as TSO has conducted research activities to support NISA. Before the license application, those activities focused on three areas. The first area was to investigate fundamental safety function of the cask, that is, confinement, shielding, heat removal and subcriticality. Especially, a long term performance of the safety function was key issues. The second one was to confirm integrity of spent fuel cladding during the storage. The third one was to improves and verify the computer codes and/or methods for safety evaluation of the spent fuel interim storage facilities. In usual safety review process in Japan, NISA sometimes asks JNES to perform independent analysis and check the adequacy of the safety analysis conducted by licensees. After the approval of the license application, the applicant should have approvals of “design and construction method”, the welding inspection of the cask and the pre-service inspection. JNES is now supporting to prepare the criteria of the design and construction method.

1. INTRODUCTION

Japan Nuclear Energy Safety Organization (JNES) was established in October 2003 as Technical Support Organization (TSO) for the regulatory authority Nuclear and Industrial Safety Agency (NISA). The mission of JNES is to establish a base to ensure nuclear safety in the nuclear utilization for energy by performing safety analysis, evaluations, etc. of the design of nuclear installations and nuclear reactor facilities, while conducting the inspection etc. of nuclear installations and nuclear reactor facilities.

In the area of spent fuel interim storage, before the establishment of JNES, researches to investigate fundamental safety functions of dual purpose metal cask had been performed by Nuclear Power Engineering Corporation (NUPEC). After the establishment of JNES, they were transferred to JNES. The transferred researches included investigation of fundamental safety function of casks under conditions of both long term storage and accident in post-storage transport, and integrity of spent fuel cladding during the storage. The researches for the fundamental safety function included relaxation and corrosion of metal gasket, sealing performance of lid with fresh or aged metal gasket, corrosion of metal body and welding part induced by deterioration of cavity atmosphere, mechanical property degradation and corrosion of basket material, and long term degradation of neutron shielding material. Furthermore, the seal performance under accident conditions was also investigated experimentally. These included the seal performances under 30 minutes of 800°C and after 9 meters dropping.

In Japan, interim storage facilities will be built at coastal area and corrosion induced by salinity atmosphere might be one of threats for long term integrity of canister for concrete type cask. Newly started researches after JNES establishment were to investigate welding method of canister and long term integrity of high burn up fuel. In Japan, interim storage facilities will be built at coastal area and corrosion induced by salinity atmosphere might be one of threats for long term integrity of canister. Therefore, corrosion resistance stainless steel is considered as material of canister. Corrosion resistance of the stainless steel was investigated by CRIEPI [1]. JNES performed studies to investigate welding process of the stainless steel and inspection procedure of the welding.

2. RESEARCHES FOR FUNDAMENTAL SAFETY FUNCTION OF CASK

2.1. Material test

In Japan, casks installing spent fuel are stored at the facility for about 50 years and transported to the reprocessing plant. When shipping from the facility, the lid is not expected to reopen in order to avoid any radiation exposure from contingent incident during the reopening. Therefore, fundamental safety function of cask should be investigated, considering both storage and post-storage transport conditions. Before starting experimental works, existing information with respect to the dry storage technology and material behavior was reviewed and the results are summarized in Table 1 [2]. Material tests program was prepared based on this table.

TABLE 1. POSSIBLE DEGRADATION PHENOMENA OF METAL CASK COMPONENT

	Heat	Radiation	Atmosphere
Cask body, Lid (Carbon steel, Stainless steal)	NS	NS	Corrosion, SCC
Basket (Borated Aluminum alloy)	Overaging, Creep	NS	Corrosion, SCC
Neutron shielding (Resin, Propylene glycol(PG)-water)	Composition change	Composition change	—
Seal boundary (Metal gasket)	Relaxation	NS	Corrosion, SCC

2.1.1. Corrosion and SCC tests of cask body, lid, basket and metal gasket

Only intact spent fuels will be installed in the cask and the inner cavity will be maintained inert atmosphere during the storage. While the fuel temperature during the storage is kept below a specified value, the integrity of the spent fuel cladding is expected to be kept. However, there is not enough data to refuse the possibility of cladding failure during 50 years storage. In the studies, it was assumed that 1% of fuel pins would be failure and FP gas released, and the influence of corrosive gas, iodine, on cask body, metal gasket and basket was experimentally investigated. Results showed that general corrosion was observed but no pitting or local corrosion for all specimens. Thickness decrease of each material during 50 years estimated based on obtained data is very small.

2.1.2. Material property change of basket

The basket would be exposed to high temperature and radiation during the storage. Borate aluminum alloy is candidate material of basket in Japan. The role of the basket is to absorbing neutron, removing the decay heat and maintaining spent fuels arrangement for subcriticality. From review of existing information on mechanical property change of aluminum due to irradiation, it was evaluated that expected cumulative irradiation dose during the storage did not significantly influence mechanical property of aluminum. Since mechanical property degradations of aluminum and borated aluminum due to irradiation seemed to be not so different, it was judged that irradiation effect on borated aluminum was not significant. On the other hand, an effect due to heat was experimentally evaluated. Aluminum (JIS H4080 A5052 H34, 5wt%B4C borated aluminum Alloy (Base: JIS H4100 A6N01), 1wt% over borated aluminum Alloy (Base: ASTM A6351-T5), 1wt% borated Aluminum Alloy (Base: ASTM A3004-H112) were investigated. In the experiments, test specimens were kept at temperature up to 250°C during 10,000 hours. Figure 1 shows comparison of proof strength at 250°C. Annealing made strengths lower. Further, these strengths were not changed if additional creep deformation was provided. Absorbing energy at impact test was almost same as or slightly more than those of fresh specimens. There was no important change for micro structure and the other properties.

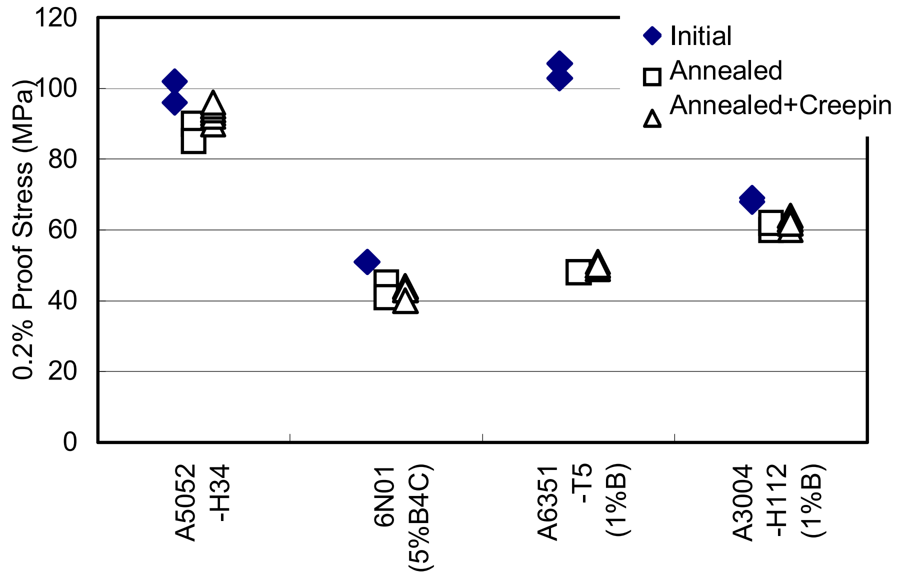

FIG. 1. Comparison of proof strength (at 250°C).

2.1.3. Neutron shielding material

Epoxy resin, silicon resin and PG-water as neutron shielding was investigated for thermal ageing and irradiation effect. Test specimens were heated up to 170°C, and the maximum heating period was 15,000 hours. Cumulative irradiation dose of neutron and gamma ray were 1.5×10^{15} n/cm^2 and 3.9×10^4 Gy at a maximum, respectively. Figure 1 shows weight loss of epoxy resin with a function of Larson-Miller Parameter (LMP) after the long term heating. The weight loss was estimated to occur by release of low molecular weight from base material and of H_2O due to dehydrate reaction of tri-hydrate-alumina. The influence of heating on weight loss was dominant but that of neutron and gamma ray irradiations was small. In addition, there was no synergistic effect of heating and irradiation. As for PG water, temperature of 100–180°C was applied during 10,000 hours. The results showed that the density decrease of PG water was small and a shift of freezing point was very small.

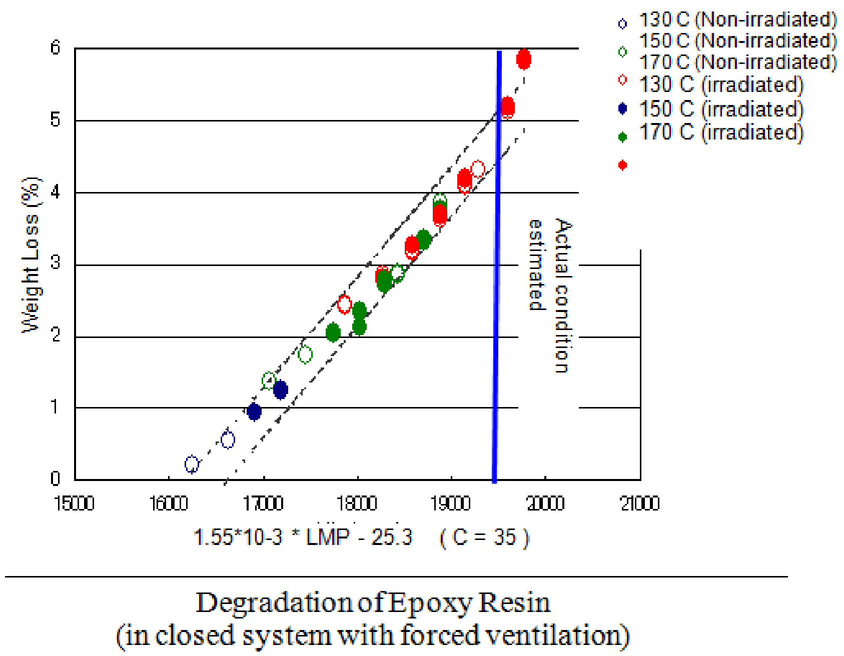

FIG. 2. Degradation of Epoxy Resin.

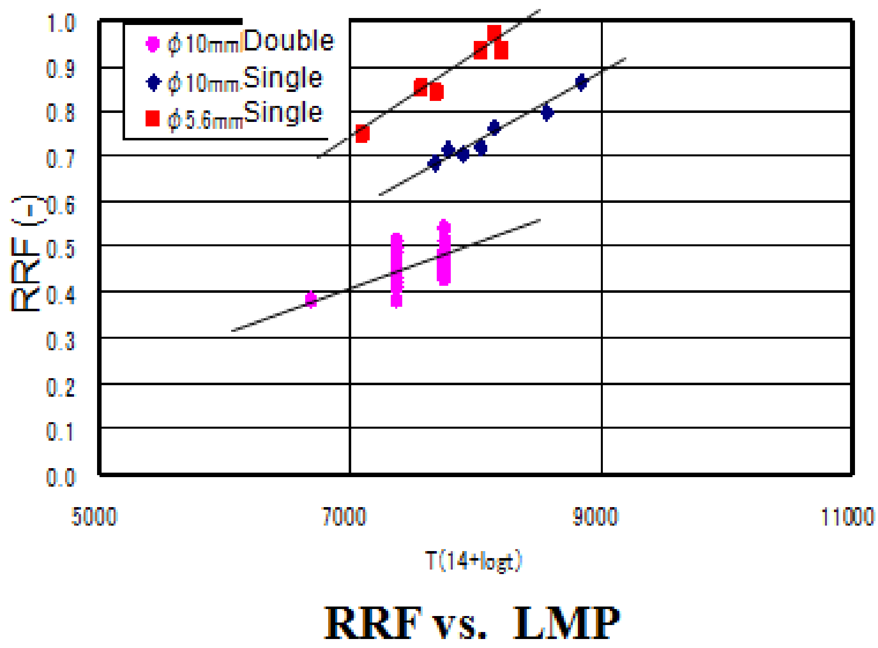

FIG. 3. Reduction ratio of reaction force vs. LMP.

2.1.4. Degradation tests of metal gasket

Metal gasket would be applied to the lid containment system and might be degraded by heat and irradiation. From review of existing information, it was estimated that expected cumulative irradiation dose during the storage would be less than a value causing significant mechanical property change of gasket materials. On the other hand, the existing studies showed that the relaxation due to continuous high temperature could occur. Generally, the metal gasket was assembled from a coil spring, inner and outer shells, and the leak tight would be achieved by the outer shell adhered to flange surface. The following tests were conducted in order to simulate the relaxation and to verify the effects, (i) heating tests to measure the relaxation of metal gasket, (ii) seal performance tests against quasi-static or dynamic with and without relaxation. Single type and double type gaskets were examined. Materials used in the tests were high nickel alloy for both coil spring and inner jacket and aluminum for outer jackets, respectively. Fig. 3 shows reduction ratio of reaction force (RRF) with a function of LMP. In the seal performance tests, lid displacement and leak rate were measured. The maximum leak rate was about 10^{-5} Pa·m^3/s and the maximum residue radial displacement was about 5 mm. It was observed that the measured leak rate increased depending on the residual displacement.

2.2. Mock-up test

To confirm the seal performance of the lid containment system under drop accident condition during post-storage transport, 9 m drop tests were conducted using an actual size metal cask. The test parameters were orientation of cask, that is, horizontal, vertical and corner drops. In horizontal drop test, two tests were conducted, one with fresh gasket and another with thermal aged gasket. Accelerations of main components, strains around body flange and of lid bolts, and relative displacements of the lid to the flange were recorded in the tests. Leak rate of primary and secondary lids, and pressure between lids were inspected before and after each test. Leak rate of the secondary lid with thermal aged metal gasket was estimated lower than10^{-4} Pa·m^3/s after dropping in each test. Measured leak rates were compared with the results of degradation tests of metal gasket. Here, relative displacement between the lid and the body flange of each drop test was the sum of go and back displacements in the drop. As shown in Fig. 4, both results are well agreed. It was verified that the sum of relative displacement was a useful factor to estimate leak rate at drop accident. Post-test analysis were conducted using DYNA-3D of general FEM code for dynamic response analysis for each drop, and the analysis results could well simulate the time histories of relative displacement between the lid and the body flange considering the initial relative position between lid and body flange.

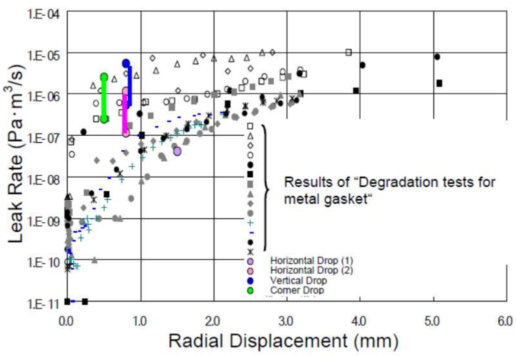

FIG. 4. Leak rate vs. relative radial displacement of lid.

3. INTEGRITY TEST OF SPENT FUEL

Japanese regulation requires that the integrity of spent fuel during shall be maintained during the storage. From the viewpoint of prevention for the failure of spent fuel due to cladding tube thermal creep and to the degradation of cladding tube mechanical properties, the temperature and cladding stress during the storage shall be limited. As shown in Table 2.2, JNES conducted the following three tests [3–6]; thermal creep test, irradiation hardening recovery test and hydride reorientation test from 2000–2008.

TABLE 2. SCHEDULE FOR SPENT FUEL INTEGRITY

Item	FY	2000	2001	2002	2003	2004	2005	2006	2007	2008
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 BWR40GWd/t, 50GWd/t, 55GWd/t	—	—	—	—
Irradiation Hardening Recovery Test		PWR48GWd/t, BWR50GWd/t (330-420°C)		—	—		(<330°C) —	—		

3.1. **Thermal creep test**

In creep test [3–4], creep deformation properties of unirradiated and irradiated BWR 50GWd/t type Zry-2 and PWR 48GWd/t type Zry-4 claddings have been investigated under various temperature (603–693K) and hoop stress (50–300 MPa for Zry-2, 30–250 MPa for Zry-4) conditions of the two different deformation mechanism regions, those are speculated as grain boundary sliding and dislocation creep, based on the Zircaloy creep deformation map by Chin and others [7].

Creep strain was expressed by Equation (1) in consideration of conventional primary creep and secondary creep concept.

$$\varepsilon = \varepsilon_p^s + \dot{\varepsilon}\, t \qquad (1)$$

Where,

ε is the creep strain;

ε_p^s is the saturated primary creep strain;

$\dot{\varepsilon}$ is the secondary creep rate;

t is the time .

The second creep rate was measured as parameters of stress and temperature, and the stress dependency are shown on Figure 5.

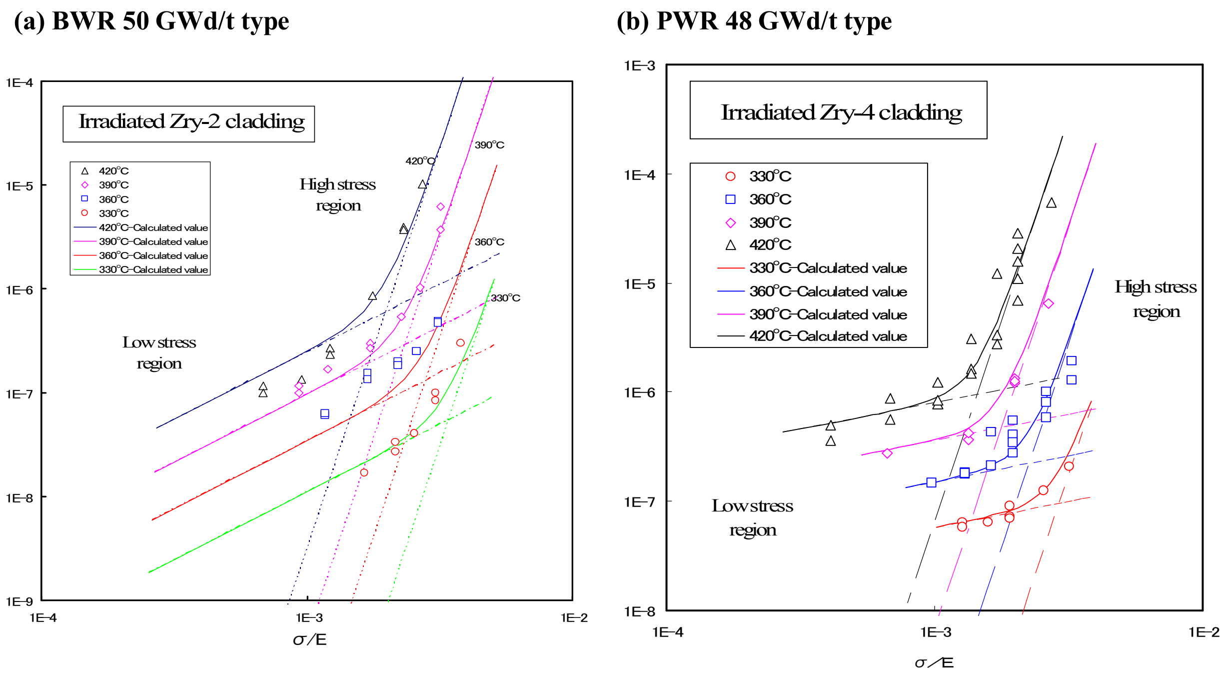

FIG. 5. Stress dependency of secondary creep rate (Hoop stress, E:Young's modulus).

The threshold creep strain of transition to tertiary creep from secondary creep was studied. The relation between the threshold strain and secondary creep rate of Zry-4 cladding is shown in Fig. 6. The threshold strain of transition to tertiary creep is likely to be more than 10% for unirradiated cladding and 1% for irradiated cladding.

The thermal creep behavior of irradiated 55 GWd/t type fuel cladding was also investigated. As for BWR cladding, the creep test results for 55 GWd/t type Zry-2 cladding showed the good agreement with those for 50 GWd/t type Zry-2 cladding. As for PWR claddings, test results indicated that there was not large difference in thermal creep behavior between 48 GWd/t type Zry-4 and 55 GWd/t type cladding material (MDA and ZIRLO).

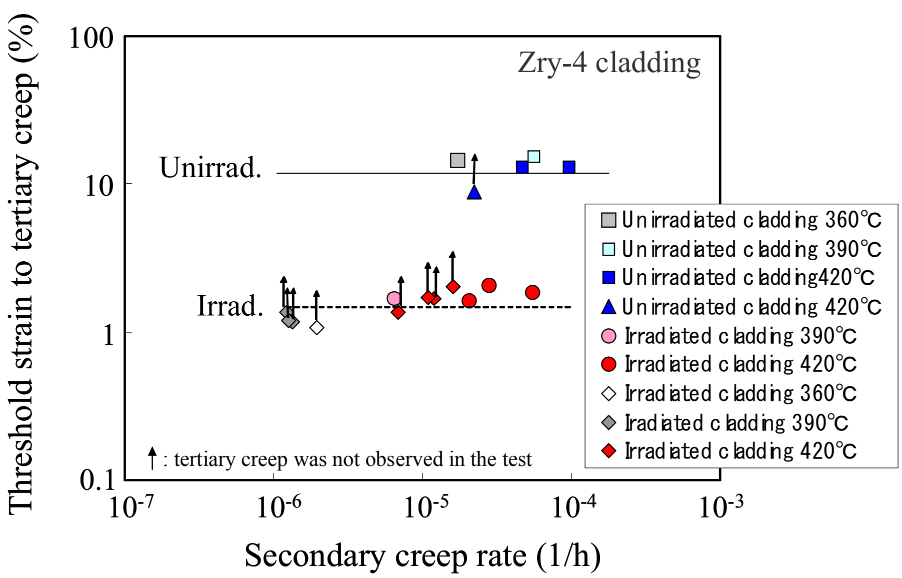

FIG. 6. Relation between threshold strain to tertiary creep and secondary creep rate (Zry4).

3.2. **Hydride reorientation test**

It is well known that the dissolved hydrogen precipitate as hydride along radial direction under the condition in which somewhat large circumferential stress is applied during the precipitation stage and the more radial hydride lead to less circumferential ductility of the cladding. Therefore the radial hydride reorientation during the spent fuel cooling stage should be considered as one of the cladding degradation mechanism in dry storage.

Zry-2 cladding from the BWR spent fuel (40 GWd/t, 50 GWd/t and 55 GWd/t type) and Zry-4 cladding from the PWR spent fuel (39 GWd/t and 48 GWd/t type) are the main test materials. In addition, MDA and ZIRLO claddings (55 GWd/t type) prepared from PWR spent fuel also were tested. Temperature (473–673 K for Zry-2, 523–613K for Zry-4), hoop stress (16–100 MPa for Zry-2, 85–130 MPa for Zry-4) and cooling rate (0.6–30 K/h) of cladding specimens are the test parameters.

In hydride reorientation test, hydride reorientation treatment (HRT) was carried out. The biaxial stress in cladding tube specimens was applied by internal pressure of Ar gas. The specimen temperature was held at the HRT solution temperature in the furnace to dissolve the hydrogen for 30 (for PWR cladding) or 60 (for BWR cladding) minutes, and then decreased to around room temperature to precipitate the hydride. The morphology of hydrides, including the orientation, was evaluated from metallography after HRT.

The mechanical-property change due to radial hydride reorientation was evaluated by the ring compression test at room temperature. In the ring compression test, ring specimens with 8mm in length were prepared from the cladding tube after HRT. Ring specimens were compressed in the radial direction on the flat plane with a crosshead speed of about 2mm/min at room temperature. The test for as-irradiated specimen was also carried out as the reference.

The results of ring compression tests were evaluated in terms of the crosshead displacement ratio. Figures 7 and 8 summarize the results for BWR Zry-2 cladding with a Zr liner and for 48 GWd/t type Zry-4 cladding, respectively. The crosshead displacement ratio for the BWR Zry-2 did not show the particular hoop stress dependence for the HRT 300°C and 250°C specimens, although some degree of reorientation was observed for HRT 300ºC, 70–100 MPa specimens. On the other hand, the crosshead displacement ratio increased for HRT 400°C, 0 MPa (only heat treatment with no stress) specimens compared to as-irradiated specimens, and it decreased with the increase of hoop stress in HRT, or the amount of radial hydride. The crosshead displacement ratio for the PWR Zry-4 was almost the same level for the specimens with HRT conditions of 250ºC/100 MPa and 275ºC/100 MPa compared to as-irradiated specimens. The specimens with HRT conditions of 340ºC/100 MPa and 300ºC/100 MPa showed lower crosshead displacement ratio compared to as-irradiated specimens. .

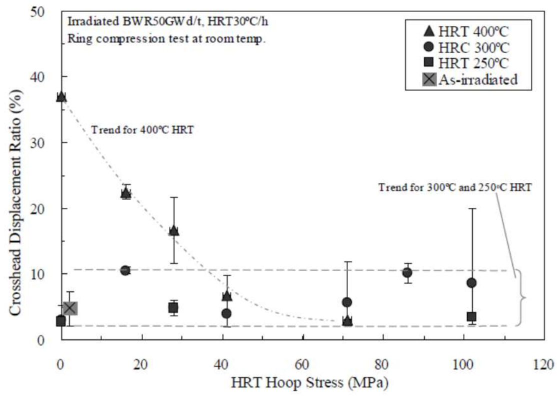

FIG. 7. Correlation between the degree of reorientation and the HRT conditions for irradiated BWR Zry-2 cladding with Zr liner.

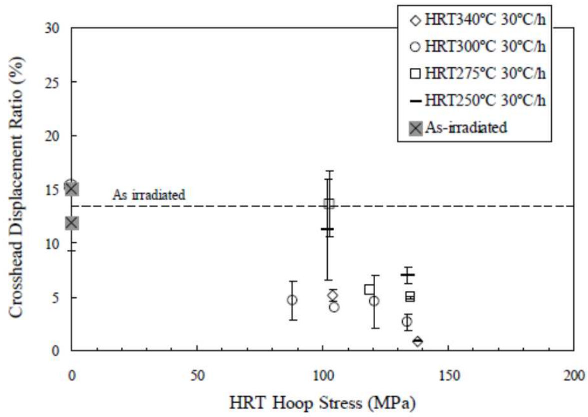

FIG. 8. Correlation between the degree of reorientation and the HRT conditions for irradiated PWR Zry-4 cladding.

3.3. Recovery of irradiation hardening

In recovery test, isothermal annealing of irradiated cladding tube was performed, and then hardness of annealed material was measured [4]. Annealing temperature conditions were from 270–420°C and annealing duration is at a maximum about 10,000 h.

The correlation of annealing conditions (time, temperature) and micro Vickers hardness of irradiated Zry-2 and Zry-4 cladding is shown in Figure 9. No recovery in hardness was observed for PWR 48GWd/t type Zry-4 cladding below 300°C. Although it is reported that below 305°C the irradiation-hardening is not annealed [8], BWR 50GWd/t type Zry-2 shows a slight decrease in hardness after 5000 h annealing at 300°C. This implies that, it is necessary to consider the recovery of irradiation-hardening to evaluate the cladding strength under dry storage, if applicable.

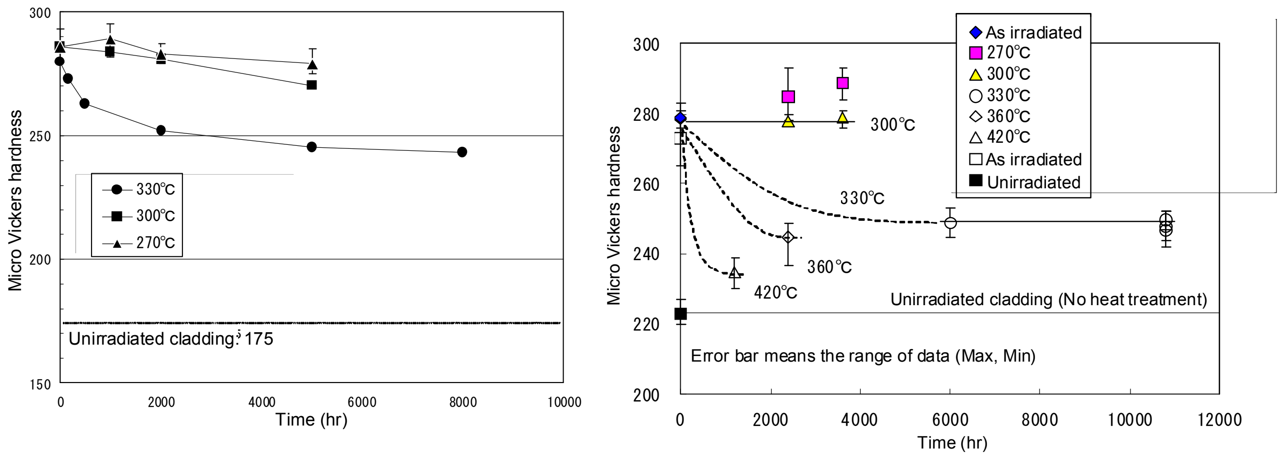

FIG. 9. Irradiation-hardening recovery property.

4. DEVELOPMENT OF SAFETY ANALYTICAL METHOD

In a license application, the applicant shall demonstrate the safety of the facility by considering the long term behavior of casks in the storage building. In the safety review process, the regulatory authority evaluates their results in order to confirm that people around the facility are well protected against any postulated radiation hazard. In this process, upon request from the authority, JNES perform independent analysis and check the adequacy of the safety analysis conducted by licensees. JNES has continuously prepared computer codes to analyze the fundamental safety function of the cask and seismic integrity and radiation shielding performance of the storage building. Table 3 lists computer codes prepared for the audit calculation of spent fuel interim storage. If computer codes applicable to the audit calculation existed, JNES usually verified them by using data obtained from our experiments or available existing experimental ones. In the heat removal analysis in the storage building, radiation heat transfer from the cask is important mechanism. JNES has developed S-FOKS to simulate the radiation heat transfer more. The calculated results are compared with theoretical value and the adequacy was confirmed for relatively simple geometry [9]. Figure 10 shows temperature profile for postulated storage building calculated by FLUENT coupling with SFOKS code. Casks are cooled by natural flow air and radiation. Radiation heat transfer from casks is important heat removal mechanism and could reduce the cask surface temperature by 20°C.

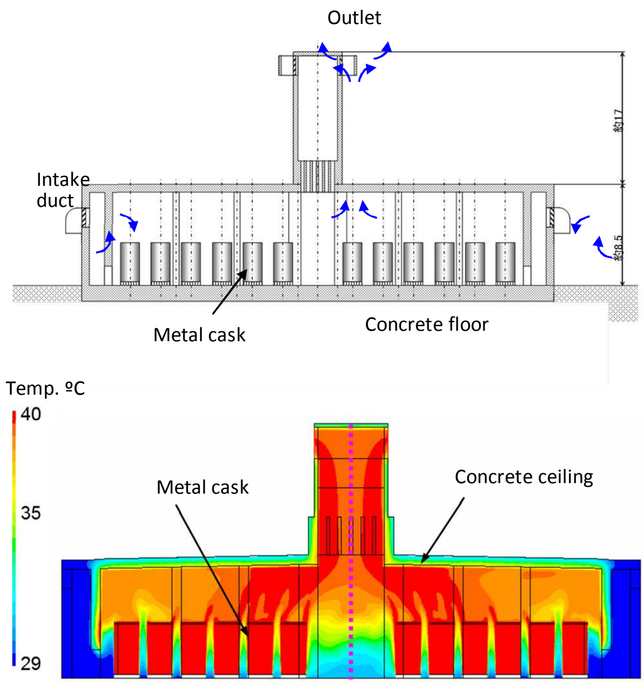

FIG. 10. Temperature profile calculated by FLUENT.

TABLE 3. COMPUTER CODES PREPARED FOR INDEPENDENT ANALYSIS

	Storage building	Cask
Thermal analysis	FLUENT: Fluid dynamics S-FOKS: Heat radiation analysis	FLUENT: Fluid dynamics
Shielding analysis	MCNP5: Monte Carlo Calculation for neutron and photon transport	MCNP5: Monte Calro Calculation for neutron and photon transport
Criticality analysis	MVP-II: Monte Carlo Calculation for neutron transport JENDL-3.3: Japanese Evaluated Nuclear Data Library	MVP-II: Monte Carlo Calculation for neutron transport JENDL-3.3: Japanese Evaluated Nuclear Data Library
Structural analysis		LS-DYNA: Large deformation

5. FUTURE REGULATORY SUPPORT ACTIVATES

After the license application is approved by the regulatory authority, the applicant shall submit application of design and construction method. Technical standard of the design and construction method is stipulated in the ministry ordinance. As described in Sections 2 and 3, some of them were used for the safety review. Furthermore, a part of those results are referred in an academic standard which will be soon published by Atomic Energy Society of Japan (AESJ) [10]. JNES is now supporting NISA's activity on preparing the technical criteria for the design and construction approval and the AESJ standard will be quoted in the technical criteria.

JNES carries out welding inspection of the nuclear facilities by law. To prepare welding inspection procedure for the metal cask, JNES support NISA’s activity to technically endorse another academic standard issued by Japan Society of Mechanical Engineers (JSME) [11].

In Japan, interim storage facilities will be built at coastal area and corrosion induced by salinity atmosphere might be one of threats for long term integrity of canister for concrete type cask. Therefore, corrosion resistance stainless steel is considered as material of canister. Corrosion resistance of the stainless steel was investigated by CRIEPI [1]. On the other hand, JNES performed studies to investigate welding procedure of such stainless steel and inspection method of the welding In Japan, the following stainless steel are nominated as canister material, ASME SA-240 S31260 and ASME SA240 S31254. The former is an enhanced corrosion resistant duplex stainless steel consisting of austenite/ferrite. The material contains extremely low C, high Cr, and high Mo and has excellent resistance against local corrosion attack (i.e., pitting corrosion, crevice corrosion, SCC, etc.), as well as excellent acid resistance against various acids. The latter is a super stainless steel with improved chloride corrosion resistance and acid resistance by adding N and Cu into high Cr – high Mo steel. Mechanical properties of them are not listed in the existing academic code [11]. Therefore, JNES conducted the measurement of main mechanical properties. Figure 11 shows comparison between proof stress and ASME Sy value of ASME SA240 S31260. ASME SA240 S31260. The result confirmed that mechanical properties complied with the criteria specified in JSME code [11].

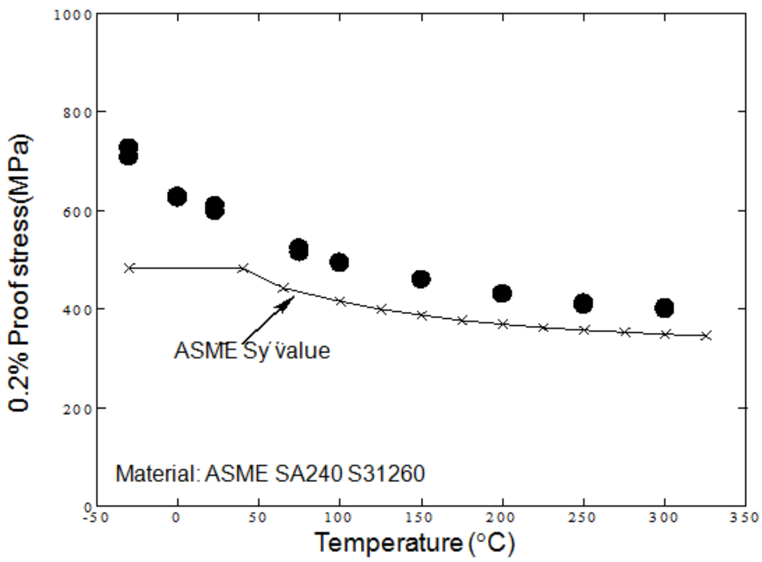

FIG. 11. Comparison between 0.2% Proof Stress and ASME Sy Value of ASME SA240 S31260.

In the Japanese regulatory, both multi-layer PT and UT inspections are required for welding of canister. JNES is now investigating applicability of those inspection methods for above stainless steels.

About the long term integrity of the metal cask and internals, the Nuclear Safety Commission requests that, the utilities and the regulatory authorities are corresponded as follows. From a viewpoint of the integrity confirmation of the cask and internals in the transport after the storage, the utilities shall continually carry out the conditions survey of the dry storage and shall plan the accumulation of knowledge regarding long term integrity of the cask and internals. The regulatory authorities shall determine the rational inspection method of the cask and internals for transport after the storage considering the long term storage. The utilities plans the demonstration test program for long term storage of PWR spent fuel. NISA and JNES will join to the program of the utilities in order to obtain the knowledge regarding spent fuels during the long term storage.

6. CONCLUSION

Since the establishment of JNES, JNES has continuously performed support activities for the safety regulation of spent fuel interim storage. Before the first license application, activities mainly focused on technical information for evaluating the fundamental safety function of metal casks, integrity of spent fuel cladding and on preparing the safety evaluation tools. The safety review by NISA was completed 22 December 2009 and the review report was submitted to NSC and Atomic Energy Agency for comment, respectively. In the safety review, JNES conducted the independent analysis. After the approval of license application, the design and construction method should be approved by NISA. At this, NISA is preparing the criteria of the approval and JNES is supporting it. JNES is also accumulating technical basis to make welding inspection methodology. And JNES is continuously accumulating technical basis concerning the integrity of spent fuels in the future.

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LICENSING PROCEDURES FOR INTERIM STORAGE OF SPENT FUEL IN GERMANY

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Abstract

In accordance with the waste management concept in Germany spent fuel is stored in interim storage facilities for 40 years until disposal in a geological repository. The storage concept bases on dry storage of the spent fuel in metallic transport and storage casks, standing upright in halls of reinforced concrete. Storage of spent fuel as well as significant modifications of the storage require a license according to art. 6 of the Atomic Energy Act. The Federal Office for Radiation Protection (Bundesamt für Strahlenschutz - BfS) is the competent licensing authority. The mode of the licensing procedure — whether formalized or non-formalized — depends on the necessity to carry out an environmental impact assessment. Formalized licensing procedures include a public participation procedure. In the following, the licensing prodecures are illustrated and a short overview over the current licensing procedures conducted by BfS is given.

1. INTRODUCTION

In accordance with the waste management concept in Germany, which bases on the amendment of the Atomic Energy Act in April 2002, spent nuclear fuel from nuclear power reactors and research reactors has to be directly disposed in an geological repository. Reprocessing of spent fuel is no longer a legal option of radioactive waste management. Pursuant to Art. 9a of the Atomic Energy Act [1] the delivery of spent nuclear fuel elements to reprocessing plants was prohibited by 1st July 2005. Thus, until deposition in a geological repository, spent fuel is stored in interim storage facilities.

In twelve on-site interim storage facilities in the vicinity or directly on the sites of the nuclear power plants, spent fuel elements from operation of the 17 reactors are stored after the necessary period of decay in wet storage basins inside the reactors. The licensed capacities cover from 450–1850 Mg of heavy metal (80–192 cask positions). One more interim storage facility is planned on the site of the NPP Obrigheim, which has already gone out of operation in 2005. The residual spent fuel elements are recently stored in a wet storage facilty that is planned to be replaced by a dry storage facility. Another on-site interim storage facility is operated for storage of the spent nuclear fuel elements from the test reactor in Jülich.

Additionally, three central interim storage facilities are in operation. They are used for storage of spent fuel of different origin, e.g. from research and test reactors. In the central storage facility of Gorleben, vitrified high active waste still originating from the reprocessing of spent fuel elements from German nuclear power reactors is stored; yet from the reprocessing plant in La Hague, France. The licensed capacities of the central storage facilities cover from 585 Mg up to 3960 Mg of heavy metals. Except for one facility, all storage licenses were granted for a storage period of 40 years.

2. CONCEPT OF DRY INTERIM STORAGE OF SPENT NUCLEAR FUEL

The German facilities are built complying with the concept of dry interim storage in metallic transport and storage casks. The confinement of the radioative material is ensured by the double lid system of the casks, of which the leaktightness is monitored constantly. The construction of the casks allows adequate heat removal and shielding of gamma and neutron radiation. Usually, the casks are stored in halls of thick concrete structures, which serve as further radiation shielding barrier. Sufficient decay heat dissipation is achieved by natural convection: cool outside air is flowing into the hall through openings in the side walls, warmed up on the surface of the casks and exhausting through openings in the roof. Safe maintenance of subcriticality is achieved especially by specified geometric configurations of the radioactive inventory in the fuel basket and the use of neutron absorbers. The storage building concept for the on-site interim storage facilities exists in two technical variants as one-nave or two-nave buildings. For the on-site interim storage facility of the Neckarwestheim NPP, a special solution below ground was developed due to a site-specific narrow location in a quarry. The storage facility consists of two tunnel tubes lined with concrete. The cooling of the casks is also effected by natural convection via exhaust air chimney.

3. LEGAL BASIS FOR INTERIM STORAGE OF SPENT NUCLEAR FUEL

3.1. Licensing requirements

Legal basis for the interim storage of spent nuclear fuel is article 6 of the Atomic Energy Act [1]. Thus, storage of nuclear fuel requires a license. As well, a license is necessary for any significant modification of the licensed storage. Additionally, article 6 specifies the licensing preconditions for the storage of nuclear fuel, as there are

- Reliability and technical qualification of the applicant and the persons responsible for leading and managing the storage;
- Precaution against harm and damages caused by the storage according to the state of the art in science and technology;
- Sufficient financial security to cover the liability for damages caused by the storage, and
- Protection against disruptive actions or other third-party intervention.

Pursuant to article 6 of the Atomic Energy Act, the license must be granted if the licensing requirements are met.

Several special ordinances as well as sub-regulatory rules and standards supplement the regulations of the Atomic Energy Act:

The Radiation Protection Ordinance [2] is the central safety standard in the sphere of the Atomic Energy Act, specifying the principles and requirements of radiation protection including admissible limits for exposure to ionizing radiation. Further ordinances specify the preconditions to ascertain reliability of the responsible persons and the stipulated financial liability for damages.

The “Safety Guidelines for Dry Interim Storage of Irradiated Fuel Assemblies in Storage Casks” of the Reactor Safety Commission [3] specify the fundamental protection goals for dry storage of spent nuclear fuel and resume all important safety requirements concerning design and operation of the facility.

3.2. Roles and responsibilities

Pursuant to article 23 of the Atomic Energy Act, the Federal Office for Radiation Protection (Bundesamt für Strahlenschutz - BfS) is the competent licensing authority for interim storage of nuclear fuel according to article 6 of the Atomic Energy Act. BfS is a scientific-technical authority within in the area of competency of the Federal Ministry for the Environment, Nature Conservation and Nuclear Saftey (Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit (BMU)).

Besides the license for storage of nuclear fuel according to the Atomic Energy Act, the construction of the storage facility requires a planning and building permission. This permission is granted by the local building authority, in accordance with the legislation of the Federal State in which the facility is located.

Once granted a license by BfS, the competence for supervising the operation of the facility is covered by the Federal State in which the facility is located (art. 24 of the Atomic Energy Act). In many cases, the supervisory authorities are the State Ministries of environmental, economic, social or inner affairs.

According to article 20 of the Atomic Energy Act, the competent authorities can assign external experts to support their licensing or supervisory activities. Thus, BfS often assigns for example the Federal Institute for Materials Research and Testig (BAM) or the Institute for Material Testing (MPA) of the university Stuttgart and the technical inspection agencies TÜV Nord or TÜV Süd with the verification of cask, inventory and storage safety as well as radiological aspects.

4. LICENSING PROCEDURE

The mode of the licensing procedure depends on the necessity to carry out an environmental impact assessment within the licensing procedure. An environmental impact assessment requires a formalized licensing procedure in accordance with the regulations of the Nuclear Licensing Procedure Ordinance (AtVfV, [4]) and the Environmental Impact Assessment Act (EIA Act – UVPG, [5]). Significant element of this formalized procedure is the involvement of the public as well as the description and evaluation of environmental impacts caused by the storage facility, including construction, operation and shut-down. If an environmental impact assessment is not needed to be carried out, a non-formalized licensing procedure in accordance with the general administrative procedural rules takes place. That includes the involvement of affected authorities and a hearing of the applicant.

4.1. Establishment of a new storage facility

According to the Environmental Impact Assessment Act (EIA Act), the establishment of a new interim storage facility for spent fuel demands an environmental impact assessment. Consequently, for the storage license as well as for the building permission, formalized licensing procedures have to be carried out. Besides the profound examinations concerning the compliance of the applied project with the legal pre-conditions for storage of nuclear fuel on the base of detailed application documents, the formalized licensing procedure includes a public participation and a comprehensive assessment of the environmental impacts induced by the project.

The parts of the licensing procedures concerning the environmental impact assessment, especially the involvement of the public, are conducted by BfS in cooperation with the appropriate building authority. The public participation includes some essential and strictly formalized steps, which serve to identify and consider the concerns of vicinity and public due to the applied storage and carefully preserve the rights of all affected parties:

(a) The application has to be announced in the federal bulletin and local daily papers. The announcement includes the basic information on the project (applicant, location of the planned project) and indicates the period and locations of the public presentation of the basic application documents. The public gets informed about the opportunity and terms to raise objections and the legal consequences of the expiry of the term;

(b) The basic application documents are disclosed for public inspection for a period of two months at BfS and a public location near the site of the planned facility. The documents include the letter of application, a safety report, a summary of the project, the environmental impact study, a landscape conservation support plan and, if already available, further decisive documents;

(c) Within the two month presentation period, everbody has the opportunity to raise objections concerning the project towards the licensing authority. On the other hand, by expiry of this term, any further objections are excluded and will not be taken into account within the final decision about the application;

(d) At least one month after the disclosure of the documents a hearing of all parties having raised objections takes place. The raised objections are discussed with the objectors an the applicant. Usually, external experts, who are assigned by the licensing authority, and other affected authorities take part in the hearing. The discussion gets precisely documented in a transcript of the hearing, which is, when finished, available for all participants;

(e) The final decision about the application is announced in the federal bulletin and local daily papers. If the license is granted, the notice of approval is disclosed for a period of two weeks. Besides the regulatory part, the notice of approval contains a detailed justification of the licensing decision, including a summarization and evaluation as well of the raised objections as of the environmental impacts induced by the project. As appropriate legal redress, a lawsuit against the licensing decision can be abandonned towards the upper administrative tribunal within one month after the end of the disclosure period.

The formalized licensing procedure includes, as far as affected, the information of the public also in neighbour states. Thus, the citizens of the neighbour states are given the opportunity to raise objections concerning the project, associated with the same legal consequences as for German citizens.

4.2. Modifications of a storage facility

In cases of minor significance for the storage concept, modifications of existing interim storage facilities and their operation can be approved by the competent supervisory authority. However, if modifications are classified as significant, they require an amending license according to article 6 of the Atomic Energy Act, granted by BfS.

Within licensing procedures for storage modifications, a preliminary assessment according to the EIA Act (case-by-case examination) evaluates the relevance of supplementary environmental impacts induced by the modification. The result decides about the necessity to conduct a further environmental impact assessment and subsequently about the appropriate licensing procedure. If the modification is not associated with relevant environmental impacts and, consequently, does not require a further environmental impact assessment, a non-formalized licensing procedure is carried out. In this case, the result of the preliminary assessment is announced to public, but further steps of public involvement, as described above, are not performed.

The non-formalized licensing procedure is structured by the general administrative procedural rules. BfS, usually assigning independent experts, profoundly verifies the compliance of the applied project with the legal preconditions for storage of nuclear fuel on the base of detailed application documents submitted by the applicant. Affected authorities and the applicant are steadily involved. In the final phase of the licensing procedure they are given the opportunity to formally express their issues about the intended licensing decision.

In opposite to the formalized licensing procedure, the final decision about the application is not announced to public and the notice of approval is not disclosed. As appropriate legal redress, an objection against the licensing decision can be filed to BfS.

5. CURRENT LICENSING PROCEDURES

The current licensing procedures conducted by BfS are related to the establishment of one new storage facility and additionally to several intended modifications of storage in existing facilities.

5.1. New storage facility on the site of the Obrigheim NPP

On the site of the Obrigheim NPP, that finished operation in 2005, a new interim storage facility is planned for dry storage of the residual spent fuel elements in transport- and storage casks. The fuel elements are recently stored in a wet storage facility closely associated with the NPP. It is intended to be taken out of operation within in the dismantling of the NPP. The dry storage facility is projected for a capacitiy of 100 Mg heavy metals in 15 casks standing upright in a hall of reinforced concrete.

Initially, in April 2005 an application according to art. 6 of the Atomic Energy Act concerning the establishment of the new interim storage facility was filed to BfS. After some modifications of the concept in october 2007, the public participation procedure was started. In April 2008, the application of the project was announced in the local daily papers and the federal bulletin.

The public presentation of the basic application documents took place from May 8th until July 7th, 2008, in the Town Hall of Obrigheim and in the BfS in Salzgitter. Within the presentation period, 897 people raised objections against the applied facility. As a lot of people signed class-action objections, altogether 20 objections of different content were raised towards BfS. They mainly concerned the storage concept, the cask safety, aspects of radiation protection and hazardous incidents as well as formal aspects of the participation procedure and general statements against nuclear energy use. Preparing the complete evaluation and discussion of the objections, their contents were classified in an agenda.

The hearing of the objectors was announced in September 2008 and took place from October 8th until 10 th, 2008, in a congress centre in Mosbach, a town near Obrigheim. About 25 objectors, the applicants, BfS as licensing authority, independent experts assigned by BfS, the building authority, the supervisory authority and some affected local authorities took part in the hearing. On the basis of the agenda, all objections were intensively discussed. In January 2009, the transcript of the hearing was completed and sent to the objectors, the authorities and the experts having taken part in the hearing.

In the further progress of the licensing procedure, the application documents, demonstrating the compliance of the intended storage concept with the safety requirements, are completed by the applicants. On this basis, BfS and the assigned external experts will take up their verifications, covering all fields of the licensing requirements according to art. 6 of the Atomic Energy Act. This includes, for example, the verification of cask safety, inventory and criticality safety, storage safety including heat dissipation, radiation protection, technical eqipments and operation, operational and external incidents, seismologic aspects, environmental impacts and physical protection of the storage facility as well as reliability and technical qualification of the responsible persons and financial liability.

5.2. Amending and supplementary licensing procedures

The licensing procedures concerning already existing facilities differ depending on the subject of the application.

Amending licenses concern modifications of the licensed storage which were applied for after the basic license had been granted. Within the according licensing procedures, preliminary environmental impact assessments according to the EIA Act are conducted in order to decide about the necessity of a further environmental impact assessment including a public participation.

Supplementary licenses, in opposite, decide about matters, which were already part of the initial storage application. This is due to the fact, that BfS at first granted partial licenses to those parts of the applications whose examination had been concluded. Up to now, BfS continues the examinations concerning some parts of the former applications within in the scope of supplementary licensing procedures. They slightly differ from the amending licensing procedures, as the initial environmental impact assessment, which was conducted within the frame of the basic license, included the matters of the supplementary licenses as well. However, possible environmental impacts of these matters are newly regarded form the present point of view within the present procedure.

The licensing procedures carried out presently concern, for example, further types of transport- and storage casks, new or modified casks, a modified inventory, further types of fuel elements or the additional storage of nuclear waste of another origin within the storage facility for spent fuel. In one case, the extension of the storage license for some years has been applied for.

The matters of supplementary and amending licenses granted up to now did not require an environmental impact assessment. In most of the current amending or supplementary licensing procedures, a decision about the environmental impact assessment has not yet been taken.

The duration of a licensing procedure always significantly depends on the application documents submitted by the applicant. They have to completely describe the amendment and its influence on the licensed storage and provide evidence, that all safety requirements are met according to the state of the art in science and technology. On the base of such completed application documents the examinations by BfS and the external experts can be carried out. Depending on the matter of the amendment, the verifications regularly affect the field of casks safety, inventory, criticality safety, technical eqipments and operation, operational incidents and, of course, aspects of radiation protection referred to the operating personnel and the site. The examinations concluded and the decision taken, a final draft of the notice of approval is sent to the affected authorities and thereafter to the applicant in order to hear and consider their issues about the decision.

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SAFETY AND LICENSING OF SPENT FUEL STORAGE FACILITIES

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Abstract

All operating nuclear power reactors in the United States (U.S.) are storing spent fuel on-site in spent fuel pools licensed by the U.S. Nuclear Regulatory Commission (NRC). Spent fuel pools at U.S. reactors were not designed to store the full amount of spent fuel generated during the lifetime of plant operation. Consequently, most utilities expanded their storage capacity by the use of high-density storage racks. Even with the high-density racks, most utilities will need additional storage capacity. When it became apparent that nuclear power plants were going to need additional spent fuel storage space, the NRC amended its regulations in 1980 to provide nuclear power plants with alternate spent fuel storage in an independent spent fuel storage installation (ISFSI). NRC provides for a 20-year specific license with the option to renew the license for additional 20-year terms. In 1990, the NRC implemented the General License option to ease the burden on nuclear power plants that have a license to either operate or possess fuel. The general license for each storage cask terminates 20 years after the storage cask is first used by the licensee. The first storage cask using a general license was loaded in 1994. This paper discusses NRC experiences and its knowledge gained in licensing over the past 30 years and renewing the licenses for three ISFSIs and how this knowledge has driven the NRC to revise its guidance and thought processes for dry storage.

1. INTRODUCTION

In July 1986, the NRC issued the first site-specific license to the Surry Nuclear Power Plant located in Virginia authorizing interim storage of spent fuel in dry storage casks. After Surry, NRC licensed H.B Robinson and Oconee nuclear power plants both of which are located in South Carolina to store spent fuel in 1986 and 1990, respectively. Subsequently, the NRC issued specific licenses to 11 additional sites for dry storage of spent fuel. In total these 14 licensees are storing over 300 casks. In addition to the specific licenses, there are 40 generally licensed sites currently storing almost 900 storage casks across the U.S.

In addition to this U.S. experience, there are 17 countries currently storing spent fuel at 50 dry storage facilities [1], with additional Member States either constructing or planning on utilizing dry storage facilities. There has been a wealth of knowledge gained operating dry storage facilities over the past 30 years. Additionally, computers, codes (whether structural, thermal, or nuclear) have taken quantum leaps since the first storage cask was licensed.

While dry storage of spent fuel is conducted safely worldwide, the nuclear community needs to keep constant vigilance to ensure that complacency is not encountered. Granted that dry storage has many passive systems that do not require constant attention like nuclear power plants, and fuel movement is done frequently at many nuclear power plants, human error can quickly lead to exceeding the storage design basis. With much of the world taking a “wait and see” approach to the back end of the fuel cycle, dry storage is likely to be a staple of the nuclear power community for a long time. International cooperation in numerous areas relative to spent fuel storage and transportation will be more important in the years to come.

2. DRY STORAGE EXPERIENCE

In January 1997, after having gained 10 years' experience in licensing ISFSIs and reviewing dry storage facilities, the NRC issued NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems" [2] as guidance to NRC staff on how to perform a review of a spent fuel storage cask application for a certificate of compliance. Subsequently, in March 2000, NRC issued NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities" [3] to provide staff guidance on licensing ISFSIs. These standard review plans coalesced the knowledge staff had gathered reviewing dry storage casks and facilities.

While there have not been any significant events in spent fuel storage, there have been a number of areas where a great deal has been learned over the years and NRC revised its guidance to reflect these experiences. As Wolfgang Bernhard, a member of the Board of Management of Daimler AG said, "This is just the first step on a long way. We cannot lean back and be complacent". This fits well with spent fuel storage, given that there is only 25 years of experience with dry storage of spent fuel with no significant events. While most people think about storage being when the fuel is physically in a dry storage system, the most likely time for errors during the storage process is during loading and preparations for storage, which may include draining, drying and welding the storage canisters.

Human error during fuel movement and preparation for storage is when extra care is needed to ensure that we are not complacent and always striving for improvement. There have been numerous examples of human error, none of which were significant, with respect to dose to workers or the public, to remind us that vigilance is paramount in the nuclear community by both licensees and regulators. While there have been examples in countries that utilize dry storage, there have been some that occurred in the U.S. of sufficient significance to effect changes to NRC guidance.

2.1. Chemical interactions with borated water

Two instances of chemical interaction with a storage basket occurred at U.S. nuclear power plants, one of which resulted in hydrogen gas generation and ignition and the other turbidity in the spent fuel pool. While neither of these events occurred recently, both have left a lasting impression on the reviews performed at the NRC.

2.1.1. Point beach nuclear power plant hydrogen ignition

In May 1996 when welding the shield lid onto a Model No. VSC-24 inner canister, hydrogen gas ignited leaving the 3000 kg shield lid tipped at a slight angle, with one edge 7.6 centimetres higher than normal. Approximately 114 litres of borated pool water had been drained from the canister to facilitate welding of the shield lid, creating an air space below the lid. On removal of the shield lid, the licensee observed a white, foam-like substance under the shield lid. Most of this substance floated to the top of the pool; however, some remained below the surface of the water. Samples of this substance were taken to Argonne National Laboratory for analysis. The licensee investigation determined that the source of hydrogen was oxidation of zinc in the basket coating (used to prevent corrosion) of the basket when in contact with the borated water in the spent fuel pool. The foam-like substance was determined to be a mixture of a precipitate formed by the reaction with residual soap (used during decontamination of the basket), boric acid and air.

2.1.2. Trojan nuclear power plant pool turbidity

After the Point Beach hydrogen ignition event in 2006, NRC was sensitized to reviewing material interactions and to ensure that no chemical or galvanic reactions would occur. In summer 1999, NRC approved a new coating to be used on the VSC-24 storage cask at the Trojan Nuclear Power Plant. The coating was used to minimize corrosion of the carbon steel basket material. The licensee chose a coating made by TranStorTM for the basket internals which would minimize hydrogen generation. The coating required high temperature curing, but the fabrication sequence resulted in areas that were uncoated. On the first basket, the uncoated areas were touched up, but the coating was not properly cured and a second basket contained uncoated areas. When the first two baskets were undergoing pre-immersion washing with demineralized water, the resulting water runoff developed a yellow tint, which was later determined to be chromate leaching.

The licensee went forward with cask loading and approximately 24 hours after initiation of spent fuel transfer operations turbidity was observed in the spent fuel pool, to such a degree that the bottom of the pool was not visible. Several problems were associated with this effort, including the carbon steel being exposed directly to boric acid due to:

- Intentionally uncoated areas;
- Protrusion of substrate through coating;
- Improperly coated subassembly parts;
- Removal of improperly applied coating by washing;
- Degradation of coating due to demineralized water, and
- Coating not fully cured.

2.1.3. Lessons learned

The NRC, vendors and licensees learned several lessons during the course of these two events including ensuring that technical experts, including the expertise of stakeholders, are thoroughly reviewed and evaluated; and recognize and control special processes and materials that are used for the first time.

NRC has found that it is vitally important when preparing or reviewing a safety analysis report to question assumptions, especially those that come from a vendor or technical expert that may have a stake in whether the material or process is approved. Share information and think broadly. Discuss concerns with other Member States, as needed and ensure that materials and processes that are a “first of a kind” are thoroughly evaluated and tested to minimize unexpected results. While some proprietary information and data are submitted for spent fuel storage casks and transportation packages, consider reviewing information from other licensees on their experiences for a similar material or process. As technical information and codes are improved, NRC looks to use the most current information possible.

Based in part on chemical interactions that occurred at Palisades and Trojan, the NRC recognized the need for specific guidance for the NRC staff for its review of materials selected by the applicant for storage cask and transportation package designs to ensure quality and uniformity in reviews performed by new or current NRC staff. The NRC developed Interim Staff Guidance No. 15 (ISG-15) [4] to fill the void in materials review procedures prior to the next revision of the standard review plans.

2.2. Light-weight transfer cask at Fort Calhoun nuclear power plant

A storage cask vendor prepared an evaluation for the licensee to add a light-weight transfer cask as an approved modification to the generally-licensed Standardized NUHOMS® Horizontal Modular Storage System, without prior NRC approval. The light-weight transfer cask was necessary since the maximum authorized weight the facility crane could lift was 68 tonnes instead of 110 tonnes for a typical nuclear power plant crane. Instead of upgrading the crane, the licensee contracted with the vendor to evaluate and provide justification for using the light-weight transfer cask.

NRC inspectors identified several issues associated with the vendor's evaluation during the pre-operational inspection at the licensee's site. For example, prior to the welding process, vendor's evaluation suggested draining the majority of the water from the canister. After welding is completed, the remaining water in the canister is to be removed prior to commencement of vacuum drying. The design-basis thermal evaluation in the safety analysis report was based on most of the water remaining in the canister during welding. NRC also questioned the vendor's evaluation of whether the light-weight transfer cask could meet the technical specifications for the transfer cask dose rates and the thermal analysis of the light-weight transfer cask on the transfer trailer with the additional shielding.

The time limit established for the vacuum drying technical specification was selected to ensure that the maximum cladding temperature would meet the limit of 400°C during vacuum drying and to ensure that the cladding temperature met the thermal cycling criteria during drying, helium backfilling, and transfer operations. The transient temperature in the thermal analysis is based on the initial canister and fuel temperature prior to vacuum drying. Changing the sequence of operations to drain most of the water from the transfer cask prior to welding would yield a higher fuel cladding temperature at the start of vacuum drying than calculated in the safety analysis report. A higher initial temperature would result in a shorter vacuum drying time than that specified in the technical specifications.

Ultimately, the licensee requested NRC grant an exemption to the NRC regulations with the following four conditions: 1) limited to loading four canisters, 2) limited the decay heat level per canister to no more than 11 kilowatts, 3) limited the cooling time of the fuel to a minimum of 16.2 years, and 4) substituted existing technical specification dose rate limits with new limits based on the specified condition of the supplemental shielding. These conditions helped to alleviate the staff's concerns regarding fuel temperatures and high dose rates potentially experienced during fuel transfer activities.

While the licensee accepted the vendor's evaluation, the licensee is held accountable for any operations and changes made at the site (even if the work is contracted out). The licensee needs to ensure that changes that alter the sequence of operations do not affect either the technical specifications or any design-basis calculations. As previously noted, the vendor's evaluation involved a change in the sequence of operations that drained the bulk of the water from the canister earlier in the process than was prescribed in the safety case.

As a result of the NRC's experience with the light-weight transfer cask at Fort Calhoun, NRC is developing an ISG title "Additional Considerations in the Shielding and Radiation Protection Reviews of Spent Fuel Dry Storage Systems" to provide guidance to the staff on establishing dose rate technical specifications and discusses a graded approach to determine the overall level of review effort for an application for a spent fuel dry storage system certificate of compliance based on risk-informed considerations. Additionally, the NRC is also in the final stages of revising its standard review plan for dry cask storage systems to develop a risk-informed approach to reviews for all other areas of review.

2.3. Damage fuel evaluation at Humboldt Bay

NRC regulations require that the spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be confined such that degradation of the fuel during storage will not pose operational safety problems. This may be accomplished by canning of assemblies or consolidated fuel rods. Additionally, the fuel must be retrievable after normal storage conditions.

The Humboldt Bay technical specifications in their original license defined damaged fuel assemblies as those with known or suspected cladding defects, greater than pinhole leaks or hairline cracks as determined by a review of records. Further their technical specifications stated that damaged fuel assemblies must be stored in a “damaged fuel container”. Intact fuel assemblies are defined as those assemblies that are not damaged. In a subsequent addendum to the Humboldt Bay ISFSI safety analysis report, the licensee reiterated the definition of intact and damaged fuel given earlier but with more qualitative measures of what they consider to be a hairline crack. Their definition was similar to the guidance in the definition of a gross breach given in Interim Staff Guidance No. 1 (ISG-1) Rev 2, “Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function” [5]. This addendum also says that the licensee would examine the fuel in accordance to the staff’s guidance in ISG-1 Rev. 0, “Damaged Fuel” [6].

The guidelines in ISG-1 Rev. 0 were a review of the records to verify that the fuel is undamaged, followed by an external visual examination of the fuel assembly prior to loading for any obvious damage. For assemblies where reactor records are not available, the level of proof will be evaluated on a case-by-case basis. The purpose of this demonstration is to provide reasonable assurance that the fuel is undamaged or that damaged fuel loaded into a storage or transportation cask is canned.

Unfortunately, the licensee had no reactor operating records that would support qualifying fuel as being undamaged. Humboldt Bay conducted four-sided visual examinations on all the assemblies and identified as-damaged those assemblies that had:

- Visible defects on the outer rod surfaces;
- Broken rods;
- Rods on the inner assembly surfaces that deformed the cladding to the extent it could be seen on a perpendicular view of the assembly, or
- Areas where they could see a deposit on the rods;
- Since the reactor records could not be used to show that the assemblies contained no damaged rods, the use solely of an external visual examination was never considered by the staff to be sufficient.

The licensee also sipped the assemblies and indicated that 27 assemblies contained a breached rod. These rods were never found to be damaged via the visual examination. This could be because the sipping indications were due to pinholes that were not visible on the inner surfaces of the assembly or that there were larger breaches that were not detectable by the Humboldt Bay visual method or a combination of both possibilities.

The videos shown by Humboldt Bay clearly indicated that in order to detect a breach on an inner rod surface it had to result in significant rod deformation. NRC does not consider this to be definitive since breaches do not have to result in large rod deformations to be considered damaged fuel. Humboldt Bay could not, with any reasonable probability, say that any of their assemblies were undamaged, since without the ability to visually examine internal rods, one cannot obtain reasonable assurance that there are no damaged rods in the assembly. The logical conclusion would be that all assemblies should be canned, based on the licensee's technical specifications and guidance in ISG-1 Rev, 0.

NRC considered the potential ramifications if damaged assemblies are loaded as undamaged (i.e., not canned). Since the Zircaloy assemblies will be in a non-oxidizing atmosphere at a temperature significantly below 400°C, no change in the condition of the rods is expected during normal or off-normal operations, even if a rod was breached. Assemblies placed in the canister would be retrievable if necessary, and would not pose operational safety problems with respect to its removal from storage.

The licensee proposed using reactor off-gas release rates, sipping results, and visual examinations to determine the status of the assemblies. In principle, the methodology and arguments made by the licensee looked reasonable, although there were a number of uncertainties that needed to be addressed. These additional efforts consisted of the identification, review, and evaluation of plant operating records, including records of fuel sipping and off-gas activity. The application of these additional fuel characterization methods resulted in a net increase in the total number of fuel assemblies classified as damaged, from 96–129, of the 390 assemblies in the spent fuel pool. The licensee concluded that the revised approach, assumptions made, and the conclusions reached are consistent with the Humboldt Bay ISFSI licensing basis.

The NRC reviewed the licensee's information and concluded that the additional analyses of reactor operating records, in conjunction with the prior video examinations performed, constitute a reasonable approach to the classification of intact and damaged fuel at Humboldt Bay. The NRC focused its review on the overall method employed and relied upon application of the method for the analysis and classification of individual fuel assemblies. NRC also noted that acceptance of this method at Humboldt Bay does not imply that the same method and assumptions are necessarily appropriate for application at another site. The general guidance provided in ISG No. 1, Rev. 2 regarding damaged fuel should be considered by other licensees in developing fuel characterization programs for their sites, with respect to dry cask storage activities.

NRC used the lessons learned and information provided during the course of the Humboldt Bay issue to rethink its guidance in ISG-2, "Fuel Retrievability" [6]. Originally guidance from the NRC on retrievability pertained to dual-purpose casks and recognized the canister could easily be transferred from a storage system into a transportation package. With dual-purpose designs, fuel no longer must be returned to the reactor spent fuel pool for repackaging. Further if a facility used for interim storage has a method to repackage the fuel into a transportation package for shipment offsite for further processing or disposal, a facility met the retrievability requirements.

The Humboldt Bay experience gave NRC pause to reconsider retrievability in terms of damaged fuel that may not be in an inner canister and would need to be retrievable. Given the discussions and review associated with the Humboldt Bay's evaluation on whether individual assemblies are damaged or undamaged and the fact that the spent fuel will be stored in a non-oxidizing atmosphere, the NRC revised its guidance in ISG-2 Rev 1 to consider a fuel assembly to be retrievable if it:

- Remains structurally sound (i.e., no gross degradation), and
- Can be handled by normal means (i.e., does not pose operational safety problems during removal), or
- Is in the case of a structurally unsound assembly or an assembly that has rods with breaches greater than a pinhole or a hairline crack that could release fuel particulate, if the assembly is placed inside a secondary container (described in ISG-1 as a "can for damaged fuel") that confines the fuel particulate to a known volume and, that container can be handled by normal means.

NRC also further defined the term "Normal Means" to include the ability to move the fuel assembly and its contents by use of the crane and grapple used to move undamaged assemblies at the point of cask loading. The addition of special tooling or modifications to the assembly to make the assembly suitable for lifting with crane and grapple does not preclude the handling from being considered "normal means". When undamaged fuel is placed in a storage canister, it is retrievable by normal means, i.e. grapple and hook. Those assemblies that cannot meet this criterion should be placed in a can for damaged fuel or otherwise modified, so the assembly or can for damaged fuel can be ready retrievable from the storage canister by normal means.

For removal of spent fuel from storage prior to transport, spent fuel should be retrievable on an assembly basis, in addition to a canister basis. After the storage period, the spent fuel must either remain in: a) a condition, known prior to storage, to be transportable, or b) a known or bounding condition that can be analyzed to determine if the spent fuel is transportable. If the configuration of the spent fuel changed during storage to a point where it could not be retrieved from the storage canister by normal means, and could not be analyzed to determine if it was transportable, it would have to be removed from the canister for analysis to determine suitability for transport. Having the canister retrievable from the storage cask does not guarantee that the spent fuel is transportable.

2.4. License renewals

NRC regulations provide for a 20-years' term for each specific license for an ISFSI with the option to renew the license for an additional 20-year term. The general license for each storage cask terminates 20 years after the storage cask is first used by the licensee and similarly, each storage cask certificate of compliance is valid for a 20-year term with the option for renewal.

Starting in 2005, the NRC renewed the site-specific licenses for Surry, H.B. Robinson and Oconee ISFSIs for an additional 40-year storage period beyond the initial license duration. NRC is currently reviewing the license renewal application for the Ft. St. Vrain modular dry storage facility. As part of the license renewals, staff reviewed and evaluated revised technical requirements and operating conditions that addressed those aging mechanisms that could affect the safe storage of the spent fuel. The staff's review was primarily a materials review to determine whether aging over a 60-year period would reduce the storage cask's ability to confine and shield the fuel; increase the risk of criticality; or change the material properties of the spent fuel such that the fuel would either have gross ruptures or become irretrievable after storage.

Prior to receiving the license renewal application from Virginia Power for the Surry Nuclear Power Plant, the staff met with the Virginia Power to discuss license renewal and provide the licensee with the staff's philosophy for the contents of the renewal application. Staff provided the applicant with its preliminary guidance in March 2001. Staff developed its guidance based on existing NRC guidance for licensing spent fuel storage facilities and the Draft Regulatory Guide DG-1104, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," August 2000 [7].

Staff reviewed renewal requests from Surry, H.H. Robinson and Oconee nuclear power plants. In their applications for renewal, the licensees requested an exemption from the 20-year license renewal term in NRC regulations and requested that the term be extended for 40 year. In addition to the exemption request, the applications included results of their aging management reviews and activities, time-limited aging analyses, supplemental information to the safety analysis report, revised technical specifications and a detailed environmental analysis covering the 40-year renewal period.

The renewals determined the structures, systems and components considered to be "in the scope" of the review. Note that the NRC does not consider license renewal to be a complete re-review of all analysis submitted for the initial licensing process. In-scope components were those that are important to safety or those that, although not important to safety, whose failure could prevent a safety function from being performed or, could prevent a safety function that is important to safety from being performed. For those in-scope components, the primary issues for license renewal are aging and environmental-related materials degradation. These effects require evaluation by an aging management review process.

- The aging management review process involved the following four major steps:
- Identification of in-scope subcomponents requiring aging management review;
- Identification of materials and environments;
- Identification of aging effects requiring management;
- Determination of the activities required to manage the effects of aging.

The in-scope subcomponents were evaluated to determine whether exposure to either the site environment at the ISFSI or the cask internal environment containing the spent fuel would cause material degradation. The spent fuel is also an identified subcomponent, and is subject to its own evaluation for potential degrading mechanisms and effects.

The second step of the aging management review process involved identification of the materials of construction for in-scope components for each type of cask and the environments to which these materials are exposed. The environmental conditions include any conditions known to exist on a recurring basis, given the original design criteria and operating experience, unless design features have been implemented to preclude those conditions from recurring. The internal environment included neutron and gamma fluence, temperatures as well as fill gas. The effects of aging on the casks were also evaluated based on their initial loading in borated water.

From an engineering assessment, two aging effects, associated with the casks, were identified to require management—loss of material through degradation and changes in material properties such as through radiation-induced changes in polymers. Additionally time-limited aging analysis, such as thermal fatigue, was identified as an issue requiring analysis.

The review of materials that may be susceptible to corrosion, confirmed the continued need for performing periodic inspections. The periodic inspections provide reasonable assurance that the casks will perform their intended safety function(s) throughout the renewal period by identifying any corrosion before it progresses to the point of adversely affecting the performance of the component. Chemical changes in material properties, radiolytic decomposition, and thermal degradation of polymeric compounds can produce off-gassing and a reduction in weight of the polymer, which could potentially affect the performance of the neutron shielding material.

The result of the time-limited aging analyses determined that the original analyses were found to remain valid. The licensees further showed that no degradation due to time-limited aging would affect the functionality/safety of any specified components for the duration of the proposed 40-year license renewal period. Based on these analyses, staff renewed the ISFSI licenses for Surry, H. B. Robinson and Oconee ISFSIs for a 40-years' period.

Staff utilized experiences gained in renewing the three ISFSI licenses to concurrently developed a rulemaking to increase the duration of spent fuel storage cask licenses and certificates of compliance from a 20–40-years' duration and NUREG-1927, "Standard Review Plan for Renewal of Independent Spent Fuel Storage Installation Licenses and Dry Cask Storage System Certificates of Compliance".

3. CONCLUSION

The NRC strives for continuous improvement. The NRC staff continually questions not only the applications under review but also the processes and procedures staff uses to perform the evaluations. Shortly after publication of the standard review plan for dry cask storage systems, the staff began the process of revising it, only to find that it would be in a continual revision process based on experiences gained during licensing and certification. In order to reduce the staff effort needed to completely revise the standard review plan, the NRC developed ISGs to fill in the gaps between revisions of the standard review plans. The first ISG was issued in November 1998.

Review methods and guidance are not stagnant. As one gains experience in performing technical reviews of spent fuel storage casks, especially in a country with a large number of nuclear power plants which are not standardized, one finds that many sites may each have a different circumstance that the review guidance didn't anticipate. Flexibility to revise the guidance, based on experiences should be the standard throughout the engineering world.

The NRC has 30 years' experience in licensing and certifying storage casks and significantly more experience in regulating radioactive material transport. Given the number of storage facilities that will be used throughout the world over the coming years, ensuring that the licensing process has strong technical bases are vital to ensuring extended storage and transportation are performed safely. NRC has been recently tasked with completely re-evaluating its technical bases for spent fuel storage and transportation. The Commission issued the staff a Staff Requirements Memorandum-09-0001, "Revisiting the Paradigm on Spent Fuel Storage and Transportation Regulatory Programs" (SRM-COMDEK-09-0001) to thoroughly re-evaluate how the NRC licenses and performs reviews of transportation packages, and storage casks.

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SPENT FUEL MANAGEMENT IN SLOVAKIA

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Abstract

The paper describes the SFM system in the Slovak Republic. In 2008, the Slovak Government accepted in its Decision Nr. 328/2008 "The proposal on the strategy of the back-end of the nuclear power engineering". The state supervision on nuclear safety of SFM is performed by the Nuclear Regulatory Authority of the Slovak Republik (UJD). The legislative framework in the Slovak Republic is based on acts and regulations. In Slovakia there are four nuclear power units in operation. The spent fuel is stored in at-reactor spent fuel storage pools and cooled by water with presence of the boric acid. After certain cooloing time, the spent fuel is removed to the Interim Spent Fuel Storage Facility (ISFSF). For the spent fuel transport transportation container C-30 is used. UJD steers various research tasks under the Research & Development program (R&D). Several years ago we started process of burnup credit (BUC) implementation in Slovakia for VVER-440 reactors. Another R&D project is focused on determination of the relation between the spent fuel residual heat generation and surface temperature of the transport container C-30. By the end of 2009 first two modules — visual inspection and gammaspectroscopy — of inspection stand SVYP-440 at ISFSF were put into operation.

1. INTRODUCTION

The skills in handling spent fuel have been collected in Slovakia for more than 35 years. During this time period a well established SFM system was created.

In Slovakia there are four nuclear power units in operation. These units produce about 300 spent fuel assemblies (approximately 36 ton of heavy metal) a year. For temporary storage of the spent fuel after its terminate reloading from the reactor core the at-reactor spent fuel storage pools are used. The spent fuel is stored in a grate and cooled by water with presence of the boric acid. 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). The capacity of the ISFSF is 14112 spent fuel assemblies. The spent fuel will be stored there for at least 50 years.

The Slovak Government established the basic policy of SFM 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 2008, the Slovak Government accepted in its Decision Nr. 328/2008 "The proposal on the strategy of the back-end of the nuclear power engineering". The material contains the philosophy of the SFM including the deep geological repository development.

The state supervision on nuclear safety of SFM is performed by 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, are guides for operators not binding.

Act No. 541/2004 Coll. on Peaceful Use of Nuclear Energy regulates the conditions for use of nuclear energy for peaceful purposes, the obligations and rights of legal persons and natural persons in the use of nuclear energy, the classification of nuclear materials, the conditions for their production, processing, procurement, storage, transportation, use, accounting and control, conditions for management of radioactive waste from nuclear installations and of spent nuclear fuel, state supervision of nuclear safety at nuclear installations, procurement and use of nuclear materials, management of radioactive waste and management of spent nuclear fuel.

Regulation No. 53/2006 Coll. on Radwaste and Spent Nuclear Fuel Management by which details of radioactive waste management and SFM are regulated. This regulation describes general requirements placed upon radioactive waste management and SFM. Spent fuel shall be managed to minimize the effect of ionizing radiation exerted upon operators, population and environment; to maintain subcriticality; remove residual heat and minimize generation of radioactive waste.

Regulation No. 57/2006 Coll. on the Details of Transport of Radioactive Materials and Radioactive Waste regulates the process and methods of road, rail, water and air transport of radioactive material, radioactive waste from nuclear facilities and spent nuclear fuel and the scope and content of the documentation required for issuance of approval for transport of radioactive material.

Guide of UJD on Construction and Operation of Spent Nuclear Fuel Storages describes requirements for design and operation of spent nuclear fuel storage, especially fulfillment of safety functions. Guide provides detailed information on realization and control of these functions during the whole operating life. Guide was developed according to the IAEA requirements for spent fuel handling and in accordance with Act No. 541/2004 and Regulation No. 53/2006.

2. SPENT FUEL MANAGEMENT

In 2009, the UJD approved the spent fuel transportation container C-30 for next utilization. The license was issued for the transport of spent nuclear fuel from four units in operation as well as from two shut-downed units. In order to be able to start the decommissioning of shut-downed units earlier, the licensee requested specific conditions for the transport. The residual heat was increased and the cooling time was decreased. Tables 1–3 show the approved conditions for transport container C-30.

TABLE. 1 THE ORIGINAL CONDITIONS FOR C-30 TRANSPORT CONTAINER

	Wet transport	Dry transport
Max. amount of assemblies	30	30
Basket type	T-12	T-12
Max. residual heat	15 kW	8 kW
Avg. burnup	40 $MWd \cdot kgU^{-1}$	14 $MWd \cdot kgU^{-1}$
Max. burnup	44 $MWd \cdot kgU^{-1}$	15 $MWd \cdot kgU^{-1}$
Max. residual heat of one assem.	630 W	—
Max. enrichment	3.6% U235	2.5% U235
Min. cooling time	2.5 years	2.5 years

TABLE. 2 THE CONDITIONS APPROVED BY THE LICENSE NR. 227/2009

Basket type	KZ-48	T-12	T-13
Max. amount of assemblies	48	30	18
Max. residual heat	24 kW	24 kW	24 kW
Avg. burnup	50 $MWd \cdot kgU^{-1}$	46 $MWd \cdot kgU^{-1}$	50 $MWd \cdot kgU^{-1}$
Max. Burnup	55 $MWd \cdot kgU^{-1}$	50 $MWd \cdot kgU^{-1}$	55 $MWd \cdot kgU^{-1}$
Max. residual heat of one assem.	605 W	605 W	605 W
Max. enrichment	4.4% U235	3.82% U235	4.4% U235

The minimum cooling time depends on initial enrichment and basket type and varies from 2.8–3.6 years.

After shut down of first unit of V-1 NPP by the end of 2006 the operator, in order to shorten the transition period from operation to decommissioning, applied for a new license for transport of spent fuel form V-1 NPP to ISFSF. The residual heat production for one assembly was increased to 800 W, which is also limit for ISFSF. Also required cooling time was shortened for some assemblies.

TABLE 3. THE CONDITIONS APPROVED BY THE NEW LICENSE FOR V-1 NPP

Basket type	KZ-48	T-12	T-13
Max. amount of assemblies	30	30	5
Max. residual heat	22,4 kW	22,4 kW	4 kW
Avg. burnup	44 $MWd \cdot kgU^{-1}$	44 $MWd \cdot kgU^{-1}$	44 $MWd \cdot kgU^{-1}$
Max. Burnup	44 $MWd \cdot kgU^{-1}$	44 $MWd \cdot kgU^{-1}$	44 $MWd \cdot kgU^{-1}$
Max. residual heat of one assem.	**800 W**	**800 W**	**800 W**
Max. enrichment	3.82%	3.82%	3.82%

The minimum cooling time has been determined to 1.8 year.

3. RESEARCH AND DEVELOPMENT

UJD steers various research tasks under the Research & Development program (R&D). The Division of Nuclear Materials has executed a task of the burnup credit (BUC) application in the criticality calculation of the VVER-440 fuel assemblies in cooperation with Nuclear Power Plants Research Institute (VUJE). The task was divided into two parts - first is already finished (years 2005–2007), second is in progress (years 2008–2010).

The aim was to examine possibilities of the VVER-440 spent fuel storage and transport with higher original enrichment in the existing storage and transport facilities. It consists of the analysis of the possibility to transport and store the VVER-440 spent fuel with original enrichment up to 5% U235 in the existing C-30 transport container with T-12 or KZ-48 casks and in the at-reactor spent fuel storage pools.

Under those subtasks we have developed methodology for BUC utilization, taking into account actinides only, and we have validated the SCALE 5.0 system as a tool for VVER-440 fuel.

The second part of the project will also include fissile products. This subtask started in 2008 and will be finished in 2010.

In order to have validated results three Slovak organizations (VUJE, JAVYS, UJD) have joined an international consortium focused on further investigation of nuclide composition of VVER-440 spent fuel within the framework of project ISTC #3958. Having these results we will continue the verification of the SCALE 5.1 and 6 systems for nuclide composition calculations. The UJD will prepare a guide on BUC application in Slovakia.

The BUC will be necessary for the licensing of the new fuel with enrichment of 4.87% 235U in at reactor pool and in basket KZ-48.

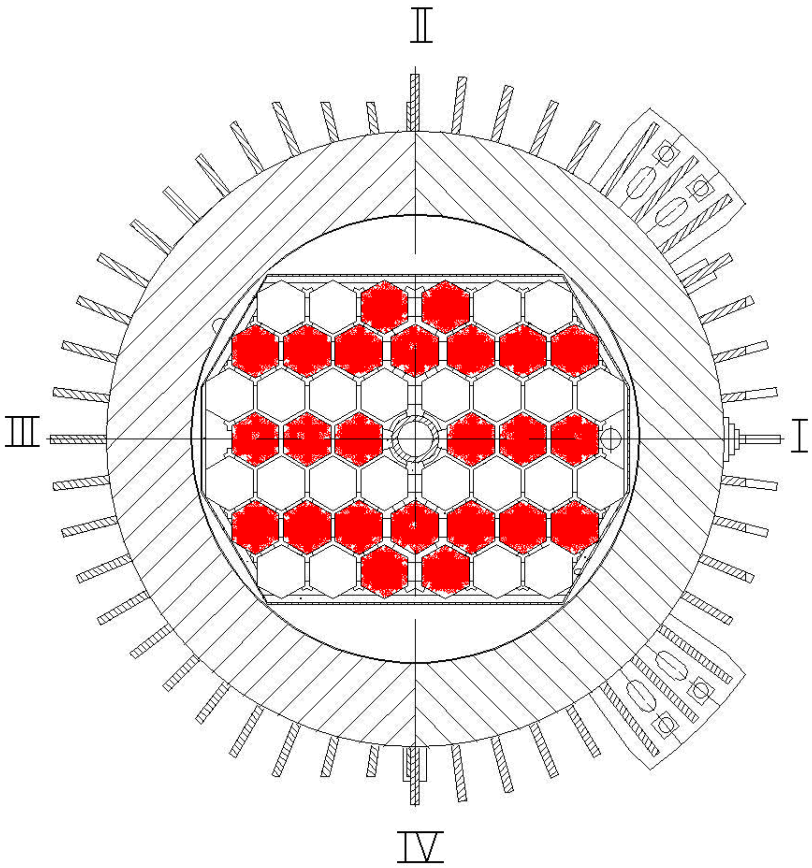

FIG. 1. Conditions of experiment.

Last item is the preparation of the safety reports (for transport and storage) for the new fuel with average enrichment 4.87% in basket KZ-48 with burnup credit application.

Another R&D project is focused on determination of the relation between the spent fuel residual heat generation and surface temperature of the transport container C-30. The residual heat generation is calcualted by a special software. During the transportation of the spent fuel the surface temperature of the transport container is limited. The results of this project will enable better anticipation of the surface temperature and residual heat release.
The project simulates real condition during the transport of spent fuel in transport container C-30 with basket KZ-48 inside. Figure 1 shows the conditions during the experiment (red means heated). In each position in the basket KZ-48 we placed a dummy assembly in order to

have the same volume of water inside the transport container C-30. Every second dummy assembly has an electrically heated coil. Temperature is measured inside of the transport container as well as on selected spots on surface. The results will be processed and a mathematical dependancy between known heat and surface temperatures will be calculated.

4. SPENT FUEL MONITORING

In 2005, the operator of the ISFSF started installation of an inspection stand. The stand is intended to be used for dismantling of leaky assemblies. Besides, the stand will be used for various measurements and monitoring of the condition of spent fuel. The inspection stand SVYP-440 will have following modules:

- Remote visual inspection of the selected surfaces of fuel assemblies and their components;
- Ultrasonic inspection of the cluster fuel elements;
- Eddy-current inspection of clad integrity of the individual fuel elements;
- Gamma-spectrometry of the individual fuel elements;
- Measurement of length of the fuel column in cluster of the fuel elements;
- Spectroscopy measurement of length of fuel column of the individual fuel elements;
- Optical measurement of length of the individual fuel element;
- Diameter and ovalness measurement of the individual fuel element by induction method;
- Optical measurement of deflection, torsion and length of the fuel assembly;
- Mechanical measurement of clearance between fuel and clad of the individual fuel elements;
- Eddy current measurement of oxide depositions on clad of the individual fuel elements;
- Oxide deposition sample intake from clad of the individual fuel elements and their consequent analysis;
- Pressure measurement of the fission products inside the individual fuel elements.

By the end of 2009, first two modules — visual inspection and gammaspectroscopy - were put into operation. The visual inspection of the spent fuel assemblies will enable to control the condition of the assemblies, whereas the gammaspectrometry will enable to measure burnup and selected nuclides.

5. CONCLUSION

New requirements on SFM (higher enrichment, higher burnup, residual heat generation, shorter cooling time, new licenses, and others) have required a new approach from UJD. UJD have started several projects, which results will be used for a better understanding of spent fuel behaviour during its storage, transportation and deposition.

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SPENT FUEL MANAGEMENT OF NPPS IN ARGENTINA

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Abstract

There are two Nuclear Power Plants in operation in Argentina: "Atucha I" (unique PHWR design) in operation since 1974, and "Embalse" (typical CANDU reactor) which started operation in 1984. Both NPPs are operated by "Nucleoeléctrica Argentina S.A" which is responsible for the management and interim storage of spent fuel till the end of the operative life of the plants. A third NPP, "Atucha II" is under construction, with a similar design of Atucha I. The legislative framework establishes that after final shutdown of a NPP the spent fuel will be transferred to the "National Atomic Energy Commission", which is also responsible for the decommissioning of the Plants. In Atucha I, the spent fuel is stored underwater, until another option is implemented meanwhile in Embalse the spent fuel is stored during six years in pools and then it is moved to a dry storage. A decision about the fuel cycle back-end strategy will be taken before year 2030.

1. INTRODUCTION

The uses and applications of nuclear energy have begun in Argentina in 1950, the year that the National Atomic Energy Commission (CNEA) has been created.

Since that time a very wide variety of activities were performed in the nuclear field by private and public entities. The radioactive waste generated is managed accordingly to the legal and national regulatory provisions in force, in agreement with the obligations derived from the "Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management".

The legal framework applicable to radioactive waste and SFM, integrates with the provisions of the National Constitution and with the legislation adopted by the National Congress by Act Nº 24804 (1997) which regulates the Nuclear Activity and Act Nº 25018 (1998) which determines the Radioactive Waste Management Regime [1]. The National Act of Nuclear Activity assigns to CNEA the state ownership of spent fuel and the responsibility for the management of radioactive wastes, thus becoming the *Responsible Organization*. The same Act sets forth that CNEA shall take full responsibility for the decommissioning of nuclear power plants and any other significant facility.

In 1994, National Nuclear Regulatory Body (ENREN) was created and in 1997, it was changed to Nuclear Regulatory Authority (ARN) by National Act 24804. ARN is empowered to regulate and supervise the nuclear activity in all matters related to radiological and nuclear safety, physical protection and safeguards. Likewise, it authorizes to ARN to supervise the use of nuclear materials, the licensing of persons and facilities and the verification of safeguards treated.

National Act Nº 25018 proposes the National Radioactive Waste Management Program (PNGRR) and CNEA as the responsible for the compliance with a specific *Strategic Plan* for Radioactive Waste Management. This *Strategic Plan,* updated in 2006, outlines the commitments that the National Government must assume for the safety of Spent Fuel

Management and Radioactive Waste Management, ensuring public health, the protection of the environment and the rights of future generations.

There are two Nuclear Power Plants in operation in Argentina, supplying about 8% of the national electricity production: "Atucha I" NPP (a unique PHWR design), and "Embalse" NPP (a typical CANDU 6). Both NPP are operated, since 1994, by Nucleoeléctrica Argentina S.A (NA-SA) which is the *Primary Responsible* for the management and interim storage of spent fuel till the end of the operative life of the plants. A third NPP "Atucha II", with a similar design of Atucha I, is under construction.

Government of Argentina exercises state ownership of special radioactive fission material contained in spent fuel from any origin: nuclear power plants and experimental, research and/or production reactors (Article 32, Act Nº 24804). Spent fuel is not considered radioactive waste and the reprocessing possibility remains open [1–2]. In that sense, according to the *Strategic Plan*, the decision for reprocessing spent fuel will be adopted before 2030. At such time the installation of an underground geological laboratory must have been started, which allows the design and construction of a deep geological repository, which must be operative by the year 2060. Meanwhile, the spent fuel generated by the two PHWR in Argentina is being stored in wet and dry interim storages. At the moment of decommissioning, an appropriate transfer of Responsible Entity will be needed and before that a decommissioning license should be required by CNEA.

2. ATUCHA I NUCLEAR POWER PLANT

Atucha I NPP (CNA I) is a 357 MWe PHWR of German origin which is in operation since 1974. It has started to operate with natural uranium fuel but from 1995–2000 the fresh fuel in the core was gradually modified from natural to slightly enriched uranium (0.85% nominal). This modification allows to increase the average burn up almost in a factor of 2 (11300 MWd/tU) [4], reducing the frequency of refuelling at full power from 1.29–0.72 fuel element per day and, subsequently, reducing significantly the number of spent fuel assemblies generated by year.

Atucha I fuel element (Fig. 1) is composed by 36 bars (plus a structural bar) with an active length of 5323 mm and a total length of 5566.4 mm. Each bar contains about 400 UO_2 pellets cladding in a zircaloy-4 alloy tube with an external diameter of 13.82 mm and 0.5 mm thick. The fuel assembly (153.5 kg of U) is very slender, with a total length of 6028.5 mm, an external diameter of 107.8 mm and a weight of approximately 200 kg. The total number of fuel elements in the reactor core is 252.

The data of each fuel element are introduced in an online computational program which allows knowing the history and location of each fuel element from the fresh fuel storage to its disposal location in the pool. The same software also calculates the burn up, the production and isotopic composition of U and the fission product inventory, based on the time of residence, the position of the fuel assembly inside the reactor core and the cooling time. This program is used as well for safeguards purposes, inter alias to obtain the mapping of the pools and to generate the itemized list of the spent fuel elements, which includes, for each fuel assembly, its uranium and plutonium masses.

The average nuclear production is 0.51 kg of Pu per spent fuel assembly in the case of natural uranium fresh fuel and 0.70 kg of Pu per spent fuel assembly in the case of 0.85% uranium fresh fuel. The inventory to the end of 2007 was about 8000 spent fuel assemblies for them with natural uranium and 1500 spent fuel for slightly enriched uranium, it means approximately 5 tons of Pu [1].

FIG. 1. Atucha I fuel element.

Spent fuel is stored temporarily under water. The Plant has two Pool Buildings, House I with two decay pools, and House II, with four decay pools, where the spent fuel is stored underwater hanging vertically in stainless steel racks, in a double layer arrangement. There is only one transfer channel between the reactor building and the maneuvering pools of the two houses. The pools are 17 m deep and the walls are made of concrete with 2 mm thick stainless steel lining meanwhile the floor, also made of concrete, has a 3 mm thick stainless steel lining.

The pool building with two storage pools is completed full with 3240 fuel elements. The initial capacity of the other Building (Fig. 2) was 6944 positions but a re-racking program was performed for a more compact arrangement between fuel assemblies in the pools [4], increasing the storing capacity of fuel elements to 8304 positions. This task implied some changes in the storage bracket-suspension beam and the relocation of about 5000 spent fuel elements.

Considering a load factor of 85%, the arrangement will satisfy the storage demand up to 2015 but the end of the operative life of the Plant would be reached in 2017. It means that the storage capacity should be increased at least in about 600 positions to cover the demand. From then on everything indicates that the implementation of a dry storage seems to be the best solution to create additional capacity for storing the spent fuel either in the case of extension of life or to transfer all the spent fuel out of the NPP in the case of decommission [3,5]. In the meantime, according to the short time available, a simplified dry storage conceptual design is being considered. It consists of underground vertical silos placed in a new building annexed to House I building, where the oldest fuel elements would be moved from the Pools. This design guarantees that the fuel elements do not need to be moved out of the controlled area and it would allow the operation at least until the end of life time of the Plant [6].

FIG. 2. Spent fuel pod building.

In the case of Atucha II Nuclear Power Plant (CNA II), with a similar design of CNA I (692 MWe), which is planning to start to operate during next year, the spent fuel will be transfer to pools (Fig. 3), with a storing capacity equivalent to ten years of normal operation, till a dry storage alternative will be defined.

FIG. 3. The CAN I spent fuel pod (under construction).

3. EMBALSE NUCLEAR POWER PLANT

The Reactor of Embalse NPP (CNE) is a typical CANDU 6 (648 MWe) on load PHWR that is in operation in Argentina since 1984. The fuel bundles (Fig. 4) are composed by 37 bars of 495.3 mm length. Each bar, containing 38 UO_2 pellets (natural uranium), is cladding in a zircaloy-4 alloy tube with an external diameter of 13.08 mm and 0.419 mm thick. The fuel assembly has an external diameter of 102.74 mm and the weight of the UO_2 contained is about 21.5 kg. The reactor core or "calandria" has 380 horizontal pressure channels with a capacity of 4560 fuel bundles (12 per channel).

FIG. 4. A typical 37 rod CANDU fuel element.

The frequency of refueling at full power is 15.2 fuel bundles per day and the maximum burn up is 7800 MWd/tU. The nuclear production is approximately 68 grams of Pu per spent fuel bundle. The inventory (wet and dry storage) at the end of 2007 was about 112 000 spent fuel bundles containing approximately 7.6 tons of Pu [1].

After leaving the core, the spent fuel bundles are transferred underwater to the reception bay with capacity for 4800 bundles. They are disposed horizontally on trays of a double array of 12 bundles each one. The trays are later transferred to the storage bay (Fig. 5) and stocked in piles with up to 19 trays. There is a third little bay for defective fuels, with a storage capacity of 300 bundles. All the pools are made of concrete with epoxy resin lining. The main bay has capacity to store 45144 spent fuel bundles, equivalent approximately to the quantity generated during 10 years at maximum power.

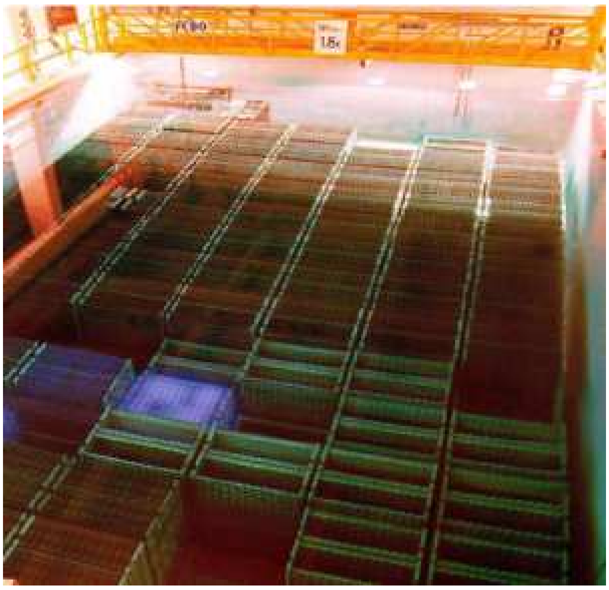

FIG. 5. Spent fuel storage bay.

A dry storage alternative was implemented in 1993 to cope with the spent fuel storage demand up to the end of the operative life of CNE. The spent fuel bundles must remain at least 6 years in the wet storage for thermal cooling and radioactive decay after being transfer to the dry storage.

The dry storage is a modular array of concrete “silos” (Fig. 6), also called “canisters”, arranged in a yard at the power station site. Inside each full loaded canister nine steel sealed baskets are piled, each one with 60 bundles. The baskets are loaded under water and then removed from the pool inside a shielded transfer cask. After living the pool, the cask is drained and introduced in the operational cell where it is sealed by welding [7–8].

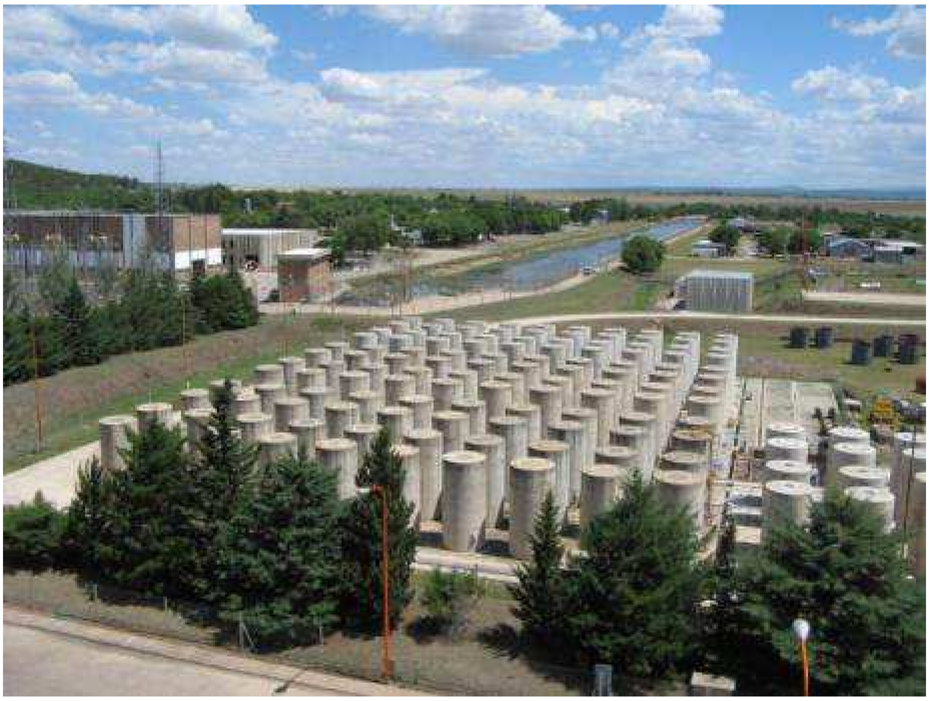

FIG. 6. Modular array of concrete silos.

A second cask named the “transfer shielding” is used to transfer the sealed basket from the operational cell to the canisters. The canisters yard is inside the double fence which surrounded the NPP for security protection.

The canisters are 6.3 m high vertical cylinders, with approximately 3 m of external diameter [8]. They are cooled by natural convection and were designed to support some accidental events as earthquakes, floods, tornadoes and the risk of explosions. The system provides three barriers of protection to avoid the release of radioactive material to the environment (fuel rod shield, steel basket and the canister). It is completely independent of the power station systems and not requires especial activities of maintenance when the canisters are filled and sealed. At present there are 216 silos in CNE and 152 of them are full loaded.

CNE was designed with a 30 years nominal life at an average load gactor (LF) of 80%. Due to its very good performance (LF=88% in the last 10 years), the end of the Plant operative life will be reached before the 30 years design life and it is estimated to be in 2011. A project for a refurbishment (life extension) is under development. This does not represent a problem regarding the spent fuel storage capacity because the modular design of the dry storage allows to be enlarged, with no fulfillment of special requirements, by the construction of a new battery of canisters when needed.

4. CONCLUSIONS

A decision about the fuel cycle back-end strategy will be taken before 2030. Meanwhile, the spent fuel in each nuclear plant is temporarily storage on site. In Atucha I, the spent fuel is being storage in pools but a dry storage design is under development. Embalse counts with a dry storage since 1993; the spent fuel, after certain period of time kept in wet storage, is transferred to dry storage silos. Enlargement of the spent fuel interim storage capacity at CNE is made easily through the construction of new modules of dry storage silos. The management of the spent fuel is a responsibility of the operator (NA-SA) during the operation life of the plants. At the time of decommissioning, the responsibility will be of the Responsible Organization (CNEA). A reasonable time before that moment both Entities must come to make the appropriate transfer agreements as well as CNEA should start the decommissioning license procedure.

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INTERFACE ISSUES ARISING BETWEEN STORAGE AND TRANSPORT FOR STORAGE FACILITIES USING STORAGE/TRANSPORT DUAL PURPOSE DRY METAL CASKS

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Abstract

The dual purpose dry metal casks were developped as a low cost and reliable design to handle spent fuel safely, not only in relation to storage, but also transportation. One of its main advantages is to enhance worker protection against radiation while reducing possible direct manipulation of the spent fuel. In order to define regulation and the use of this type of casks, a traditional approach can be used, based on the study of every individual aspect. However a new type of approach is possible, called the “holistic approach”, taking into account the different aspects as a whole.

The amount of spent fuels stored in nuclear facilities is increasing worldwide. In order to meet the increasing demand to store spent fuels safely and for a longer term, various types of casks are being developed. Dual purpose dry metal casks (DPDMCs) are one of such designs, and the advantage of introducing DPDMCs lies in that they do not need loading operation in storage facilities, both before and after storage, thus will contribute to the reduction of radiation exposure for facility workers.

One possible regulatory approach for DPDMCs can be named as “traditional approach”, traditional in the sense that regulators will make safety analysis from “transport” aspect and “storage” aspect separately. Typically, DMDMCs can be designed as transport casks whose long-term integrity is well considered so that they will survive decades of use. Storage facilities, on the other hand, can be so designed that safety of the facility that would accommodate such casks. One of the merits of this approach is that this will fit the regulatory schemes (including those laid out by the IAEA as safety standards and guides) so far.

However, there can also be another regulatory approach for DPDMCs, which can be named as “holistic approach”. This approach is to see the DPDMCs’ character as transport casks and as storage facilities in a holistic manner. In fact, the technical characteristics of DPDMCs lie in that they are rather “interdisciplinary” in nature, in the sense that that they are typically be loaded with spent fuels in NPP sites (not in storage sites), transported to storage facilities and stored for decades, and will be re-transported to their destinations. In other words, the “length” of one typical operational cycle of DPDMCs is longer than one operational cycle of storage facilities.

This paper is intended to emphasize the importance of introducing holistic approach for storage facilities that use DPDMCs. In addition, I would like to argue that holistic approach be applied not only on safety analysis but also in “system” issues such as allocation of responsibilities among various operators.

1. BASIC IDEAS OF DPDMCS AND THEIR APPLICATION IN JAPAN

DPDMCs are intended to be licensed both for storage and transport. Their designs are based on passive safety concepts, and are aimed to achieve longer term storage period with lower emissions of nuclear wastes as well as lower operational costs. DPDMCs have already been introduced in storage facilities in countries including USA, Germany, Switzerland and Belgium as well as several Eastern European countries.

In Japan, there are more than 12,000 tU of spent fuels stored in 55 NPPs across the country, while almost all of them stored in wet pools. Dry storage began in 1995 in Fukushima-Daiichi (No.1) NPP by TEPCO, followed by Tokai-Daini (No.2) NPP by JAPCO in 2001. Today, the total amount of spent fuel stored in dry casks is about 235tU (in 24 casks). While the advantages of DPDMCs have been broadly recognized in Japan, these two on-site dry storage facilities did not use this design concept, and these casks are simply designed as storage casks. This is mainly because the regulatory schemes in Japan at that time did not expect the appearance of DPDMCs, and did not match well with their technical characteristics.

In response to the increasing demand to store spent fuel safely and efficiently, following the above two dry storage facilities, the first large scale off-site dry storage facility was planned by TEPCO and JAPCO. In order to manage this project, they established their subsidiary company named RFS in 2005. For the regulatory sector, guideline for DPDMCs has been laid out by NSC of Japan in 2002. Thus, RFS decided to introduce DPDMCs design for their new storage facility. RFS submitted application for storage facility in 2007, and NISA, having examined this application, issued storage license in 13 May 2010. The operation for this facility is scheduled to begin in July 2014.

2. SAFETY ANALYSIS FOR STORAGE FACILITIES USING DPDMCS

2.1. Traditional approach

Traditional approach of regulation toward storage facilities using DPDMCs are based on the following basic approaches for storage safety and transport safety. As these two safety perspectives are checked separately, there can be a chance, for example, that one of these licenses can expire during storage period. In addition, it is not clear how to secure post-storage transport safety, as any degradation of spent fuels during stored period are not taken into consideration in transport safety analysis.

2.1.1. Storage safety

Safety analysis for storage facilities using DPDMCs generally examines whether they can maintain basic safety functions (e.g., containment, heat removal, subcritical and radiation protection functions) throughout the designed storage period. In addition, spent fuels stored are generally required that claddings deterioration be kept within designed safety margins. Such deterioration mechanisms can be classified as:

- Chemical deterioration;
- Thermal deterioration;
- Mechanical deterioration;
- Radiation deterioration.

2.1.2. Transport safety

Safety analysis for transport of DPDMCs generally examines whether they can satisfy conditions described in transport regulations (TS-R-1 of IAEA). Transport licenses for DPDMCs are generally issued with up to 5 years duration. As DPDMCs are intended to be used for storage casks during storage, transport license for DPDMCs are supposed to be renewed several times during storage periods.

2.2. Holistic approach — safety issues

In order to fully extract the merits of DPDMCs, storage facilities can be such that do not equip with "hot cells". Or in the case that such facilities are equipped with "hot cells", it is more desirable that they are not used unless necessary in order to avoid unnecessary radiation exposures for workers. In both cases, DPDMCs are loaded in NPP sites and brought into storage facilities for storage, and after their designed storage period of decades, will again be transported to their next destinations. Important point here is that their lids (especially primary lids) are not considered to be opened in storage facilities.

Under such situation, there appear several interface issues that relates both storage and transport safety, and it seems necessary for regulatory bodies to introduce holistic approach for the better handling of their transport and storage safety analysis. NSC's guideline in 2002 introduced this holistic approach, and NISA, using this guideline, conducted safety analysis as described below.

2.2.1. Interface issue No.1 — Post storage transport safety depends on safety during storage

After storage of decades, and when initially designed storage period elapsed, DPDMCs will be transported to other facilities for reprocessing or disposal, depending on the national nuclear fuel cycle policy. Transport safety for such post storage should be conducted based on the proper evaluation of long term integrity of DPDMCs and their contents.

Therefore, in making safety analysis for the transport license of DPDMCs, transport regulatory authority should well take into consideration of possible long-term deterioration of cask materials as well as spent fuels inside the casks. In reality, transport licenses are to be issued with shorter terms than storage period, and such consideration can be conducted at the time of license renewal. However, as the DPDMCs are not supposed to open their lids throughout the whole storage period, it is advisable that necessary analysis of such deterioration should be conducted well in advance. In the case of Japan, transport safety division of NISA will well keep in touch with storage safety division, and utilize mutual analysis for license evaluation.

2.3. Holistic approach — system issues

In addition to issues related to safety analysis, system issues should also be considered holistically. System issues here indicate allocation of responsibilities among operators and handling of related records. Typical examples for this issue can be observed in the following fields.

2.3.1. Interface issue No.2 — Storage safety significantly depends on safe transport from NPP to storage facility

DPDMCs are already sealed when they are carried into storage facilities. Important safety operations that would affect safety during whole storage periods (i.e., pick-up of appropriate fuels, vacuum dehydration, inert gas filling, etc.) have already been conducted in NPPs when

DPDMCs are sealed, not by storage facility operators but by NPP operators (not by storage operators) as a part of transport operation. Considering the importance of such operations toward storage safety, it is necessary that storage facility operators to obtain make accurate directions. In addition, records of such operation should be accurate and handed to storage facility operators from NPP operators. Such operation conducted by NPP operators are similar to pre-transport inspection, but the purpose of such operation is not to ensure transport safety but to ensure storage safety.

Another interface issue lies in their difference of the duration of license period between storage and transport. Transport license is generally issued with maximum 5 year duration, which is much shorter than storage period. As it is necessary for storage facilities that do not equip with hot cells and use DPDMCs, casks whose primary lids have failed should be transported to other facilities to fix them as soon as possible, thus casks are required to maintain transport license during storage.

In the case of Japan, NISA, considering the importance of these records, has decided to expand its safety regulatory framework to include NPP operators (spent fuel owners), as long as DPDMCs are intended to be used for storage. Also, for the transport license, NISA requests NPP operators (transport license holders) to conduct every possible effort to maintain transport license throughout storage period in order to secure reliable, continuous renewal of transport license during storage period. NPP operators, in response to this request, have decided to conduct long term integrity demonstrations of stored spent fuels. In an unlikely event that unknown deterioration mechanisms be revealed by this demonstration, NISA will examine its effects at the time of next transport license renewal, at the latest.

2.3.2. *Interface issue No.3 — Safety analysis for storage depends on the transportability of casks during storage*

As it is not expected to open lids of DPDMCs throughout storage period, as they are stored in a very static manner and their safety functions are passive in nature. However, should any unlikely faults be detected, and DPDMCs need to repair works that would accompany opening primary lids, they need to be transported to facilities that are equipped with hot cells.

In order that DMDPCs to be transported through public domain, it is required that they have valid transport licenses by appropriate regulatory authority. Thus, it is necessary that DPDMCs maintain transport license throughout storage period, in order to exercise their storage functions. On the other hand, transport sector should examine, whether DPDMCs are designed so that they can safely maintain transport license throughout storage period (taking into consideration of deterioration during storage period, of course), and that should such unlikely events occur, they still satisfy transport requirements.

In addition to the above issues, there has been an established practice in Japan that, in conducting low-enforced pre-shipping inspections for spent fuel casks, to conduct visual inspections of their contents' integrity. As DPDMCs are not considered to be open their lids at storage facility, and visual inspections of casks' contents not possible, records received at the time of receiving casks from NPP operators and records kept during storage period will substitute this visual inspection for ensuring integrity of contents. It has been discussed in NISA in 2009, with the help of many experts, that integrity of casks' contents can safely be said without visual inspections, that they are not significantly deteriorate that would make their transport to be unsafe, as long as spent fuels at the time of receiving to the storage facility are "sound" and various data taken during storage periods indicates no irregular signs, and such data can safely substitute visual inspection for pre-shipping inspections.

3. CONCLUSION

DPDMCs have their unique advantages in meeting increasing demand to store spent fuels safely, with low operational costs. In order to fully exercise the advantages of such casks, while providing public with sufficient protection against radiation, there seems to be a strong need to clearly recognize possible interface issues, especially issues interrelates to storage and transport should be considered well in advance. In response, regulatory sectors should provide possible solutions to such interface issues.

As the amount of spent fuel stored continue to increase for coming decades worldwide, it seems to be eminent that in some future, a major portion of transport of spent fuel consists of fuels stored long time. Regulators of MS should, I think, continue to utilize available resources to demonstrate the safety that relates to the use of DPDMCs storage, and conduct international cooperation on this field if necessary.

Annex

METHOD OF INSPECTION FOR TRANSPORT AFTER STORAGE FOR DPDMCS

Inspection at the time of loading operation

- Fuel selected based on instruction by storage operators;
- Fuel integrity to be confirmed (no leaking fuels allowed);
- Vacuum dehydration to be conducted based on instruction the by storage operators;
- Cask cavity to be filled with inert gases and lid to be sealed based on instruction by storage operators.

Inspection for transport before storage

- Fuel integrity to be inspected visually.

Inspection during storage

- Possible deterioration mechanisms can be classified into the following four categories:
- Chemical;
- Thermal;
- Radioactive;
- Mechanical;
- Inspection during storage will be so conducted that can detect any irregular incidents (that exceeds initially designed safety margins);
- As long as there are no irregular signs, cask contents can be considered to remain within safety margins at the time of initial design.

↓

Inspection for transport after storage

Based on the records of the all the above three inspections, it can safely be said even without visual inspections that integrity of contents and cask cavities for the DPDMCs are kept within safety margin for transport (inspection will be conducted based on records.)

SIMULATION OF SPENT PWR FUEL ASSEMBLY BEHAVIOR UNDER NORMAL CONDITIONS OF TRANSPORT

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Abstract

The behavior of a PWR high-burnup spent fuel assembly under normal conditions of transport is simulated in a dynamic analysis of a 0.3-m free drop of a transportation cask unprotected by impact limiters striking a flat rigid surface in the horizontal orientation. The structural analysis employs a finite element numerical model consisting of the cask, the fuel assemblies, the fuel rods, the guide tubes and the cask's internal structures that hold the fuel assemblies in position. Appropriate mechanical properties for the cask's structural components, as well as the elastic-plastic properties typical of high-burnup Zircaloy-4 cladding, are utilized. Emphasis is placed on fuel rods responses at locations where maximum forces would be expected, which include end-plate positions and spacer-grid positions at assembly mid-span. Temporal and spatial variations of the forces acting on the fuel rods are calculated and post-processed to obtain frequency distributions, which statistically represent the total fuel rod population in the cask. The results show that the largest pinch force, (ror-to-rod contact force), is 1700 lb, the maximum axial force is 600 lb, and the largest bending moment is 175 in-lb. Failure analysis of fuel rods using these force quantities, and considering the effects of potential hydrides re-orientation on cladding failure resistance, indicates, under conservative assmptions, a factor of safety of least 2 against longitudinal tearing, and no failure is predicted for transverse tearing or rod breakage. Fuel reconfiguration is predicted not to occur, and although partial tearing of guide tubes is possible, it is not enough to impair post-accident assembly retrieval.

1. INTRODUCTION

Transportation regulations [1] require that an evaluation of a transportation package design under normal conditions of transport include a determination of the effect on that design of a 0.3-m free drop onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected. Under such loading conditions, the geometric form of the package contents should not be "substantially altered", with no loss or dispersal of spent fuel, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the spent fuel package. The expected failure modes for the 0.3-m drop-accident simulation are depicted in Figure 1; they include: transverse tearing resulting in a pinhole failure, designated as Mode-I; extension of this mode to partial or complete (guillotine) breakage mode, designated as Mode-II; and longitudinal tearing, designated as Mode-III. An important consideration that is not regulatory prescriptive, but must be considered in the response analysis, is the potential for hydrides reorientation during dry storage. As the pressurized fuel rods cool down during dry storage, most of the hydrogen in solution in the cladding precipitates in the form of zirconium hydrides that may, in part, be oriented in the radial direction, which can affect the cladding resistance to Mode-III failure to some degree. The consequences of such hydride evolution to the integrity of high-burnup spent fuel are described in detail in EPRI-1015049, June, 2007 [2].

There are two elements to the evaluation methodology of high-burnup spent fuel subjected to dynamic forces resulting from cask drop events; these are: (1) performing global cask-drop analysis to determine the dynamic forces acting on the fuel rods, and (2) performing detailed local analyses to determine fuel rod local response and possible failure states. The first element is weakly dependent on high-burnup effects; it depends primarily on the effects of irradiation (fast fluence) on cladding and fuel assembly hardware mechanical properties. The second element is highly dependent on burnup and the cladding hydrides structure, and it utilizes the damage/failure model described in EPRI 1009693 [3], which has the capability to predict failure in any of the three possible deformation modes depicted in Fig. 1.

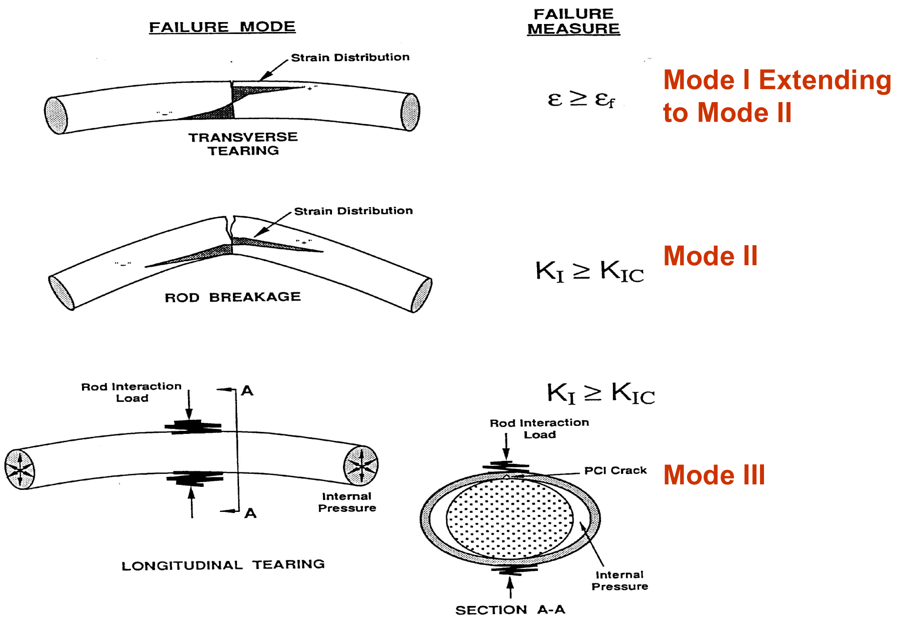


FIG. 1. Possible failure modes under cask drop.

2. GLOBAL STRUCTURAL ANALYSIS

A series of analyses were performed, following the modeling hierarchy employed in References [4–5]. This modeling hierarchy is summarized as follows: A 0.3-meter drop onto a rigid surface of a fully loaded cask, unprotected by impact limiters, is analyzed in the horizontal orientation using the three-dimensional slice finite element model depicted in Figs 2–3, and utilizing the ABAQUS-Explicit finite element computer program [6]. Two assembly models are shown in Figure 2: center-span model and top-span model. The center-span model encloses the spacer grid plane and two half spans, with two planes of symmetry at the half spans simulating continuity of the whole cask and fuel assemblies. The top-span model is intended to capture the behavior at the upper end plate, and it includes the top end plate, the spacer grid closest to the end plate, the fuel rods and guide tubes in the top span of the fuel assembly, and the spacer grid below the top plate. The guide tubes are fixed to the end plate. In order to capture sufficient variation of the dynamic response of the assemblies within the cask and obtain statistically significant probability distributions of forces, three different assembly configurations are considered: cells 2, 22 and 24 shown in Fig. 2. The assemblies in these positions are designated as “Control Assemblies” and are modeled in detail while the remaining assemblies are modeled as super beams, each having equivalent mass and stiffness of one assembly. The fuel rods in the Control Assembly are modeled individually, with each rod modeled as a beam having the stiffness of an empty tube and the total mass of the fuel and the cladding; the guide tubes are modeled as empty tubes.

The modeled cask is composed of a series of concentric structural shells made from A516 Grade 70 pressure vessel steel: a 2″ thick inner shell, followed by four 1.5″ thick shells, and a 1.0″ thick outer shell. The inside diameter of the inner shell is 68.75″ and the outside diameter of the structural shell is 84.75″. The outer structural shell is covered with a 4.4″ thick layer of neutron shielding material, which is then contained within an outer skin of 0.5″ thick steel. The total weight of the loaded cask is 124 tons. These geometric details are shown in Fig. 3, which illustrates the finite element model showing a blow-up view in the area of the bottom assembly in Cell 24. Fig. 3 also shows the 0.5″ thick stainless steel canister and the fuel basket assembly composed of strips of 5/16″ thick stainless steel. The canister with basket assembly is inserted as a free body inside the transportation cask.

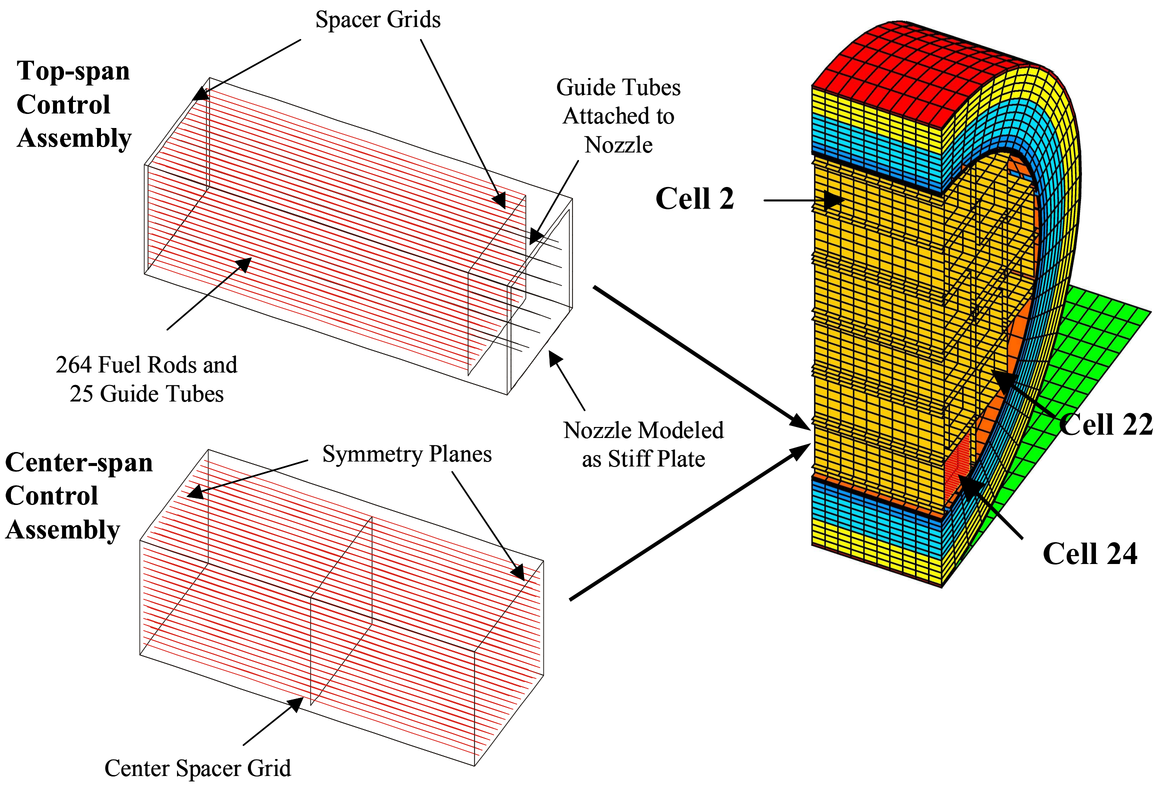


FIG. 2. Fuel assembly models within a three-dimensional slice of the global model.

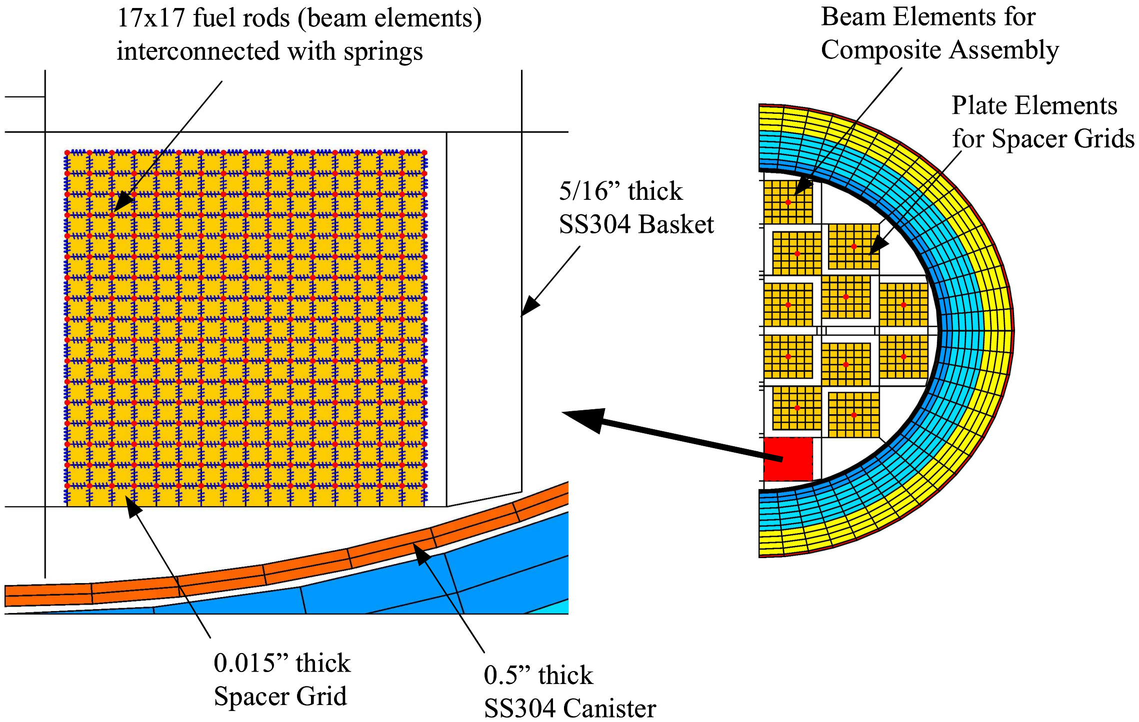


FIG. 3. 3D Centre spacer grid slice models.

Each of the 17 × 17 rods in the control assembly is modeled with 10 beam elements having mass and bending properties representative of a single fuel rod. The fuel pellets are assumed not to contribute to the fuel rod's strength or stiffness. Individual spring elements in both the horizontal and vertical directions are used to interconnect the fuel rods. These nonlinear spring elements represent a series of resistances to deformation, namely, the stiffness of the spacer grid leaf springs until the springs bottom out, then the stiffness of the spacer grid deforming up to buckling, and finally, the stiffness of rod-to-rod contact. In addition, a plane of plate elements with reduced shear stiffness properties is overlaid on the nodes with the spacer grid springs to provide some shear stability to the assemblage of spring elements. This is a numerical artifact to allow the fuel rods to stack up without rolling over, thereby maximizing the rod-to-rod interaction forces. A side numerical study was conducted to select the appropriate value of the shear stiffness that would not mask the true response of the assembly. Along the axis of the rods, contact springs are used to define any pinch forces that may occur away from the spacer grid plane. The rod-to-rod spring elements define the pinch forces that act on the fuel rods during the impact. Each of the other (non-control) fuel assemblies are modeled as a single composite beam. Numerous contact surfaces are used to insure that the reaction forces are distributed correctly between the various components of the model. Contact surfaces are defined between the outside surface of the canister and the inner surface of the cask and between the inside surface of the canister and the outer surfaces of the basket. There are 5 axial rows of elements on each side of the spacer grid mid-plane; contact springs are used to capture the interaction forces between the fuel rods.

2.1. Material properties

Table 1 lists the material properties for the four material types used in the cask/assembly model. Fig. 4a depicts the true-stress versus true-strain curve for steel material, and Fig. 4b illustrates the non-linear characteristics of the spacer grid and contact spring elements, This figure was generated through separate modeling of the spacer grid isolated from the assembly.

TABLE 1. SUMMARY OF MATERIAL PROPERTIES

Property / Units	A516 Gr70	SS304	Zircaloy	Shielding Material
Wt. Density / Lb/in^3 (kg/m^3)	0.284 (7861.8)	0.284 (7861.8)	1.5964[a] (44,192)	0.0607 (1680.3)
Modulus / msi (GPa)	29.5 (203.4)	29.5 (203.4)	14.1 (97.22)	0.561 (3.868)
0.2% Yield / ksi (MPa)	38.0 (262.0)	34.0 (234.4)	130.0 (896.3)	8.0 (55.16)
Ultimate Strength / Ksi (MPa)	70.0 (428.6)	84.0 (579.2)	-	10.5 (72.40)
Elongation %	30	50	-	5
Failure Strain %	8	20	-	5
Poisson's Ratio	0.3	0.3	0.425	0.25

[a] Density is adjusted to account for weight of fuel pellets.

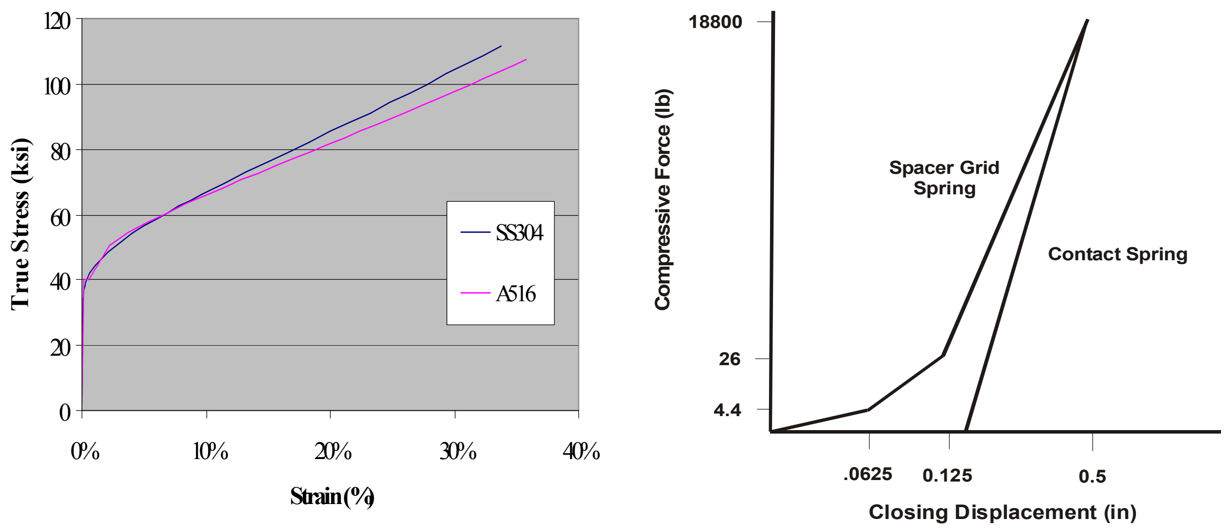


FIG. 4. (a): Stress-strain curve for steel materials and (b): force-displacement model for spacer grid springs & rod-to-rod contact springs.

2.2. Cask global response and fuel rod forces

Three separate analyses were performed for the cask's three configurations representing the three assembly positions depicted in Fig. 2. The dynamic forces were calculated at thousands of points; however, space limitations allow only selected results to be presented here for illustration. Figures 5 and 6 illustrate the time histories for the peak pinch forces in Assembly 22, which experienced the highest impact forces, at the spacer grid and at mid-span away from the spacer grid, respectively. The force analysis results from the 867 spacer grid springs in Cells 02, 22 and 24 were used to calculate frequency distributions for the forces and moments acting on the fuel rods. Fig. 7 shows the maximum-value distribution for the pinch force, which governs Mode-III failure, and Fig. 8 shows similar distributions for the bending moment, which governs Mode-I tearing of guide tubes.

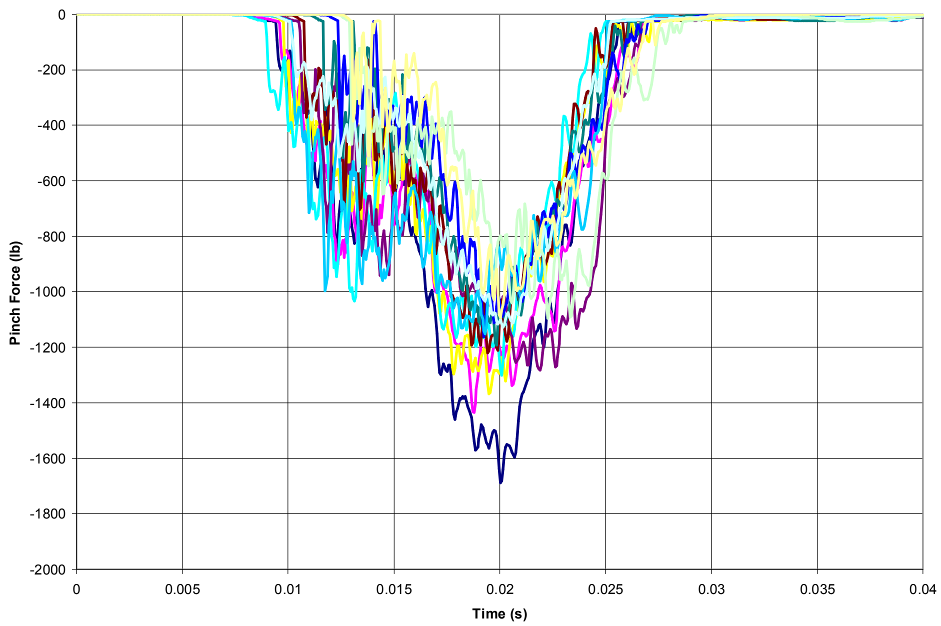


FIG. 5. Assembly 22 maximum pinch force time histories at spacer grid.

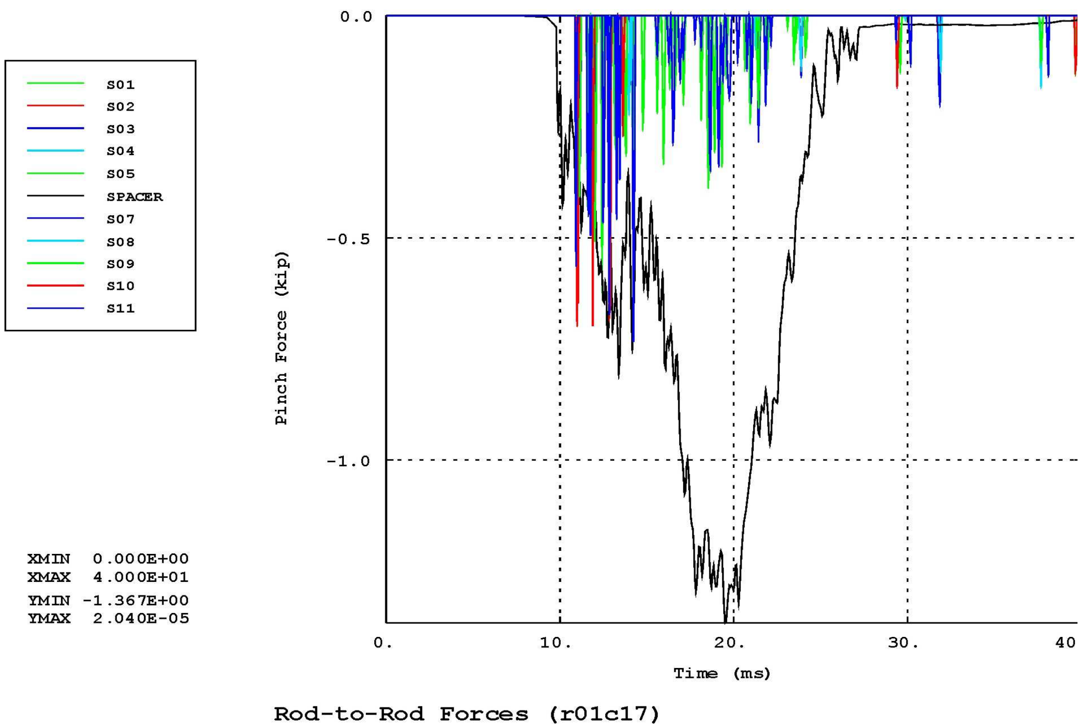


FIG. 6. Cell 22 Maximum rod-to-rod contact force time history at mid-span.

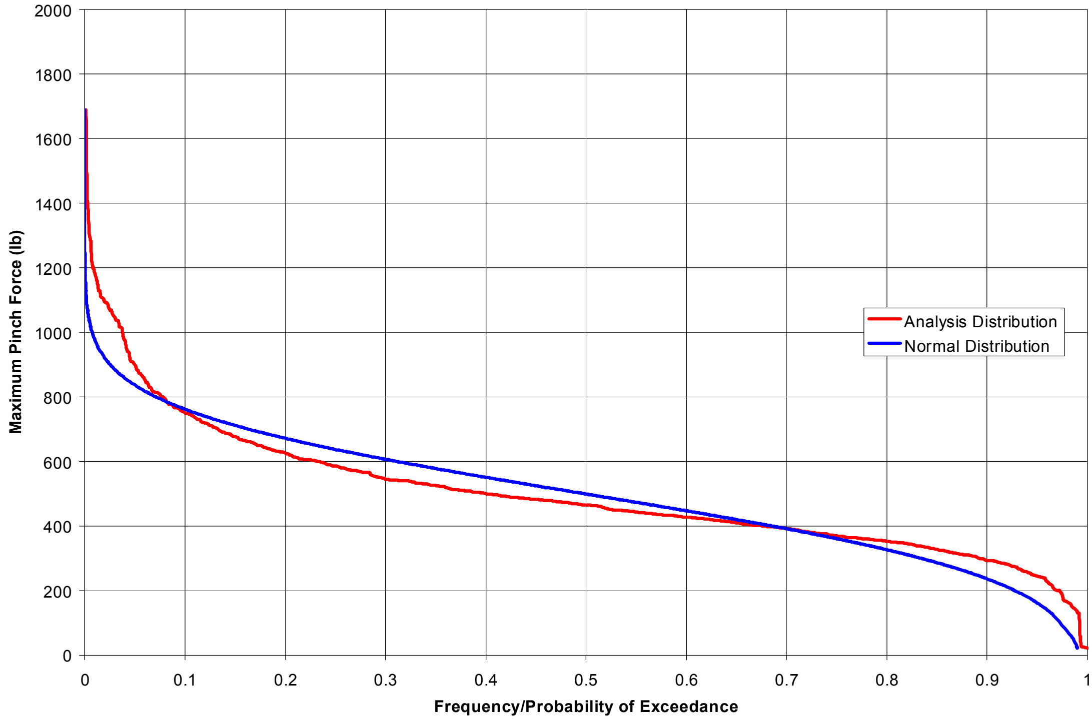


FIG. 7. Maximum pinch force frequency/probability distribution — spacer grid.

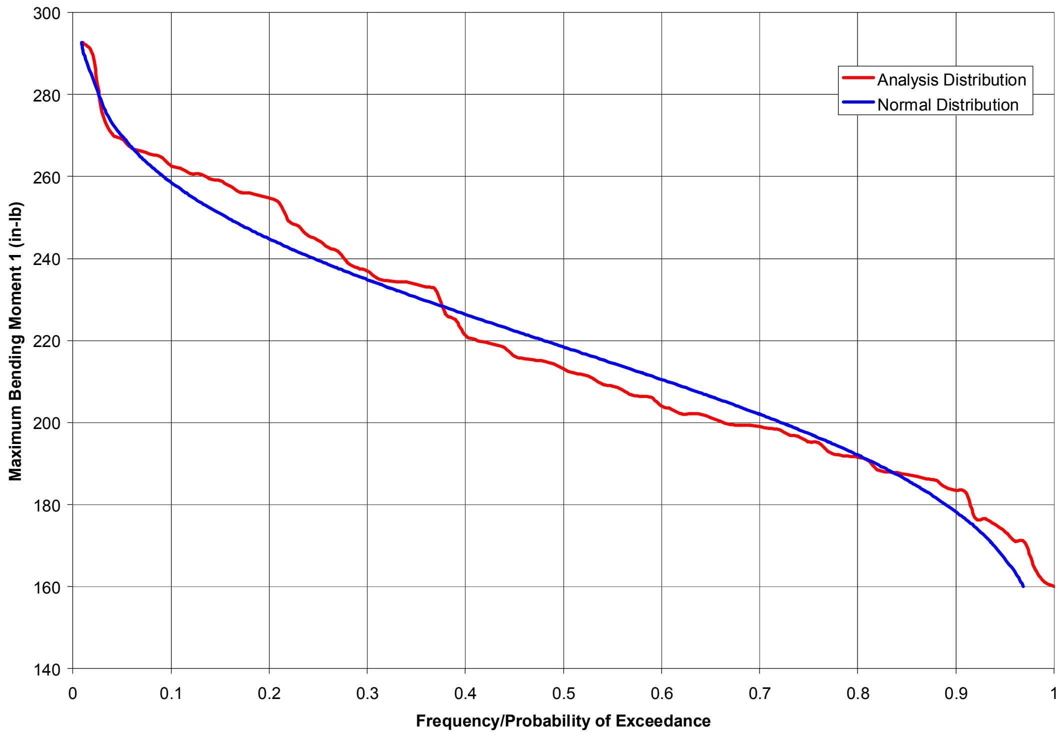


FIG. 8. Maximum bending moment 1 frequency/probability distribution top plate model.

3. FAILURE ANALYSIS

The foregoing cask drop calculations considered the cladding to be an elastic-plastic material, but without the ability to sustain damage or failure. This maximizes the forces acting on the fuel rods, which means that the bending moments and axial forces described earlier should be considered bounding target values that can be reached only if no cladding damage is possible. The type of damage that can be expected is of two types: fuel rod cladding fracture, possibly leading to fuel dispersal, or geometric distortion that would be equivalent to substantial alteration of the package contents. To comply with the 10 CFR 71.71 regulations governing normal conditions of transport [1], it is necessary to evaluate both of these damage scenarios, considering the maximum forces calculated, which are summarized in Table 2 below.

TABLE 2. MAXIMUM FORCES AND MOMENTS

Force type (units)	Center span (FR*)	Center span/End span (GT**)
Max pinch force (lb)	1700	640/520
Max axial (lb)	600	415
Max bending moment (in-lb)	175	348

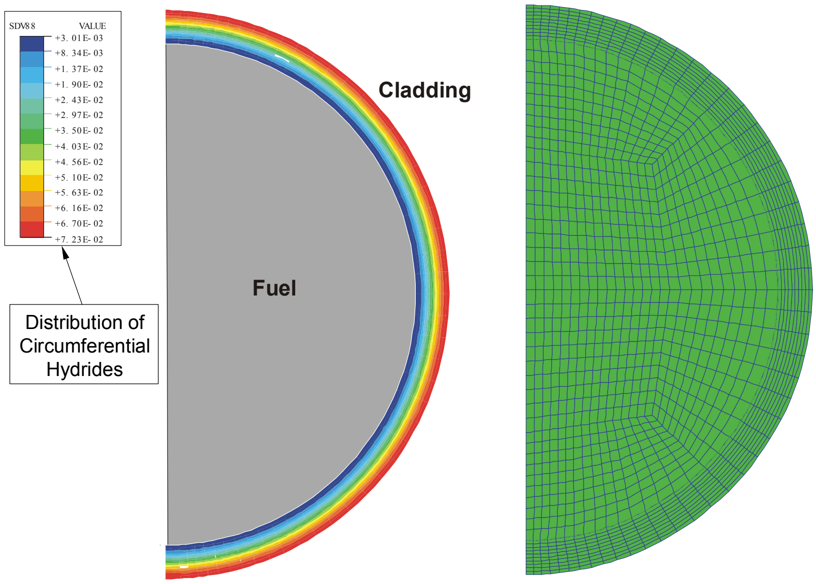
* Fuel Rods, ** Guide Tubes

3.1. Damage evaluation of fuel rods

Figure 1 in Section 1 depicts the three possible failure modes: transverse tearing Mode-I and rod breakage Mode-II, both of which are triggered by axial force combined with bending moment, and longitudinal tearing Mode-III, which is triggered by pinch forces. Mode-I/Mode-II failure is evaluated by applying the axial forces and bending moments to a detailed finite element model of a single fuel rod isolated from the assembly in an incremental static-analysis procedure combined with the three-phase mixture model, EPRI-1009693, 2004 [3], which calculates the evolution of damage as part of the response. Figure 9 shows the finite element grid and the circumferential hydrides fractional distribution in the cladding, which varies from 50 ppm (3E-3) at the ID to over 1050 ppm at the OD, with an average value of 600 ppm. The fuel is modeled as fractured material with no tensile capacity and reduced compression stiffness. The analysis results for potential Mode-I failure indicate that the fuel rod was able to resist 100% of the bending moment and axial force. Contours of the axial stress, plotted in Fig. 10, show a peak tensile stress in the range of 600–680 MPa. A stress of this magnitude is sufficiently close to the yield strength as to initiate damage at the outer surface of the cladding wall, but no significant damage or failure can be expected.

Fuel Rod Failure in the Longitudinal Tearing Mode-III is the only mode that is affected by hydrides re-orientation. The cask's thermal and stress histories during dry storage are such that radial hydrides of some concentration can form. In EPRI-10013448 [7], a 40-year dry storage period that maximizes radial hydride formation was considered. At the end of this storage period, a fuel-cladding diametric gap of 70 μm due to cladding creep is calculated. This represents the displacement that the cladding must experience under the 1700 lb pinch force, (see Table 2), before it begins to transfer the load resistance to the fuel pellets. At that stage of deformation, the stress-strain state in the cladding reaches a strain-energy density (SED) value of 9 MPa, which represents the demand on the cladding. The cladding capacity, the critical strain energy density (CSED), also calculated in EPRI-10013448 [7], is 16 MPa. This value of CSED is nearly a factor of 2 higher than the demand SED. An even higher margin would be obtained if more realistic stress conditions in the fuel rods were considered during the earlier period of cask handling..

The above results indicate that no axial splitting of the cladding is likely.



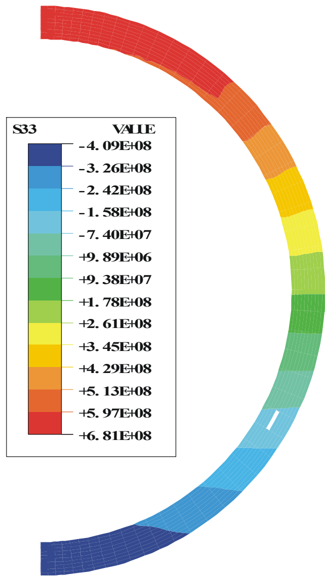


FIG. 9. Hydrides distribution & FE model.

FIG. 10. Maximum axial stress.

3.2. Damage evaluation of fuel assembly

Plastic distortion of the spacer grids and the breakage of the guide tubes are the two damage modes that could impair the fuel assembly's geometric continuity. As to the former, Fig. 11 shows that plastic distortion of the spacer grids does not occur. This is evidenced from the fact that the fuel rod compaction depicted in Fig. 11 B at the maximum response time of 20 ms recovers upon returning to rest at 40 ms, Fig. 10 C. Note the crushing in the shielding material, shown as missing elements from the grid, and the consequential local plastic indentation of the outer steel shell.

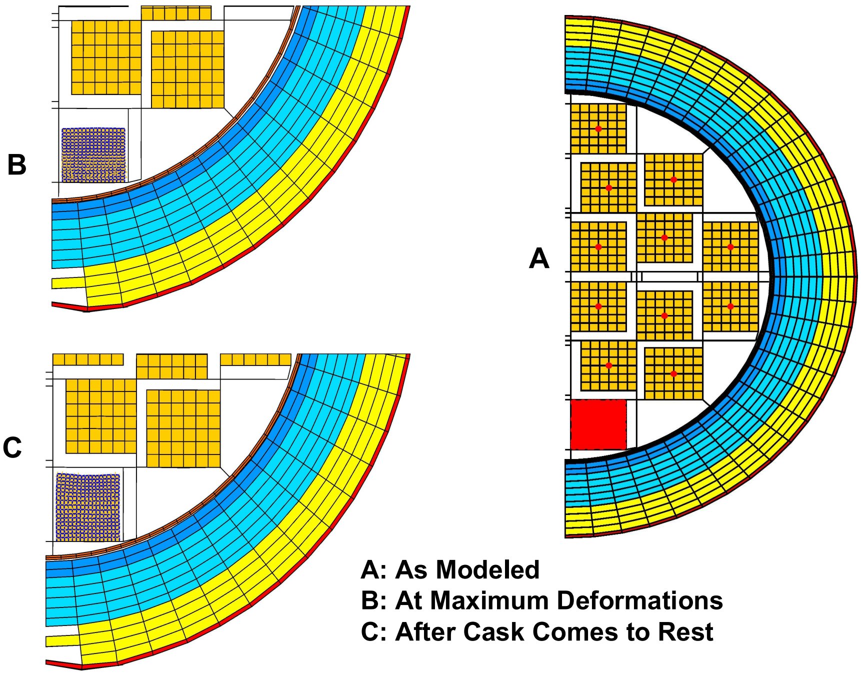


FIG. 11. End snapshot view of cell 24 assembly deformations at maximum response and after cask comes to rest.

3.2. Damage evaluation of guide tubes

The forces that affect the guide tubes transverse tearing are the bending and axial tension forces. The pinch forces, on the other hand, could cause plastic ovalization of the guide tubes. These potential damage states are evaluated in a local failure analysis using the finite element model shown in Fig. 9, adapted to the geometry of an empty guide tube. The Metal/Hydride-Mixture Damage Model [8], with the aid of the position-dependent failure criteria developed in [3], is used for predicting damage initiation in and progression through the wall as Mode-I failure. The results are shown in Fig. 12 in terms of axial stress and axial strain distributions in the tube at time of maximum damage. This figure shows that the guide tube cross-section experiences fracture by virtue of the fact that the Damage Model calcultes zero axial stress over the top 20% of the cross section. To evaluate the relevance of this result, the Metal/Hydride-Mixture Damage Model [8] was used to evaluate the ultimate load capacity by subjecting the partially damaged guide tube to a tension force until total loss of tensile capacity was predicted. The force calculated in this manner is 216 lb per tube, which constitutes the residual carrying capacity of the partially damaged guide tube. Assuming all of the 25 guide tubes have sustained the same level of partial Mode-I failure, the calculated 5400-lb assembly force is roughly 3.5 times higher than required to lift a 1500-lb assembly,.

A static elastic-plastic analysis, employing the Metal/hydride-Mixture Damage Model [8], was carried out for a one-inch axial slice of the guide tube. The guide tube deformed shape at maximum capacity is shown in Fig. 12. The static capacity was found to be 130 lb per inch. Using a typical dynamic load factor of 2, the guide tube dynamic capacity is 260 lb per inch, which is equivalent to 520 lb for the 2-inch wide spacer grid. Comparing these results to the calculated maximum pinch forces of 640 lb at the center span and 520 lb at the end span, (see Table 2), indicates that some of the guide tubes are likely to experience plastic distortion.

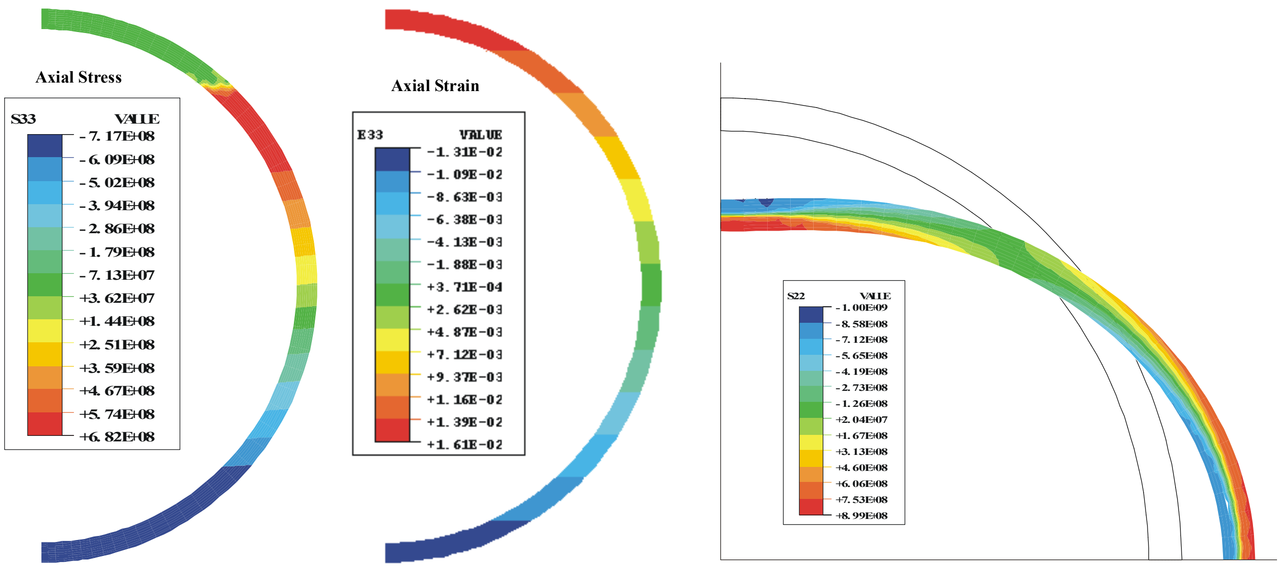


FIG. 12. Axial stress and axial strain distributions in the guide tube at maximum force & moment

4. SUMMARY AND CONCLUSIONS

The paper describes the response of high-burnup fuel rods to dynamic forces that could result from an accidental 0.3-m drop of a spent fuel cask unprotected by impact limiters onto a rigid surface in the most damaging orientation. The peak values of axial force and bending moment at the end of the loading are approximately 600 lb and 170 in-lb, respectively. The maximum pinch force calculated is 1700 lb. Hydrides re-orientation is shown to have no effect on cladding susceptibility to failure. A factor of safety against the longitudinal tearing mode occurring is calculated to be at least 2 under conservative assumptions. No failure is predicted for the transverse tearing (pinhole type) mode; however, stresses at or just below the yield strength of the material are calculated. This implies that damage initiation is possible, but progression of the damage to form a through-wall failure can be ruled out because the loading is totally exhausted upon damage initiation. Fuel reconfiguration is unlikely by virtue of the fact that spacer grid plastic distortion is predicted not to occur. However, plastic distortion and partial tearing of the guide tubes are possible, but not enough to impair post-transportation handling of the package contents.

ACKNOWLEDGEMENT

This work was sponsored by EPRI in a multi-year research program conducted under the direction of Dr. Albert Machiels in support of high-burnup spent fuel under conditions of storage and transportation.

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SAFETY AND LICENSING OF SPENT FUEL STORAGE AND TRANSPORT — SAFETY ISSUES WITHIN SPENT FUEL TRANSPORT

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Abstract

We can consider the different safety issues within French fuel transport as follows: (a) the proof as regards the leaking fuel assembly transport with hydrogen generation coming from potential in leakage water inside fuel rods; (b) the measures taken to enforce the new design as well as the new manufacturing which have been decided since January 1st 2007 in the frame of the 96 IAEA Regulation as regards the full water penetration as compared to the 85 IAEA Regulation, the latter allowing partial water penetration on certain conditions; and (c) the obligation of implementing various risk controls on exploitation site in order to take into account the possible human failure which are intrinsically increasing the permissible doses rates for workers. Even quite recently the leaking fuel assembly transport has been considered with no specific measure as regards the radiolysis phenomenon or the quality of drying cask holds. All these measures were sufficiently in accordance to rule out this issue. Lately, the leaking fuel assembly transport needs the implementation of equipment controls involved in nuclear power plants as regards the hydrogen rate before loading departure in order to determine on the evolution law, the maximum duration authorized for the transportation to not exceed the lower limit of inflammable status. As regards the proof of the criticality-safety casks, the main justification to be held on the irradiated fuel assembly on drop accident conditions could find a key in the hypothesis of the important damage of the fuel but should be in this matter, compensated by a limit of containment penetration for safety reason. For this case, the application of the 96 IAEA Regulation involves the use of independent leak tightness barriers. TN International is introducing different examples in France linked to the selection of multiple barriers. When limited in leakage quantity of water inside the cask is considered for the criticality studies, the French Competent Authority requires various independent controls before loading transportation in order to eliminate human failure. All this process being ensured for supplemental guarantees on the casks closing quality and without leaking after drying. TN International is introducing some examples to be implemented to suit this requirement of independent and various controls.

1. INTRODUCTION

The following paper develops some of the main safety issues within French fuel transport raised during the last years:

- The leaking fuel assembly transport;
- The double barriers design for spent fuel package;
- The double operations & controls associated to the water exclusion assumption.

2. LEAKING FUEL ASSEMBLY TRANSPORT

Cladding rupture can occur on irradiated fuels. These are then known as leaking fuels. These ruptures cause loss of confinement for the cladding and a loss of internal pressure, and therefore potentially allow water to penetrate inside the cladding. This water is subject to radiolysis under the influence of radiation, causing hydrogen generation. When transporting leaking fuels in the sealed cavity of a transport package, this hydrogen generation can cause hydrogen to accumulate at the top of the upper cavity to the lower limits of inflammability (LII) that are acceptable from a safety point of view.

It should be noted that in the case of the transport of spent fuel in France, transport is carried out with dry cavities, but the drying operations related to loading the assemblies underwater cannot guarantee the absence of water in the ruptured cladding, as water can remain trapped in the cladding due to capillary action during drying operations. Among other things, this phenomenon justifies the delivery, by the French authority, of approval of type B(M) for the transport of spent leaking fuel. The transport time is therefore limited to comply with safety demonstrations.

In December 2007, the French authority imposed the systematic measurement of hydrogen rates in the cavity for packages transporting at least one non-sealed fuel rod. From January 2008, TN International has been measuring rates of hydrogen upon arrival at the LA HAGUE treatment site for the whole of the TN12/2 and TN13/2 packages transporting fuel identified as ruptured. The measurements carried out on all shipments of ruptured fuel in 2008 gave rates of hydrogen far below the LII values (around 3%) for the packages' safety analysis reports, which are nevertheless penalising.

Upon request from the French authority, TN International carried out linear extrapolation for the generation of hydrogen, taking into account the real periods of transport and the associated measurements related to the regulatory periods of the conditions of approval which, other than the authorised transport periods, include a "variation" part, which is 7 days for transport in France, 15 days for continental transport and 21 days for intercontinental transport. For certain shipments carried out, this extrapolation lead to rates of hydrogen greater than the LII that were determined in the safety analysis reports, which led the competent French authority to classify the shipments concerned at level 1 of the INES scale and also to request changes to the shipment procedures for approvals that are currently valid.

The new shipment procedures consist of:

- Measuring the rate of hydrogen present in the upper part of the cavity before transport and at least two days after closing the cavity;
- Carrying out linear extrapolation according to the hydrogen generation time, to determine the maximum period before re-opening the confinement envelope so as not to exceed the LII determined in the safety analysis report.

These procedures nevertheless remain extremely restrictive, particularly the linear extrapolation.

The areas for improvement that are currently being studied by TN International are:

- Making the cavity inert before transport, using a neutral gas to increase the inflammability threshold. An extension request is currently being studied by the French authority. The acceptable LII in relation to transport under air (dependent, of course, on the temperatures of the packages in the various regulatory conditions of transport) changes, for example, from 3– 5% in a helium atmosphere;
- The justification of the penalising character of the linear extrapolation law using measurements of hydrogen rates upon arrival at the reprocessing plant, in addition to those taken on departure, and correlation using a law representing the hydrogen generation that is actually observed;
- The development of technological solutions aiming to reduce the concentration of hydrogen produced by radiolysis. One of the solutions envisaged is to introduce a catalytic recombiner into the cavity. The role of the recombiner is to catalyse the oxidation reaction of the hydrogen produced by radiolysis with the oxygen. The reaction is as follows $H_2+1/2O_2 \Rightarrow H_2O$. This type of solution has been implemented since 2007

on a TN17T package at TIHANGE (Belgium). A recombiner is used for the wet transfer of spent fuel. Introducing the recombiner into the package would stabilise the concentration of hydrogen, during transport, at a value of about 0.6%, while in shipments not fitted with recombiners, we see a constant increase in the concentration of H_2. The future objective for TN International is to implement this technology on a first dry transport of spent leaking fuel in France before long. Simultaneously, TN International is carrying out collaborative work with manufacturers and research centres on several technological solutions for reducing the concentration of H2 by trapping H2 or by catalytic recombination.

3. DOUBLE BARRIERS DESIGN FOR SPENT FUEL PACKAGES

Packages for the transport of radioactive material have to comply with national and / or international regulations. These regulations are widely based on the requirements set forth by the International Atomic Energy Agency (IAEA) in the “Regulations for the Safe Transport of Radioactive Material”. In this framework, packages to transport spent fuel assemblies have to meet the requirements for packages containing fissile material. The applicant for a package design approval shall demonstrate that the package remains sub-critical in all conditions of transport.

According to the latest edition of the IAEA transport regulation, the sub-criticality of a package may be demonstrated assuming water exclusion from the containment system, if and only if the design is based on a multiple high standard water barriers. It is widely accepted by Competent Authorities that a double watertight barrier design is enough to comply with this requirement. Application of this requirement to spent fuel package may result in different types of design.

In most cases, spent fuel packages are constituted of a thick forged vessel (a shell and a welded bottom) made of steel or cast iron, to lower the external radiation level around the package. Thus, an application of the double watertight barriers is to design a double lid system. Another way of application is to design a full double vessel cask.

3.1. Double-lid design

Figure 1 represents the principle of a double-lid design. The vessel is made of a thick shell in steel or iron cast, with a thick bottom welded to the shell by full penetration. The primary lid equipped with seal is insuring the first watertight barrier; the secondary lid equipped with seal is insuring the second watertight barrier.

Main advantage is the strength of the body, due to the thick shell in one-piece. Resistance to bending during side dropt test is high.

Other advantages are:

- High thermal conductivity through the shell allowing the heat dissipation of the content thermal power;
- Simplification of the design of drying orifice;
- Optimization of mass, dimensions of the package and so cost of the packaging.

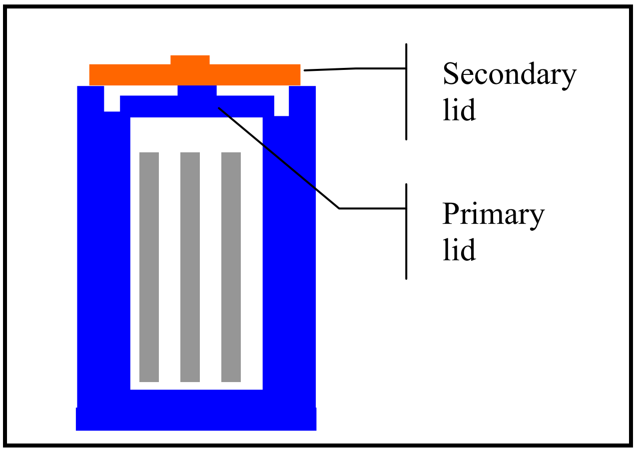

FIG. 1. Double-lid design principle.

The main drawback is to demonstrate the containment barrier when the material is subject to brittle fracture at low temperature.

3.2. Double-vessel design

Figure 2 represents the principle of a double-vessel design, based on two independent thin vessels, each one equipped with a welded bottom and a lid.

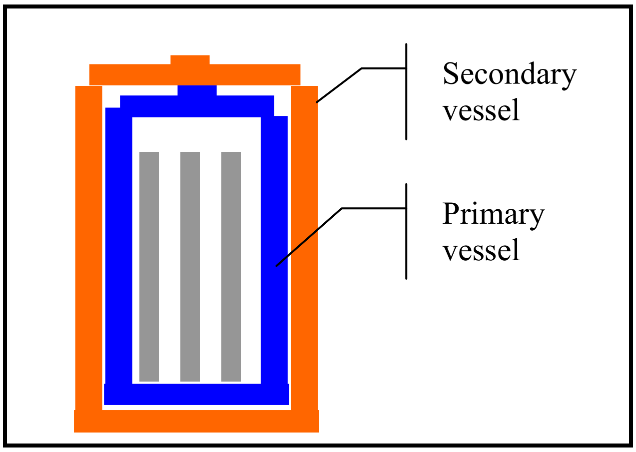

FIG. 2. Double-vessel design principle.

Advantages:

- Two complete different barriers.

Drawbacks:

- Compared to the double-lid design, the thermal conductivity is reduced because of the gap between the two vessels;

- In case of loading or unloading underwater operation, difficulties occur to design and perform drying orifices through the two barriers when the drying of the cavities before transportation is needed;
- Risk of retention of water between the two cavities;
- Increase of the mass and the size of the package, compared to the double-lid design (each barrier have to satisfy to the safety conditions regarding drop test for example);
- Manufacturing more difficult than a double-lid design.

3.3. Conclusion

Depending on the use and the performance level of the package, the double watertight barriers required by the transport regulation to consider water exclusion in the criticality analysis may be implemented either by a double-lid or a double-vessel design. Nevertheless a double — vessel design, especially when loading is under water, is more complicated to design, manufacture and to use than a double–lid design and does not represent a real improvement of the safety level compared to the double–lid design.

Both designs have been licensed by the French competent authority according to the 2005 edition of the IAEA transport regulations.

4. DOUBLE OPERATIONS & CONTROLS ASSOCIATED TO THE WATER EXCLUSION ASSUMPTION

As explained previously, and according to paragraph 677 of the IAEA transport regulation, sub-criticality of package design shall be demonstrated assuming water leaking within all void spaces of the packaging, except *if the design incorporates special features to prevent such leakage of water into or out of certain void spaces, even as a result of error.*

Such feature may be either a single barrier design licensed according to the 1985 edition of the IAEA transport regulation and submitted to multilateral approval, or a multiple barrier design licensed according to the 2005 edition of the IAEA transport regulation.

In both cases, these features are based on design components (leak tight sealing…) that we may call “Confinement Components”, by analogy with the Confinement System defined in the regulation as the fissile content and the packaging components intended to preserve the criticality safety.

However, to be efficient regarding water exclusion (i.e. presence of water that does not result from the regulatory tests), the confinement components are required to be correctly used during operations, such as draining and drying the package cavity, when packages are loaded and/or unloaded in pool. Confinement operation may be defined as any operation that may result in reducing the efficiency of a Confinement Component as a result of error, standing for a human error or a tool failure.

To prevent common cause error, French competent authority requires that confinement operations need to be performed twice independently (different operators and tools).

This paper describes typical critical operations for package, and the double controls system associated.

To be noticed that in case of sub-critiatility demonstration based of a complete flooding of all the void space, double controls system are not required.

4.1. Case of a single barrier design

Most of designs confinement components are:

- A shell and a welded bottom;
- A lid equipped with seal (elastomer or metallic) and secured by bolts;
- Orifice covers equipped with seal (elastomer or metallic) and secured by bolts.

Confinement operations concerned by the water exclusion are:

- Draining and drying of the cavity;
- Fastening of the bolts to the required torque;
- Check of the leak tightness of the lid and orifice seals before shipment.

4.2. Case of a double barrier design

Most of designs confinement components are:

- A thick shell and a welded bottom, or two shells and their bottom as discussed previously;
- Two lids equipped with seals (elastomer or metallic) and secured by bolts;
- Orifice covers equipped with seal (elastomer or metallic) and secured by bolts.

By nature, the presence of a double barrier system prevents from one error to result in a water in-leakage in the package. Then, the only confinement operation is the draining/drying operations.

However, fastening of the bolts and leak tightness check of the two lids should not be considered as confinement operation when implemented by different operators and tools. In that case, no common cause error may result in the leakage of both barriers.

It is noticeable that multiplying operations and controls has a significant effect on the radiation dose of operators. Such a program should not be extended to excess.

4.3. Application to drying operations

For packages loaded in pools, water is removed from the cavity by draining then vacuum drying. To prevent ice formation, the vacuum pressure shall be maintained over 6 mbar abs. the drying criteria are usually a pressure rise limit during a short period of time.

Therefore, control of the pressure inside the cavity is the key operation to survey. A double control system may be implemented by measuring continuously the pressure in the cavity through a double recording measuring device. Then, a different operator from the one performing the test may check the pressure level and the pressure rise on the recordings of the second device. In such process, neither a measuring device flaw nor a human error may occur and jeopardize the drying operation