TopFuel

REACTOR FUEL PERFORMANCE 2012

Transactions

Manchester, United Kingdom
2 - 6 September 2012

TopFuel 2012 Gold Sponsor

Westinghouse

EUROPEAN NUCLEAR SOCIETY

ENS CONFERENCE

organised in cooperation with:

AMERICAN NUCLEAR SOCIETY

ATOMIC ENERGY SOCIETY OF JAPAN

CNS

IAEA
International Atomic Energy Agency

AEN NEA

European Nuclear Society
Rue Belliard 65
1040 Brussels, Belgium
Phone + 32 2 505 30 54
Fax +32 2 502 39 02
E-mail ens@euronuclear.org
Internet www.euronuclear.org

ISBN 978-92-95064-16-4

These transactions contain all contributions submitted by 7 September 2012.

Spent Fuel and Transportation

OXIDATION STUDIES ON IRRADIATED UO2 FUELS	Papaioannou, D. (1); Hollas, S. (1); Rondinella, V. (1); Sasahara, A. (2) 1 - European Commission, Joint Research Centre, Institute for Transuranium Elements, Germany 2 - Central Research Institute of Electric Power Industry, Japan
RESULTS OF THERMAL CREEP TESTS ON IRRADIATED ZRY-2	Lloret, M. (1); Quecedo , M. (1); Conde , J. M. (2); Rey , J. M. (2); Alejano, C. (2); Fernandez, F. J. (3) 1 - ENUSA Industrias avanzadas SA, Spain 2 - Consejo de Seguridad Nuclear, Spain 3 - ENRESA, Spain
DRY STORAGE RELIABLE SOLUTIONS FOR THE MANAGEMENT OF SPENT NUCLEAR FUEL IN THE LONG TERM	Issard, H. (1) 1 - AREVA TN International, France
FRAPCON-3.4 EXTENSION TO MODEL HYDROGEN EFFECTS ON CLADDING MECHANICAL BEHAVIOR DURING DRY STORAGE	Feria, F. (1); Herranz, L. E. (1) 1 - CIEMAT, Spain
STUDSVIK CMS CAPABILITY FOR SPENT NUCLEAR FUEL	Grandi, G. (1); Simeonov, T. (2) 1 - Studsvik Scandpower, Inc, United States 2 – Studsvik Scandpower, GmbH, Germany
CRITICALITY SAFETY ANALYSIS OF FRESH AND SPENT FUEL STORAGE AND HANDLING FOR VVER REACTOR UNIT USING CODE MCNP5	Slugen, V. (1); Farkas, G. (1); Hascik, J. (1); Vrban, B. (1); Lulej, J. (1); Petriska, M. (1); Hinca, R. (1); Urban, P. (2) 1 - Slovak University of Technology, Faculty of Electrical Engineering and Information Technology, Slovakia 2 - SE a.s., NPP Mochovce, Slovakia
IAEA'S COOPERATION IN SUPPORTING LONG TERM INTERIM STORAGE AND BACK-END MANAGEMENT OF RESEARCH REACTOR FUEL	Tozser, S. (1); Pablo, A. (1); Bradley, E. (1) 1 - International Atomic Energy Agency, Austria
RETHINKING SPENT FUEL MANAGEMENT, A POST-FUKUSHIMA PERSPECTIVE	Chiguer, M. (1); Lelievre, F. (1); Froment, A. (1) 1 - AREVA, France
REVIEW OF QUALIFICATIONS FOR FUEL ASSEMBLIES FABRICATION	Slabu, D. (1); Zemek, M. (1); Hellwig, C. (1) 1 - Axpo AG, Switzerland
IMPACT OF THE WWER FUEL ASSEMBLY WITH THE REACTOR CORE BAFFLE DURING THE SIMULATION OF THE SEISMIC IMPACT	Makarov, V. V. (1); Afanasiev, A. V. (1); Matvienko, I. V. (1); Tutnov, A. A. (2); Kiselev, A. S. (2); Krutko, E. S. (2); Volkov, S. V. (3) 1 - OKB "HYDROPRESS", Podolsk, Russian Federation 2 - NRC "Kurchatov institute", Moscow, Russian Federation 3 - JSC "TVEL", Moscow, Russian Federation

Spent Fuel and Transportation

OXIDATION STUDIES ON IRRADIATED UO_2 FUELS

D. PAPAIOANNOU, S. HOLLAS, V.V. RONDINELLA
European Commission, Joint Research Centre, Institute for Transuranium Elements, P.O. Box 2340, 76125 Karlsruhe, Germany

A. SASAHARA
Central Research Institute of Electric Power Industry, 2-11-1 Iwado-kita, Komae, Tokyo 201-8511, Japan

ABSTRACT

UO_2 oxidation has been examined since the early age of nuclear energy civilian applications. Most studies refer to unirradiated analogues or low burnup spent nuclear fuels. Unirradiated or low burn up UO_2 oxidation follows the scheme $UO_2 \rightarrow U_3O_7 \rightarrow U_3O_8$ having all oxides present during the process, while fuels with burnup > 35 GWd/tHM are initially converted to U_4O_9 showing a certain resistance to further transformation. The final product U_3O_8 has a different crystal structure (orthorhombic) than the cubic (fcc) UO_2 and U_4O_9 phases; its formation is associated with a 36% volume expansion. Oxidation to U_3O_8 would negatively affect the structural integrity of the fuel rod, compromising handling and storage of the fuel after discharge from the reactor. An originally small clad defect can turn into a big crack due to fuel oxidation and swelling. Fuel fragmentation would result in larger surface areas available for corrosion processes and radionuclide release.

Information on the oxidation behaviour of high burnup UO_2 fuel, representative of the current burnup trend in nuclear power plants, is relatively limited in the literature. This paper reports on oxidation experiments carried out on BWR UO_2 with nominal burnup of 50 and 65 GWd/t, respectively. Fuel specimens were oxidized at 300, 350 and 400°C in air in a hot cell autoclave, to characterize the transformation of UO_2 into U_3O_8. Weight gain data were combined with x-ray diffraction analysis for phase identification. Selected specimens were subjected to microscopy examination to determine extent and effects of the oxidation process on phase distribution and overall morphology of the fuel. High burnup UO_2 oxidises initially to a cubic phase resembling U_4O_9, but with a composition beyond its stoichiometry, closer to $UO_{2.4}$. The new phase, a derivative of the initial cubic UO_2 with slightly higher density, forms very rapidly along grain boundaries and grows into the grains. The process is controlled by oxygen diffusion to the grain interior; the reaction rate follows a parabolic kinetic law. The U_4O_{9+x} is stabilised by fission products present in the matrix, with a characteristic plateau (whose duration depends on the oxidation temperature) occurring at bulk oxygen to metal ratio ~2.45. Afterwards, the oxidation proceeds with the formation of the final U_3O_8. A nucleation-and-growth mechanism displaying sigmoid reaction kinetics was observed.

1. Introduction

The safe handling and storage of used fuel is one of the major issues in nuclear energy production. Interim dry storage of spent fuel over extended periods has been implemented worldwide as a safe and economically sound solution. The dry storage temperature could exceed 200°C for LWR UO_2 and 300°C for MOX fuel [1, 2]. These temperatures may accelerate the degradation mechanisms of initially slightly defective spent fuel rods. Infiltrated air inside the fuel rod at temperatures ranging between 200 and 400°C will transform the UO_2 to higher oxide structures up to complete oxidation and formation of the friable U_3O_8 [1-10]. The conversion is associated with 36% net volume expansion; due to this significant swelling, the structural integrity of the rod's cladding material is further deteriorated. This could eventually result in complete failure of the cladding. Possible fuel cracking would expose larger fuel surface areas to air, potentially enhancing the release of radiotoxic species.

The oxidation of UO_2 has been and still is widely studied since the early age of nuclear energy [3-12]. Nevertheless, many important aspects of the oxidation process are not yet fully understood, as reported e.g. by McEachern and Taylor [13]. In their extensive review on the oxidation behaviour of UO_2 at temperatures up to 400°C, the outcome of several studies on the influence of temperature, moisture, presence of dopants, oxygen partial pressure, radiolysis of the gas atmosphere, density, as well as particle and grain size of the fuel is discussed.

It is generally accepted that unirradiated or low burn up UO_2 (≤15 GWd/tHM) follows the oxidation scheme $UO_2 \rightarrow U_3O_7 \rightarrow U_3O_8$ and proceeds with the initial formation of a product layer on the fuel surface; transitions occur at temperatures above 250°C [14]. Fuels with burn up higher than ~35 GWd/tHM is initially converted to an intermediate cubic phase resembling U_4O_9, but with an oxygen to metal ratio (O/M) closer to $UO_{2.4}$ [1-14] than the nominal $UO_{2.25}$. The new phase is formed very rapidly along the grain boundaries, and grows into the UO_2 grains, showing a certain degree of resistance to further transformation. After longer oxidation times, for both fresh and spent fuel the oxidation results to the same final product U_3O_8. At temperatures below 400°C U_3O_8 is stable, but friable. It is characterized by a distinctly different crystalline structure (orthorhombic) than the phases UO_2, U_4O_9 (cubic, fcc) or U_3O_7 (tetragonal).

Many studies provided kinetic data on simulated or low burn up spent nuclear fuels. Information on the oxidation behaviour of UO_2 fuel above 60 GWd/tHM is relatively scarce, in spite of the recent tendency in the nuclear power industry is to increase the fuel burnup. In this study we performed isothermal oxidation experiments in dry air and at several temperatures of two spent BWR UO_2 fuels with nominal burn ups 50 and 65 GWd/tHM, respectively. Some preliminary results obtained on 39 and 53 GWd/tHM UO_2 during initial testing of the autoclave equipment are also reported in this paper as indicative information.

2. Experimental

2.1 Specimens

Spent LWR UO_2 fuels, discharged from commercial nuclear power plants, were used for the oxidation experiments. Table 1 summarizes the main experimental parameters of all the tests. The main programme was carried out on fragmented fuel pellets with burn ups of 50 and 65 GWd/tHM. Fragments with typical size ~ 3-4 mm, produced by crushing decladded fuel discs 3-4 mm thick, were subjected to oxidation at 300, 350 and 400°C. One experiment was performed at 400°C on powder from the 65 GWd/tHM fuel with grain sizes between 30 and 60 μm, produced by milling of fragments and subsequent sieving.

T, °C	Fuel			
	31.5 GWd/tHM	**53 GWd/tHM**	**50 GWd/tHM**	**65 GWd/tHM**
268	Fragments	Fragments	-	-
300			Fragments	Fragments
350			Fragments	Fragments
400			Fragments	Fragments, powder

Tab 1: Temperature, fuel burn up and morphology adopted for the oxidation experiments

In addition to the main oxidation campaign, 31.5 and 53 GWd/tHM UO_2 samples were oxidized at relatively low temperature (268°C) to examine the morphology changes corresponding to the onset of the oxidation process.
The burnup of the fuels examined was determined both by longitudinal γ-scanning spectroscopy of the delivered fuel rods and by fuel chemical analysis in the frame of other post irradiation examination campaigns.

2.2 Equipment

A heating apparatus was designed, adapted for manipulator use and installed in hot cell for the oxidation studies. The system can be operated at temperatures up to 400°C under different oxidising atmospheres. For our experiments dry air has been used as oxidative medium. The fuel fragments were placed in pre-weighed, oxidation-stable ceramic crucibles; at given time intervals the oxidation was interrupted to record the weight change of the crucibles and their content.

Fig.1 shows the autoclave system before installation in hot cell. the setup has the advantage that during heating or cooling down ramps the specimens can be kept in the chamber under inert or vacuum atmosphere. The oxidation during the cooling or heating of the specimens (lasting typically 0.5-2 h) can thus be avoided. As a result the oxidation process can be studied only at the desired temperature. This solution permits minimizing the uncertainties associated with the determination of the oxidation time, which is essential to analyze steep changes in the weight gain diagram.
The setup consists of a tight cylindrical chamber equipped with valves for gas input-output, vacuum connections and thermocouple. Homogeneous constant temperature conditions within the chamber are guaranteed by the round, symmetric geometry of the furnace and the test chamber, and by the metallic holder for the crucibles. The temperature is measured in one of crucibles filled with alumina powder. After oxidation times corresponding to significant weight changes, the content of one or more crucibles is removed from the furnace and sent for supplementary examination, for instance x-ray diffraction (XRD), Scanning Electron Microscopy (SEM) or optical microscopy.

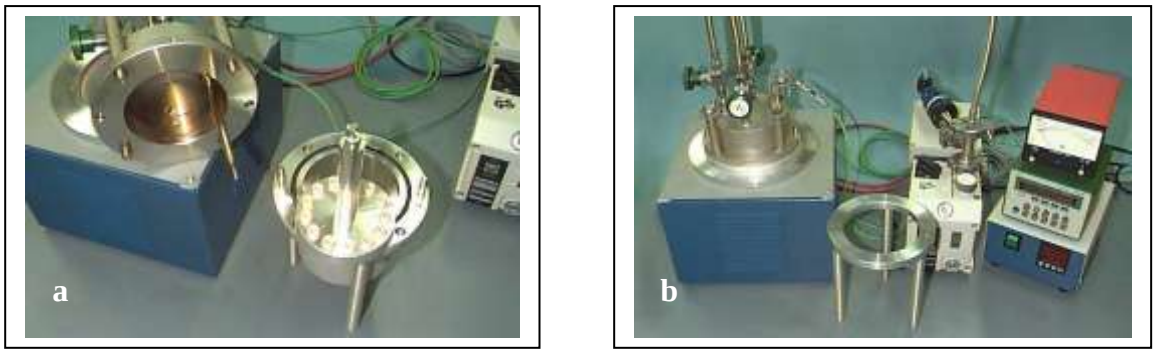


Fig 1. The experimental set-up for long term oxidation experiments. The spent fuel samples are placed in the pre-weighed ceramic crucibles inside the testing chamber (a); sealed autoclave ready for the oxidation tests (b).

3. Results and discussion

The results of the oxidation experiments confirm the stepwise oxidation of UO_2 with the characteristic plateau corresponding to bulk O/M level of 2.45 visible in all low temperature graphs.
Fig. 2 shows the weight change or bulk O/M ratio as a function of time for the 31.5 and 53 GWd/tHM samples oxidized at 268°C. The data of Fig. 2 include only the first oxidation step $UO_2 \rightarrow U_4O_{9+x}$ and are very consistent by comparing the two specimens, in spite of the different burnup. These results are also matching well oxidation curves of irradiated fuels reported in the literature [15]. The SEM micrographs shown in Fig. 3 illustrate the morphology changes corresponding to the progression of the oxidation process. Initially, the fuel grains are oxidised along the grain boundaries. An oxidized layer forms very rapidly and further oxidation is controlled by oxygen diffusion to the interior of the grains through the outer oxide layer. The reaction rate of this internal oxidation of irradiated UO_2 grains follows parabolic kinetics. The intermediate phase U_4O_{9+x}, a derivative of the initial cubic UO_2 with slightly higher density, is stabilised by the fission products present in the matrix [6-8, 11-14] and the characteristic plateau corresponding to bulk O/M level >2.40 appears in the oxidation curve. The duration of this plateau tends to decrease with increasing oxidation temperature. The mechanism by which the fission products or dopants present as a solid solution in the UO_2 matrix induce the stabilisation of the cubic U_4O_{9+x} phase remains poorly understood. Remarkable is the high initial rate of oxidation in comparison to unirradiated fuel. The higher surface area due to the presence of cracks, pores and grain boundary porosity which accelerates the oxygen diffusion and rapid transport may be the main reason for this behaviour.
Figs. 4, 5 and 6 show the weight change or bulk O/M ratio as a function of time for the 50 and 65 GWd/tHM samples at increasing oxidation temperature. Our data confirm the stepwise oxidation

of UO_2. By increasing oxidation temperature the plateau of the U_4O_{9+x} phase in the oxidation curves becomes shorter (see Fig.5 and 6(a)); it completely disappears in the case of the 65 GWd/tHM (Fig. 6(b) and (c)). Afterwards, the oxidation proceeds with the formation of U_3O_8, following a nucleation-and-growth mechanism [2,13,16] and displaying sigmoid reaction kinetics. This is best shown in the oxidation curves corresponding to the experiments of both 50 and 65 GWd/tHM fuels at 300°C (Fig. 4). The final friable product U_3O_8 is stable at temperatures below 400°C and has a distinctly different crystal structure (orthorhombic) than the cubic (fcc) phases for UO_2 and U_4O_9. Its formation is completed at nominal O/M ratios of about 2.70-2.75 and has a density of 8.35 g/cm^3, significantly lower than the values for UO_2 and U_4O_9, 10.96 and 11.30 g/cm^3, respectively. This significant density change, which corresponds to 36% volume expansion, can produce fuel fragmentation and pulverisation.

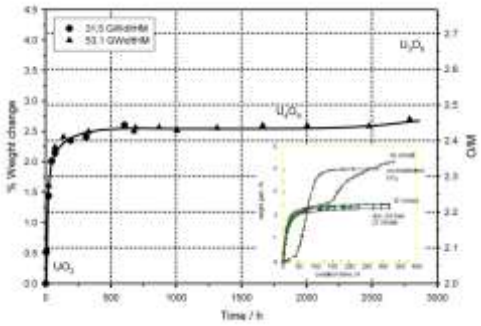


Fig 2. Oxidation of spent fuel fragments in dry air at 268°C. Literature data at 275°C from [15] are shown in the inset for comparison.

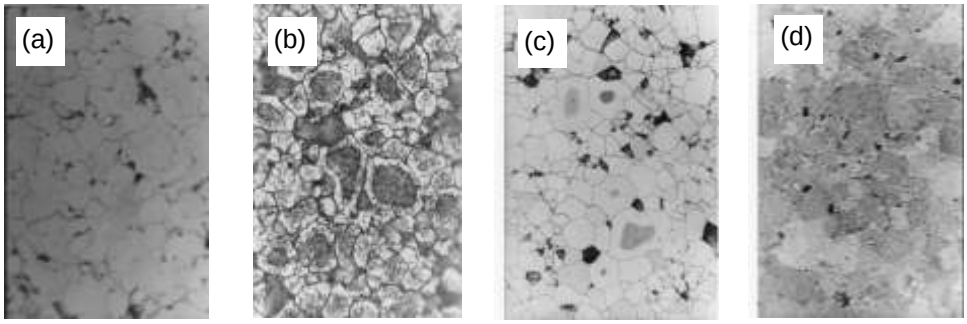


Fig 3. Internal oxidation of irradiated UO_2 (burn-up 53 GWd/t) during exposure to dry air at 268°C. The sequence of micrographs illustrates the rapid growth of U_4O_9 along grain boundaries: (a) initial fuel, (b) after 41 h, (c) after 48 h and (d) after 530 h.

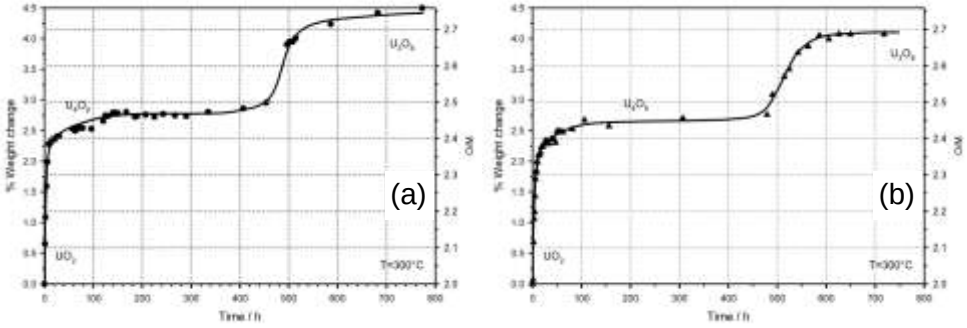


Fig 4. Weight gain of UO_2 fuel at T=300°C as a function of time
(a): 50 GWd/tHM, (b) 65 GWd/t

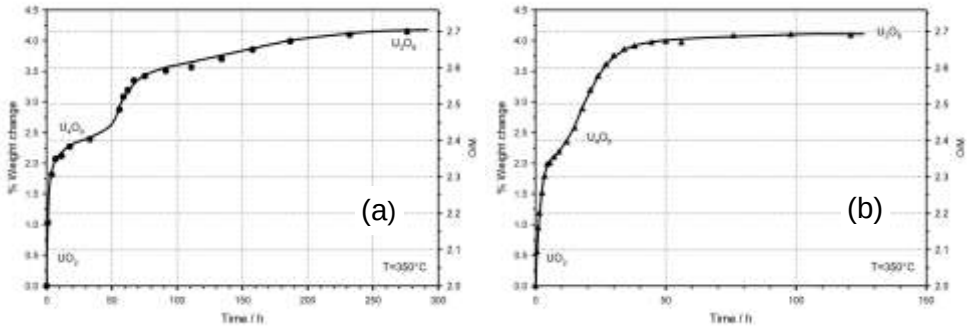


Fig 5. Weight gain of UO_2 fuel as a function of time at T=350°C
(a): 50 GWd/tHM, (b) 65 GWd/t

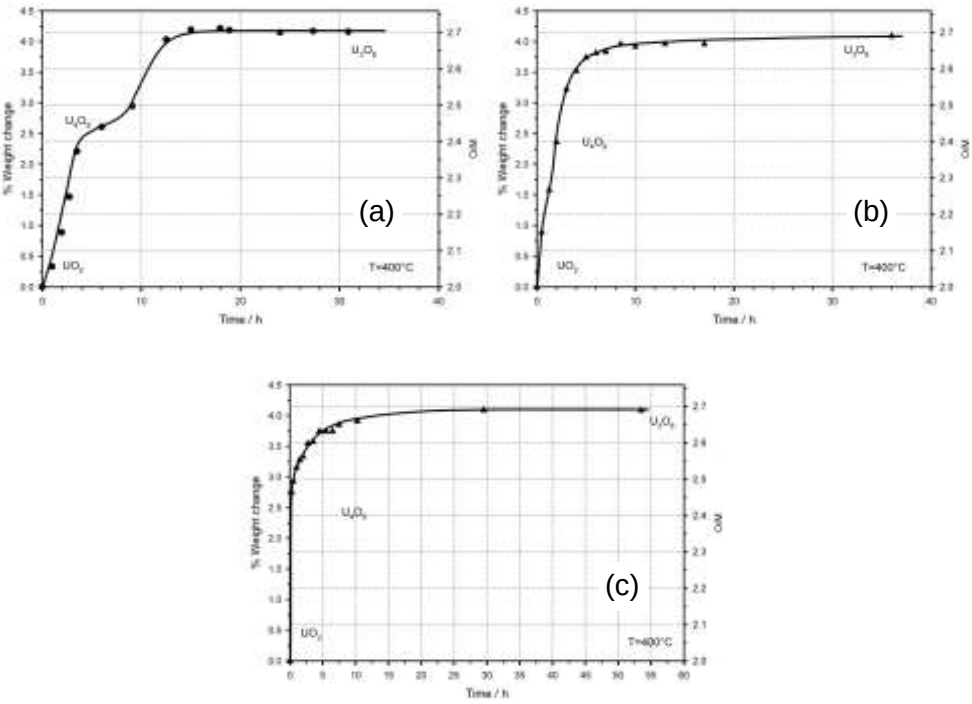


Fig 6. Weight gain of UO_2 fuel as a function of time at T=400°C; (a): 50 GWd/tHM, (b) 65 GWd/tHM, and (c) 65 GWd/tHM powder with grain size <56 µm.

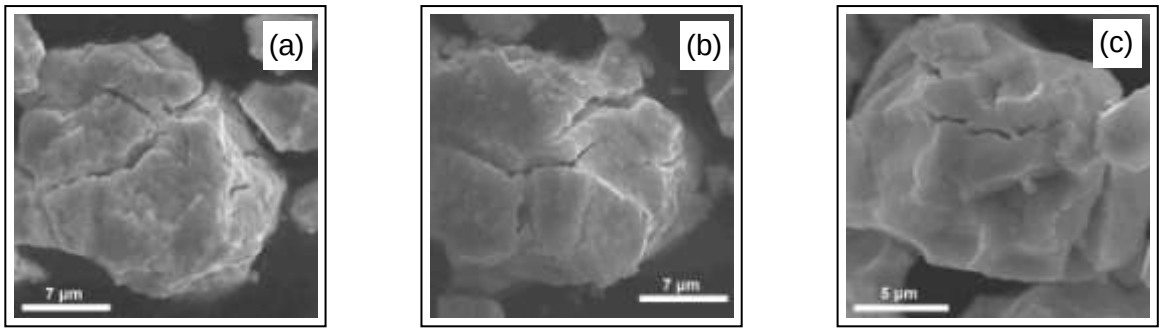


Fig 7. SEM micrographs of the 50 GWd/tHM fuel, oxidised (a) for 912 h at 300°C with bulk O/M 2.771; (b) for 157.8 h at 350°C with bulk O/M 2.650; (c) for 19.98 h at 400°C with bulk O/M 2.707

The SEM micrographs of Fig 7 illustrate the final outcome of the oxidation process: grains of the 50 GWd/tHM fuel oxidised at several temperatures with final O/M ratios close to 2.7 are shown, revealing the same cracked morphology.
Fig 8 shows XRD spectra acquired with a θ/θ diffractometer on samples of the 50 GWd/tHM fuel after oxidation at 400°C. The phases appearing in the spectra are the initial cubic UO_2 material, the intermediate and also cubic oxide U_4O_9 and the orthorhombic final oxidation product U_3O_8. Although the UO_2 and U_4O_9 give basically similar X-ray patterns, the two phases can be easily distinguished by the characteristic broadening and shifting to higher angles (slightly smaller lattice parameter, i.e. more dense phase) of the U_4O_9 peaks. U_3O_8 peaks were detected for first time at nominal O/M 2.44, in agreement to the oxidation mechanism and corresponding weight gain curve shown in Fig. 6 (a). By increasing oxidation time up to O/M 2.708, the cubic phase peaks are gradually decreasing, till only U_3O_8 peaks remain.

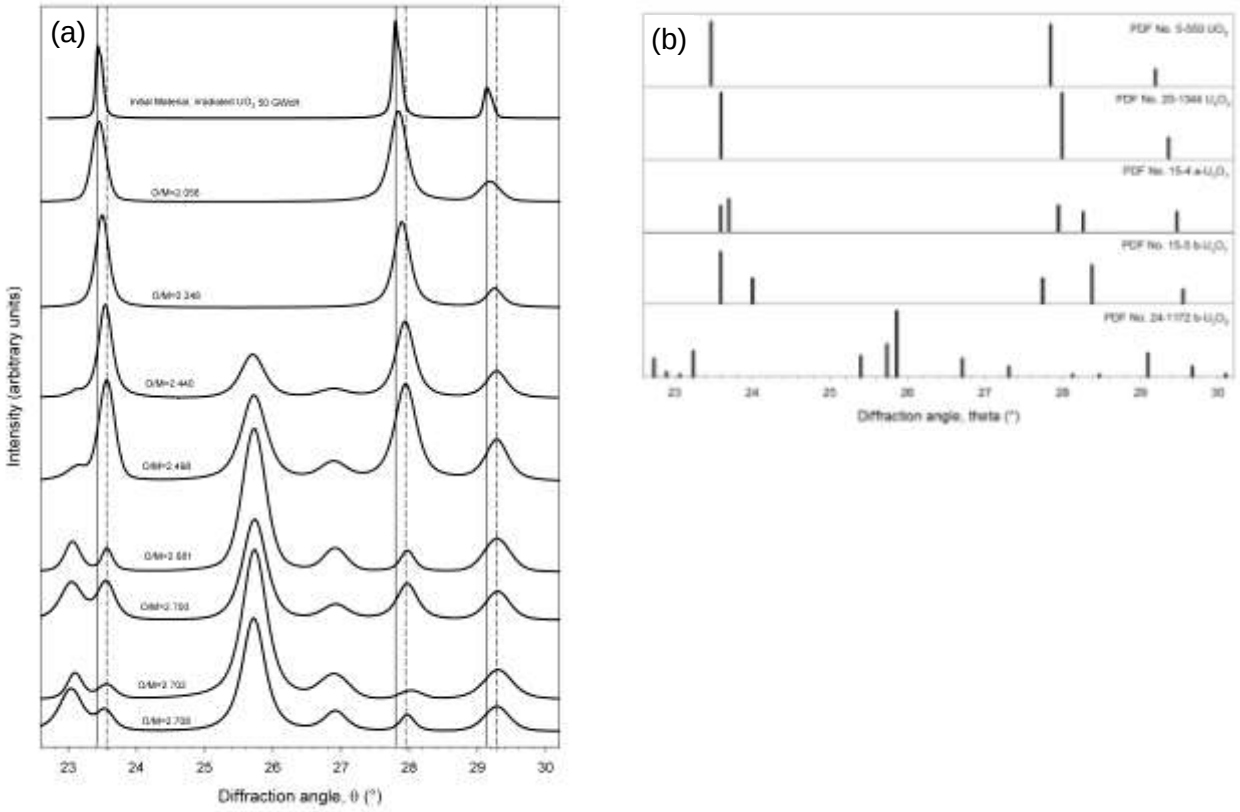


Fig 8. (a) X-ray spectra of spent fuel samples with 50 GWd/t oxidised at 400°C. The solid lines denote the peak positions of UO_2 and the dashed lines those for U_4O_{9+x}. The remaining peaks in the spectra correspond to U_3O_8 peaks; (b) data from the JCPDS data base show reference peak positions and relative intensities for various non irradiated uranium oxide phases.

As last in this section, we mention the higher oxidation rates of the 65 GWt/tHM compared to the 50 GWt/tHM. With exception for the lower temperature data at 300°C, where both fuel behaved similarly and the U_3O_8 formation began at oxidation times between 400-500 h, at 350° and 400°C the 65 GWt/tHM UO_2 fragments oxidised faster than those from the 50 GWd/tHM fuel. This is most probably due to the higher surface area exposed to the dry air for the 65 GW/tHM fuel, in particular due to inter- and intra-granular porosity, which could speed up oxygen diffusion and formation of the U_4O_{9+x} phase. This occurred very rapidly at 400°C (within ~2 h for the 65 GW/tHM against 8 h for the 50 GW/tHM fuel). At 400°C, after reaching the O/M 2.4 level, the $U_4O_9 \rightarrow U_3O_8$ transition required equivalent oxidation time for both fuels, approximately 9 h. The effect of surface area is confirmed by comparing the oxidation curves in Fig 6(b) and (c): the powder samples (Fig. 6 (c)) are oxidised more rapidly than the fragments.

4. Conclusion

The oxidation rates of highly burnt irradiated UO_2 LWR fuel up to the final product U_3O_8 under conditions relevant for dry spent fuel storage were studied and provided new data. Low temperature oxidation tests allowed examining the growth of a higher oxide starting at grain boundaries and growing towards the interior of the grain. This process is controlled by oxygen diffusion through the oxidized layer. The oxidation curves of fuels with burnup of 50 and 65 GWt/tHM at 300, 350 and 400°C confirmed the formation in two step mode of the U_3O_8 as final product. The irradiated UO_2 oxidises very rapidly to the intermediate cubic phase U_4O_9 with hyperstoichiometric composition near $UO_{2.4}$, which is stabilised by the fission products in the matrix for period whose duration decreases with increasing oxidation temperature. The fuel with 65 GWd/tHM is oxidised faster than that with 50 GWd/tHM fuel at temperatures above 350°C, indicating a possible effect due to differences in surface area caused by different burnup levels.
These studies are part of a series of examinations aimed at determining the key mechanisms and properties responsible for the oxidation behaviour of spent fuel at low temperature.

5. References

[1] Gil-Sung You, Keon-Sik Kim, Duck-Kee Min, Seung-Gy Ro; J. Nucl. Mater., 277 (2000) 325.
[2] G. Rousseau, L. Desgranges,F. Charlot , N. Millot , J.C. Niepce , M. Pijolat , F. Valdivieso, G. Baldinozzi , J.F. Berar; J. Nucl. Mater., 355 (2006) 10.
[3] S. Aronson, R.B. Roof Jr., J. Belle, J Chem. Phys. 27 (1957) 137.
[4] J. Belle (Ed.), Uranium dioxide: properties and nuclear applications, Naval Reactors, Division of Teactor Development, USAEC, 1961.
[5] M. J. Bannister, J. Nucl. Mater. 26 (1968) 174.
[6] L. E. Thomas, O. D. Slagle, R. E. Einziger, J. Nucl. Mater.184 (1991) 117.
[7] P. Taylor, D. D. Eood, A. M. Duclos Mate, J. Nucl. Mater.189 (1992) 116.
[8] R. E. Einziger, L. E. Thomas, H. C. Buchanan, R. B. Stout; J. Nucl. Mater., 190 (1992) 53.
[9] K. J. Schneider, S. J. Mitchell, A. B. Johnson Jr., Proc. 3rd Int. Conf. on High Level Radioactive Waste Management, Las Vegas, 1992, p. 1159.
[10] K. M. Wasywich, W. H. Hocking, D. W. Shoesmith, P.Taylor, Nucl. Technol. 104 (1993) 309.
[11] L. E. Thomas, R. E. Einziger, H. C. Buchanan; J. Nucl. Mater., 201 (1993) 310.
[12] L. E. Thomas, R. E. Einziger and R.E. Woodley, J. Nucl. Mater., 166 (1998) 243.
[13] R. J. McEachern, P. Taylor, J. Nucl. Mater., 254 (1998) 87-121.
[14] B. D. Hanson, PNNL-11929, (1998).
[15] K.M. Wasywich, 3rd IAEA RCM, Paris, (1995).
[16] J. Rouyer, A. Poulesquen, L. Desgranges, C. Ferry, J. Nucl. Mater., 395 (2009) 89.

RESULTS OF THERMAL CREEP TEST ON IRRADIATED Zry-2

M. LLORET, M. QUECEDO
Product Engineering, ENUSA Industrias Avanzadas
Santiago Rusiñol 12, 28040 Madrid – Spain

J. M. CONDE, C. ALEJANO, J. M. REY GAYO
Consejo de Seguridad Nuclear
Pedro Justo Dorado Dellmans 11, 28040 Madrid – Spain

F.J. FERNÁNDEZ,
HLW Engineering Department, ENRESA
Emilio Vargas 7, 28043 Madrid – Spain

ABSTRACT

This paper presents the thermal creep tests under internal pressure and the post-test characterization performed on Zry-2 material from a GE-14 rod manufactured by ENUSA and irradiated in Forsmark 3 up to a rod average burnup of 41.3 MWd/kgU. This program has been performed by the CSN, ENRESA and ENUSA in collaboration with Vattenfall, in order to investigate the behaviour of irradiated BWR cladding material under dry cask storage and transport conditions. Therefore, the test parameters have been selected to represent the limiting conditions during storage and transport: the maximum temperature allowed by ISG-11 (400ºC) and a lower value (360ºC) to analyze the effect of the temperature; and the corresponding internal pressures to produce hoop stresses between 100 and 190 MPa. The results show coherence between the test parameters and the measured creep rates as well as a significant effect of the temperature decrease. The results have been compared with similar tests performed in the past with irradiated ZIRLO and, as expected, the creep rate for RX material is lower than the creep rate for CWSR material.
In addition, in order to investigate hydride reorientation, a special cool-down procedure has been performed for every sample. After evaluating the hydride distribution by metallography and fn factor, Ring Compression Tests (RCTs) have been performed to analyze the hydride distribution effect on the material ductility.
Axial Tensile Tests (ATTs) have been performed also on the irradiated material before and after creep testing. A significant difference in yield strength and elongation has been observed. This difference could be related to the irradiation damage annealing due to the high temperature (360ºC) and long duration (800 hours) of the creep test.

1. Introduction

The strategy followed in Spain for the spent fuel management, except for a limited amount of fuel that was reprocessed in the past, is the open cycle. Currently, Spain has eight nuclear reactors in operation (6 PWRs, 2 BWRs). The spent fuel is stored in the respective spent fuel pools, and due to the limited capacity of these pools together with the decommissioning of the José Cabrera NPP, the interim dry storage of the discharged fuel assemblies has been already initiated. There are currently two interim storage facilities operating at reactor sites, and another one in construction.

Additionally, the generic design for a centralized dry spent fuel and high level waste interim storage facility (ATC in Spanish acronym) was approved by the Regulatory Authority (CSN) in 2006 and, at the end of 2011, the Government decided the site for the facility. The installation is based in a vault system, where the fuel assemblies are stored in capsules; these capsules are inserted in wells and cooled by natural circulation.

Fuel conditions during dry storage period are completely different than during pool storage. The differences between pool conditions, where clad temperature is around 50ºC with an external pressure due to at least 7 meters of water, and dry storage conditions, where clad temperature goes up to 400ºC, limit imposed by regulation [1] and there is a relevant difference between internal and external fuel rod pressure, should be taken into account to analyze the mechanical behavior of the fuel rods.

The higher fuel temperatures will increase the cladding tensile loading due to the increment in rod internal pressure. Phenomena such as irradiation hardening annealing, hydrides dissolution and creep, should be analyzed. Additionally, the temperature and pressure evolution during transients such as thermal cycles for cask atmosphere drying, or during long term storage, may re-precipitate in the radial direction the partially dissolved hydrogen picked-up during in-reactor operation, as a consequence of the temperature reduction under stress. In those conditions, the clad could become brittle. Additionally, the clad behavior during a spent fuel transport considering transport accident conditions is also an important issue, taking into account that the transport will be performed at lower temperatures, with lower cladding ductility as T decreases.

To simulate a transport accident, ATT has been selected as a representative test for the bending loads supported by the rod material between grids; and RCT as representative of the pinch loads transmitted by the grid to the rod/grid contact point.

Therefore, considering the elevated cladding temperatures and stresses that may be achieved, the thermal cycles during storage period, and the loads that will be transmitted from the fuel assembly grids to the rods during a transport accident, the thermal creep of the cladding and the mechanical behavior of the rod during bending and pinch loads are relevant considerations in assessing the fuel rod integrity during dry storage and transport.

Spanish organizations CSN, ENRESA and ENUSA have carried out a test program to characterize the behavior of spent fuel under dry storage and transport conditions. In the past, the program was focused on PWR advanced cladding material, ZIRLO, irradiated up to a high burnup. Due to the interesting results obtained with ZIRLO, a new program with BWR fuel clad material was initiated, and the results, obtained in December 2011, are presented in this paper.

2. Test Materials

The tested material was obtained from a GE-14 (10 mm outer diameter) fuel rod manufactured by ENUSA, using standard Zry-2 clad with liner in recrystallized condition, RX. This fuel rod was irradiated during 5 cycles in Forsmark-3 up to a rod average burnup of 41.3 MWd/kgU. After the irradiation period, the rod was extracted from the fuel assembly and shipped to the Studsvik hot cell laboratory for a comprehensive examination before performing the tests: oxide thickness, visual inspection, metallography, micro hardness and cladding H-analysis. Additionally one Axial Tensile Test (ATT) and two Ring Compression Tests (RCT) were also performed.

The oxide thickness of the rod was about 20 microns with an average hydrogen content of 200 ppm, following the typical BWR liner material distribution, the highest content at the periphery and at the liner /Zry-2 interface. Five samples of 120 mm were cut to manufacture the creep samples. As rod oxide thickness was very homogeneous, the axial position of the samples in the rod is not a significant parameter.

3. Creep and Mechanical Tests

3.1. Test Conditions

The creep tests were done in the Studsvik Mechanical Laboratory using a pressure-servo controlled system to keep a constant specimen internal pressure. In order to minimize the specimen oxidation at the elevated tests temperatures, Argon was used as pressurizing medium and for the test chamber purge. The internal pressure was specified for each test in order to obtain the target hoop stress considering the cladding wall thickness reduction due to the in-reactor corrosion layer of each specimen. The specimen is placed in a furnace and the deformation is monitored in mid-sample position using a laser system.

Test conditions were selected to yield data on secondary creep rates within reasonable test duration while keeping the test temperature limited by the ISG-11[1] limit value of 400ºC. Five thermal creep tests have been conducted using the Zry-2 rod material. Table 1 summarizes the test conditions, the special cool-down procedures and the mechanical test performed after creep test for each sample.

Test identification	Temperature (ºC)	Hoop Stress (MPa)	Shutdown	Mechanical Tests (135ºC)
D	400	190	Reduce to σ_θ = 70 MPa before standard cool-down	RCT
B	400	160	Standard	RCT
C	400	130	Standard	RCT
A	400	100	Standard	RCT
E	360	160	Increase to T= 400ºC before standard cool-down	ATT

Table 1. Test Conditions

Standard cool-down is performed at 2ºC/min and the pressure evolving with temperature.

Pre and post test characterization from every specimen have been performed: profilometry, visual inspection, length measurement and metallography with hydride measurement by SEM and radial hydride fraction (fn). The fn factor is analyzed from the images used for evaluating the amount of hydrogen and is evaluated according to standard ASTM B811-90 [2], defined as fn = *Total number of features in the analyzed field* $\pm\Psi$ */ f number of features inside the radial direction (*$\Psi = 40$ degrees)

The tests A, B, C and D, performed at the same temperature and different hoop stresses should provide the hoop stress dependence of the creep. The tests B and E, performed at the same hoop stress and different temperatures should provide the temperature dependence of the creep material. Additionally, the special cool-down procedures, with the total hydride content and radial hydride fraction measurements, could provide some “hoop stress/ fraction of radial hydrides” relation that could be later on connected with the ductility obtained from RCT and AT.

The test equipment used for both RCTs and ATTs consisted of an Instron universal testing machine type 1271 of load cell capacity 50 kN installed in a hot-cell chain, with different arrangements depending on the test. For RCT a pair of plates, with plane, parallel faces, both for holding and applying compressive loads to the specimen, was mounted between the moving and the fixed crossheads of the testing machine. The relative displacement of the loading plates was measured from the cross-head position data. Both RCT and ATT were performed at 135°C as representative condition of the transport.

3.2. Test Results

From the laser records, the engineering hoop creep strain is calculated as

$$\varepsilon_{\theta,creep} = \frac{D(t,T,P) - D(t=0,T,P)}{D(t=0,Room\,Conditions)}$$

Thus, considering as initial diameter the value once temperature and pressure have stabilized, the creep strain is calculated from the diameter increment at the laser position (mid-sample). This diameter change is normalized using the diameter before the test, measured in room conditions. Figure 1 depicts the creep test results at the laser position.

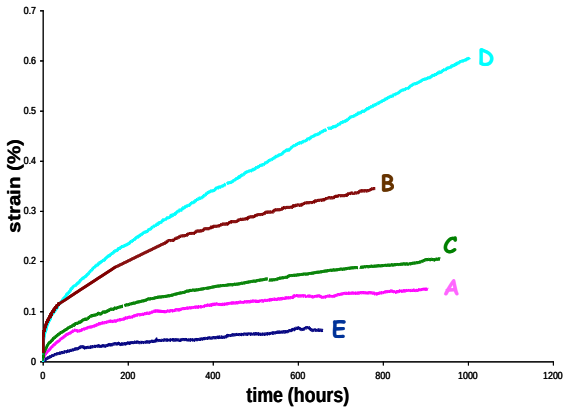


Figure 1. Creep test results.

The five samples developed primary and secondary creep and no specimen did fail either during test or cooling-down.

Table 2 summarizes the secondary creep rates estimated from the straight portion of the curves and a comparison of these results with the data from the ZIRLO tests performed in the past [3], if the temperature and pressure conditions were equivalent.

Test identification	Zry-2 creep rate (%/day)	ZIRLO creep rate (%/day)
D	0.011	0.014
B	0.0053	0.0091
C	0.0022	Not available
A	0.0016	Not available
E	0.0015	Not available

Table 2. Estimated secondary creep rates

Creep rate decreasing with lower hoop stress shows coherence between test parameters and measured creep rates. These results bounds the conditions during fuel dry storage, as 190/160 MPa hoop stress envelopes with big margin the actual hoop stress due to internal pressure in BWR irradiated fuel rods; additionally during dry storage the temperature decreases along the time, with the corresponding diminishing of the creep rate, as results from specimen E versus D show.

Additionally, a significant effect of temperature is observed. The comparison between the Zry-2 results and the ZIRLO results provides clear indication that the creep rate for RX material is lower than the creep rate for CWSR, which is consistent with other public data [4].

Once the material was creep-tested, and then storage period simulated, some RCT and ATT were performed to simulate transport accident conditions. Results are compared to tests performed on irradiated material that had not been creep-tested.

The RCT results show higher ductility for the creep-tested material, which should be a consequence of the irradiation damage annealing [6]. On the other hand, the radial hydrides seems to increase with hoop stress during shutdown and some degree of reorientation has been observed even at 70 MPa hoop stress, consistent with published data for Zry-2 [5]. Based on published data, the situation is different for CWSR material where the reorientation has been observed for higher stress values [6]. Additionally, load at cracking for RCT usually increase as fn factor decrease, but deeper analysis should be performed, as some influence of previous deformation due to the creep-tests has been clearly observed on test D results. Table 3 shows these results.

Test identification	Strain from Creep Test(*) (%)	fn	Load at crack in RCT (N)
B	0.45	0.93	595
C	0.25	0.92	767
A	0.19	0.61	1035
D	0.70	0.67	631

Table 3. RCT results

(*) Local strain at specimen position

Figure 2 shows the different degree on hydrides reorientation from specimens C and A

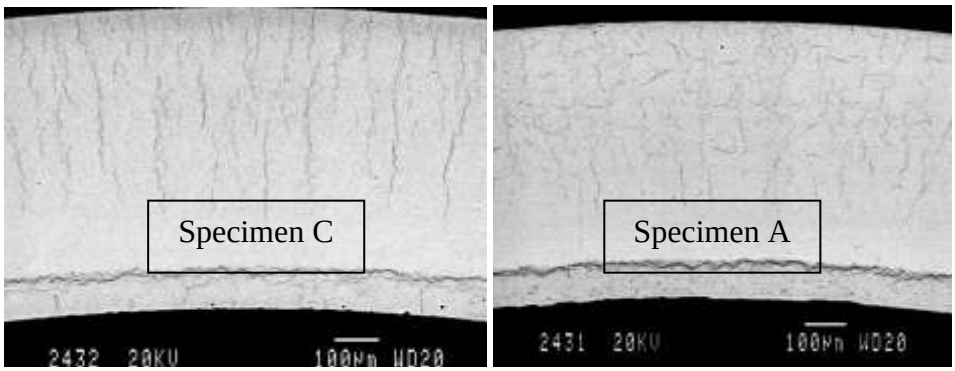


Fig. 2. Metallographies

Figure 3 shows results for both ATT, one on irradiated not creep-tested material and another one for material from creep test E.

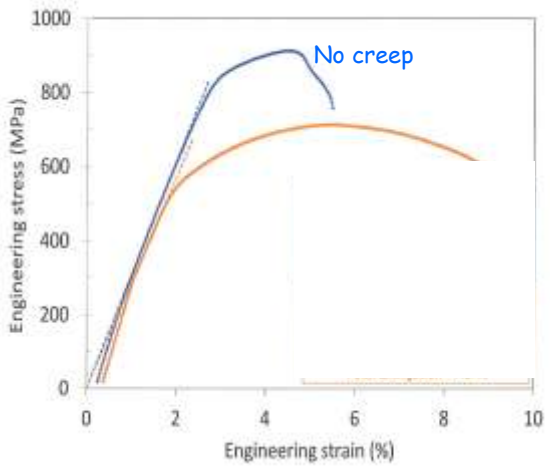


Fig. 3. ATT results

Higher ductility and lower yield strength for the creep tested material can be observed, probably due to the irradiation damage annealing.

4. Conclusions

The CSN, ENRESA and ENUSA work together in research and development programs related to dry storage and transport of spent fuel.

Creep tests performed on limiting stress and temperature conditions, show lower creep rates for RX material than those obtained in the past for ZIRLO, consistent with published data.

In order to reorient different fractions of hydrides, special procedures for temperature reduction have been implemented at the end of each test. The radial fraction increase with

the stress and some degree of reorientation has been observed even at 70 MPa hoop stress, consistent with published data.

The fn/ductility relation should be deeper analyzed, considering also the previous thermal-stress process suffered by the material and the annealing of irradiation damage.

Axial tensile tests show the effect of the irradiation damage annealing on mechanical properties.

5. References

1- US NRC Spent Fuel Office Interim Staff Guidance-11, Revision 3, Novembre 2003
2- ASTM B811-90 “Standard Specification fo Wrought Zirconium Alloy Seamless Tubes for Nuclear Reactor Fuel Cladding”
3- M. Quecedo et al. “Results of Thermal Creep Test on Highly Irradiated ZIRLO”. . WRFPM 2008. Korea
4- K.Ito et al “Evaluation of irradiaton effect on spent fuel cladding creep properties”. International meeting on LWR Fuel Performance. Florida, 2004.
5- Masaki AOMI et al. “Evaluation of Hydride Reorientation Behavior and Mechanical Property for High Burnup Fuel Cladding Tube in Interim Dry Storage”.. 15th International Symposium on Zirconium in the Nuclear Industry, June 2007.
6- C. Cappelaere et al. “Long Term Behaviour of Spent Fuel Cladding in Dry Storage Conditions” ICEM 2001,Bruges October 2001.

Acknowledgements

The collaboration from Vattenfall providing the material is gratefully recognized.

Dry storage reliable solutions for the management of spent nuclear fuel in the long term

H.ISSARD
AREVA TN International, 1 rue des Hérons, 78180 Montigny le Bretonneux, France

Abstract

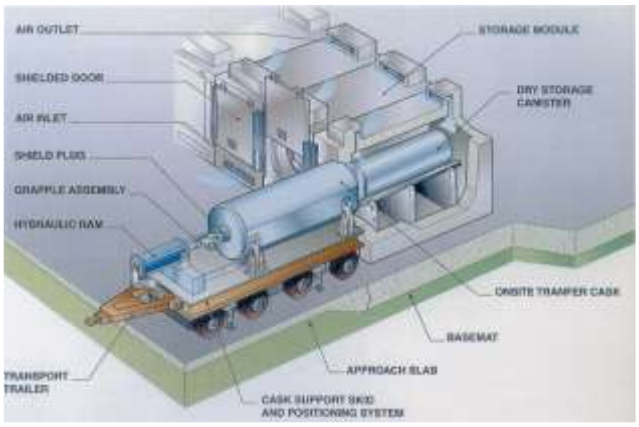
Interim storage manages the used fuel stockpile and provides an intermediate solution waiting for evaluation and decisions of disposal sites. Spent fuel pools (SFP) allow for used fuel cooling. More and more utilities choose to move fuels from pools to dry storage systems. In principle interim storage is for a limited time, 30 to 40 years. Dry storage contributes to the reduction of the inventory in SFP. In 2012, 93 reactors in the USA contracted for dry storage. Even in very severe accidents and natural disasters these systems provide a very high level of safety.
The dry storage industry has matured. Now the license submittals of cask vendors are focused on high BU fuels and resulting thermal issues expanding fuel types and maximizing capacities. For example Nuhoms® system can accept from 61 to 69 BWR fuels, and from 32 to 37 PWR fuels. The PWR, BWR or VVER fuel characteristics may have various enrichment (value up to 5%), various cooling time (down to 2 years) and various burn-ups (up to 65,000 MWd/tU). Secure and reliable cask supply is critical to the industry as nearly 90% of all US reactors will rely on dry storage by 2015 for continued operation.
There is an important industrial experience feedback and excellent safety records. Metal casks TN®24 and GNS Castor were first loaded in 1985 and Nuhoms® system in 1989. All type of fuel LWR BWR, PWR Candu etc are now stored in dry casks. During 25 years, there has been no incident, showing that interim storage is excellent from the point of view of safety.
One ground reason for the excellence of dry storage is that it is a highly resistant and passive system.
In the past years, safety studies cover extreme conditions : aircraft crash, fire, earthquake, cask burial, cask tip-over, fuel cladding breach. Aircraft crash testing has been achieved successfully on cask specimen.
Dry storage has been submitted to actual severe conditions: March 2011 East Japan Magnitude 9 earthquake and tsunami, August 2011 Central Virginia, USA Magnitude 5.8 earthquake. For both natural disasters (which have different magnitudes) the dry storage safety was not affected**:** no change to radiation levels, no abnormal conditions present.
Facing the current trend towards expanding Spent Fuel Interim Storage capabilities, AREVA Logistics Business Unit has developed various designs world-wide:
Metallic casks – TN®24 family: Casks in operation in USA, Japan, Belgium, Switzerland, Upcoming casks in Germany, Italy.
Canister systems:
NUHOMS® system in operation in USA, Armenia.
TN NOVA™ system in operation in 2014 in Switzerland.
For very long term storage, for example after 100 years, gap analysis investigations are carried out by international common research program such as EPRI ESCP. The identified gaps concern the following issues : thermal model for casks, degradation process of fuel material (cladding creep…), degradation of neutron poison material, degradation process of canister material , stress corrosion cracking, concrete degradation, seals degradation, technical basis for amount residual water, measurement of residual water , drying procedures and criteria, corrosion issues.
To conclude, a large industrial experience and many safety studies show the ability of the interim storage systems to protect the public and manage safely in the long term the spent fuel from nuclear power reactors.

1. Background

Dry storage is a safe and economic intermediate solution contributing to the reduction of the inventory in Used Fuel Pools while awaiting the deployment of a back-end used fuel strategy being it direct disposal or reprocessing and recycling. More and more utilities choose to move used fuels from pools to dry storage systems given the reduced operation and maintenance costs and the lengthening intermediate storage periods. This paper gives an overview of advantages and safety performance of dry storage systems for the management of used nuclear fuels.

2. Industrial experience

The dry storage industry has matured resulting into industrially proven solutions applied worldwide. There are mainly three technologies as shown on fig 1, 2 and 3 : Canister based systems, Metal casks and vaults. Vaults are used for Spent Fuel storage in Hungary, in Korea and will be used in Spain.



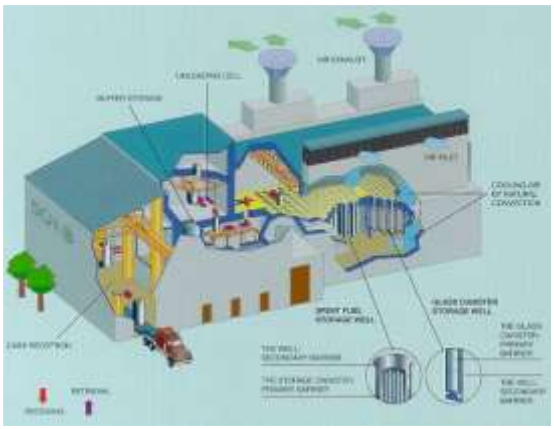


Fig 1: Dry storage in concrete modular systems NUHOMS® **Fig 2: Dry storage Vaults**

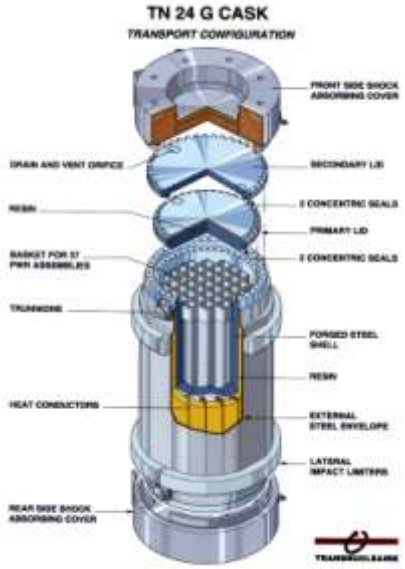


Fig 3: Metal casks TN®24

In Europe metal casks are mainly used whereas canisterised systems are being used in the USA. The PWR, BWR or VVER used fuel characteristics may have various initial enrichment (value up to 5%), various cooling time (down to 2 years) and various burn-ups (up to 70,000 MWd/tU).

Facing the current trend towards expanding Spent Fuel Interim Storage capabilities, AREVA TN International Unit has developed various designs world-wide:
NUHOMS® systems are widely used in USA, and also selected in Armenia. In total, 500 NUHOMS® systems have been ordered. Nuhoms® system can accept from 61 to 69 BWR fuels, and from 32 to 37 PWR fuels. The table 1 summarizes the capacity, burn ups, cooling times of the contents accepted in these storage designs.

Type	Number of fuels	Burn up (MW d/tU)	Cooling time (years)	Enrichment (%)
NUHOMS® 24P	**24 PWR**	**45 000**	**5**	**4.0**
NUHOMS® 24PT1	**24 PWR**	**45 000**	**10**	**4.05**
NUHOMS® 24PT2	**24 PWR**	**45 000**	**5**	**4.0**
NUHOMS® MP 187	**24 PWR**	**38 300**	**7**	**3.43**
NUHOMS® 24PT4	**24 PWR**	**45 000**	**5**	**4.8**
NUHOMS® 32PT	**32 PWR**	**45 000**	**5**	**5.0**
NUHOMS® 24PHB	**24 PWR**	**55 000**	**7.3**	**4.5**
NUHOMS 24PTH	**24 PWR**	**62 000**	**5**	**5.0**
NUHOMS® 32PTH	**32 PWR**	**60 000**	**7**	**5.0**
NUHOMS® 37PTH	**37 PWR**	**65 000**	**7**	**5.0**
NUHOMS® 52B	**52 BWR**	**45 000**	**5**	**4.0**
NUHOMS® 61BT	**61 BWR**	**40 000**	**5**	**4.4**
NUHOMS® 69BTH	**69 BWR**	**70 000**	**5**	**5.0**
NUHOMS® 56V	**56 VVER**	**42 000**	**5**	**3.6**
NUHOMS® FF	**Damaged PWR**	**38.300**	**7**	**3.4**

Table 1: Typical characteristics of some NUHOMS ® Systems, non exhaustive

AREVA TN International has developed metallic dual purpose casks with more then 200 casks ordered: it is the TN®24 family. Many of them have both transport and storage licences. Casks are in operation in USA, Japan, Belgium, Switzerland, and upcoming casks in Germany, Italy. The table 2 summarizes the capacity, burn ups, cooling times of the contents accepted in these storage designs.
TN NOVA ™ system is the latest system designed. It is a canister system with a storage overpack which can be tilted vertically. It will be in operation in 2014 in Switzerland.
The new challenges for dry storage solutions are the maximisation of storage capacities and the acceptance of higher BU fuels (up to 70,000 MWd/tU) and shorter cooling times.

Packaging	Number of fuels	Burn-up (MW d/tU)	Cooling time (years)	Enrichment (%)	Country
TN24 D	28 PWR	36 000	8	3,4	B
TN24 DH	28 PWR	55 000	7	4,1	B
TN24 XL	24 PWR	40 000	8	3,4	B
TN24 XLH	24 PWR	55 000	7	4,3	B
TN24 SH	37 PWR	55 000	5	4,25	B
TN24 G	37 PWR	42 000	10	3,81	CH
TN24 (F1)	37 BWR	33 000	4	3,2	J
TN24 E	21 PWR	65 000	5	4,65	G
TN32	32 PWR	45 000	7	4,05	US
TN40	40 PWR	45 000	10	3,85	US
TN24 P	24 PWR	33 000	5	3,5	US
TN52 L	52 BWR	55 000	Mini 2,5	4,95	CH
TN24 SWR	61 BWR	70 000	Mini 5,5	5,0	G
TN68	68 BWR	45 000	7	4,4	US
TN97 L	97 BWR	35 000	10	4,0	CH
TN24 BH	69 BWR	50 000	6	5,0	CH
TN24 (F1)	52 BWR	33 000	4	3,2	J
TK69	69 BWR	40 000	10	3,2	J
TN24 ER	32 BWR (Th)	13 700	40	5,2	I

Table 2: Typical characteristics of some Dual-purpose TN® 24 casks, non exhaustive

3. Flexibility and transportability

The dry storage concepts answer to the need to quickly evacuate used fuels which is important for nuclear operators.

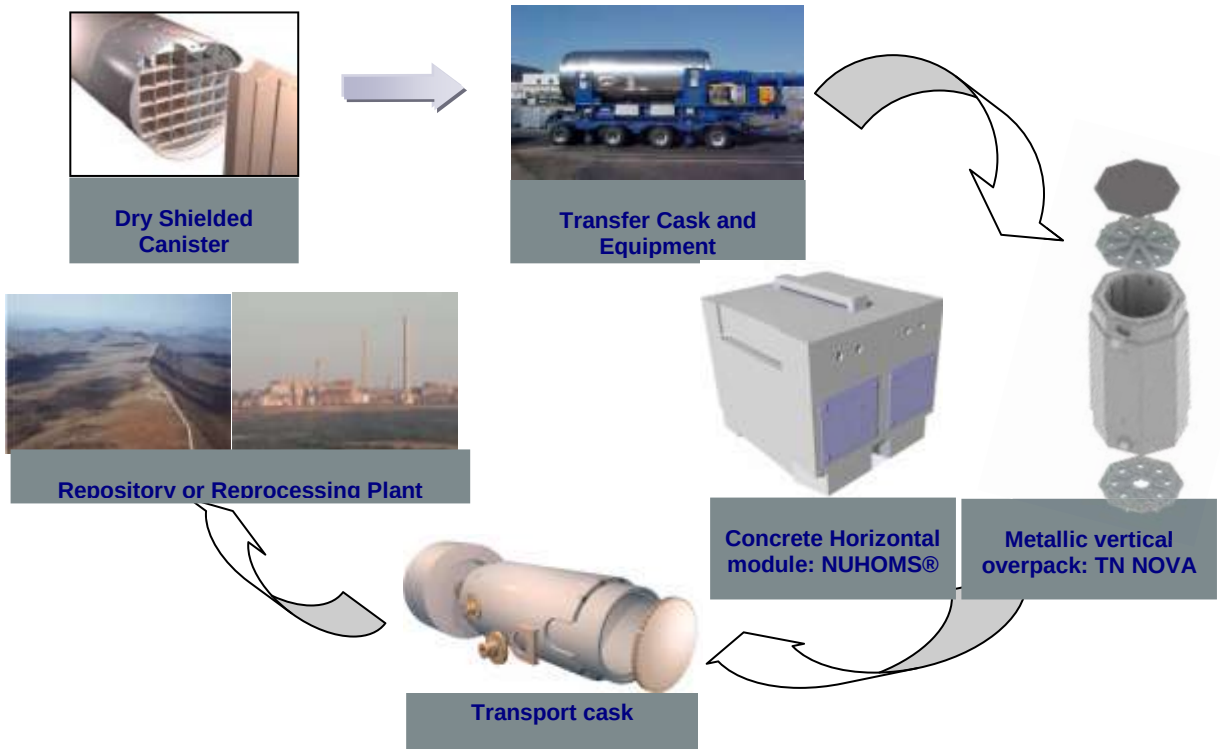


Fig 4: Transportability of spent fuel after interim storage

Another aspect of their flexibility is the compliance with various fuel designs and specifications, the trend being towards higher burn-ups 65 or 70 GWd/MTU, and shorter cooling times.
Dry storage systems show compliance with various specific site constraints, and commitment to keep occupational radiation exposures ALARA.
And the foremost: transportability is considered basically at the development stage of the dry storage solution, therefore, the system allow for decision towards a final used fuel management: reprocessing or repository (Fig 4).

4. Safety of dry storage

4.1 Background

Safety of dry storage is adressed in IAEA document SSG-15 [1]. Before Fukushima accident the unavailability of off-site routes led to use Spent Fuel Pools as used fuel storage. Safety regulators have allowed high density racking in Spent fuel Pools. Although it was not the main issue at Fukushima Daiichi NPP, the spent fuel pools were an aggravating factor in the difficult emergency situation by the operators. Concerns were to keep the pools filled with water, and the challenge was to adequately cool the spent fuels. As a consequence, among post Fukushima views expressed in IAEA expert meetings in March 2012 [4] there was the remark and recommendation to reduce spent fuel inventory through development of dry interim storage, should recycling not foreseen in the near term (priority will be likely put on dry storage).

4.2 Principles of safety of dry storage

Originally, metal casks or canister dry storage systems designs are based on the experience of transport safety. The cask designs have similar features as transport casks as for containment, shielding, criticality safety, and many have both storage and transport license (dual purpose casks). TN International has developed a dedicated fleet of casks which has transported more than 44000 tHM (tons of heavy metal) of used fuel to AREVA La Hague reprocessing plant without accident during 40 years. Dry storage is generally considered as a highly resistant and passive system, and the records during 30 years have confirmed this statement.

4.2 Verification

There is an important industrial experience feedback and excellent safety records. Metal casks TN®24 and GNS Castor were first loaded in 1985 and Nuhoms® system in 1989. All type of fuel LWR BWR, PWR Candu etc are now stored in dry casks. Also fuels from Gas cooled reactors may be dry stored. During 25 years, there has been no incident, showing that interim storage is excellent from the point of view of safety.
One ground reason for the excellence of dry storage is that it is a highly resistant system in severe accidents. In the past years, safety studies cover extreme conditions : aircraft crash, fire, earthquake, cask burial, cask tip-over, fuel cladding breach. Aircraft crash testing (Fig 5) has been achieved successfully on cask specimen. For example AREVA TN International has tested successfully the TN Nova™ design. The same successful results are obtained for TN®24 casks (with anti-aircraft crash covers) for Belgium dry storage installation.

In addition, it should be underscored that dry storage has been submitted to actual severe conditions: The March 2011 East Japan Magnitude 9 earthquake and tsunami, and the August 2011 Central Virginia, USA Magnitude 5.8 earthquake. For both there was spent fuel dry storage in metal casks and in North Anna, there was also Nuhoms® system. For both natural disasters (which have different magnitudes) the dry storage safety was not affected**: c**osmetic damage to the dry storage, and no change to radiation levels, no abnormal conditions present.

Fig 5 : TN NOVA™ dry storage system - aircraft crash test

5. Long term dry storage issues

There is now the recognition that used fuel will need to be stored past the regulatory licensing timeframe [3]. In the US, this is mainly a result of the cessation of licensing activities for the planned Yucca Mountain repository.
Dry storage systems were originally designed for limited periods. With extended dry storage, it is generally believed that used fuel can be stored safely past the regulatory time period. However, the technical arguments need to be made to justify extended licensing periods. Addressing the data gaps will provide the necessary arguments to justify license extensions [2].
The specialists now agree that the main technical gaps are:

1. Fuel integrity – creep and hydride effects on fuel cladding and retrievability of fuel after storage
2. Canister integrity– general canister corrosion and stress corrosion cracking of canister closure welds, particularly in marine environments
3. Closure – degradation of bolts and seals (metallic and elastomeric)
4. Concrete degradation

Verifications and analysis are carried out by international common research programs (by EPRI and IAEA) and by national programs (for example in France [5]): thermal model for casks, evaluation of embrittlement of the used fuel cladding due to hydride re-orientations, creep of fuel material, degradation of neutron poison material, degradation of canister material (stress corrosion cracking in the welding zone), concrete degradation, seals degradation [6], evaluation of residual water , drying procedures and criteria, general corrosion resistance.

Ageing and degradation of package components were evaluated by R&D national programs. Tests or modelisation methods were used to demonstrate the good behaviour of seals [6] and of neutron shielding materials in storage conditions.
For the evaluation of spent fuel integrity in transport, especially in accident conditions, AREVA TN International has carried out a joint program with INS (UK) to better assess fuel assemblies mechanical behaviour and to confirm hypotheses for safety-criticality studies. The methodology [7], gathered in a technical guide has been reviewed by authorities and their comments are now integrated in the final document.

6. Conclusions

A large industrial experience and many safety studies show the ability of the interim dry storage systems to protect the public and manage safely and economically in the long term the used fuel from nuclear power reactors.
Verifications are carried out to demonstrate that the fuel can be stored safely and can be transported after storage for the deployment of the back-end strategy: direct disposal or reprocessing and recycling.

7. References

[1] IAEA 2012 SSG15 Safety of dry storage.

[2] NUREG/CR-7116/ SRNL-STI-2011-00005 Materials Aging Issues and Aging Management for Extended Storage and Transportation of Spent Nuclear Fuel.

[3] IAEA Tecdoc 1293 Long term storage of SNF, survey and recommendations.

[4] IAEA International Experts' Meeting on Reactor and Spent Fuel Safety in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant, IAEA, 19–22 March 2012.

[5] CEA France 2005- Rapport CEA-R-6084, Synthesis on the spent fuel long term evolution.

[6] Ageing of metallic gaskets for spent fuel casks: Century-long life forecast from 25,000-h-long experiments, H. Sassoulas, L. Morice, P. Caplain, C. Rouaud, L. Mirabel, F. Beal *CEA, GARLOCK, Pierrelatte, France*, Nuclear Engineering and Design 236 (2006) 2411–2417.

[7] Description of fuel integrity project methodology principles, A. Zeachandirin, M. Dallongeville *TN International (AREVA), France*, P. Purcell, A. Cory *International Nuclear Services (INS), United Kingdom*, PATRAM 2010, London.

FRAPCON-3.4 EXTENSION TO MODEL HYDROGEN EFFECTS ON CLADDING MECHANICAL BEHAVIOR DURING DRY STORAGE

F. FERIA, L.E. HERRANZ
Unit of Nuclear Safety Research, CIEMAT
Avda. Complutense, 40 28040 Madrid-Spain
francisco.feria@ciemat.es

ABSTRACT

One of the main issues in dry storage is to guarantee cladding integrity. In this sense, clad creep and the hydrogen content play a decisive role. The latter is involved in postulated failure mechanisms during dry storage as delayed hydride cracking (DHC), hydride radial reorientation, or even creep (hardening effect). CIEMAT has launched a project to encapsulate all those effects in an analytical tool capable of modelling fuel rod behaviour along dry storage. The platform chosen to do that was FRAPCON-3.4 code and the extension to dry storage conditions is called FRAPCON-3.4xt. This paper presents the progress achieved so far in this venture.

A new creep law accounting for the hydrogen effect (hardening) has been derived and implemented. Likewise, a criterion for DHC initiation has been also included in the FRAPCON-3.4 code. FRAPCON-3.4xt has been applied to Zry-4 fuel rod postulated scenarios. The results have revealed that resistance to deformation due to hydrogen hardening dominates over the strain related to clad thinning due to oxide layer. Additionally, limited influence of the initial gap pressure on DHC has been shown. The most challenging scenario used is far from reaching the DHC onset threshold.

This study has been sponsored by ENRESA in the framework of the AICAST project.

1. Introduction

Experimental tests simulating conditions throughout more than 20 years of dry storage are impractical. This highlights the need to develop analytical tools capable of predicting pin behaviour along dry storage. The present work forms part of a project launched to extend FRAPCON-3.4 capabilities to dry storage conditions. The extension is called FRAPCON-3.4xt.

The main code changes are focused on the mechanical module to simulate cladding mechanical behaviour over the dry storage of nuclear fuel. The mechanisms to model (related to loss of cladding integrity) are: creep strain and hydrogen related mechanisms as delayed hydride cracking (DHC), embrittlement or hydride radial reorientation. Moreover, creep model should take into account the resistance to deformation due to hydrogen content (hardening effect). Code limitations involve modelling the effect caused by these mechanisms, except in the case of creep in which phenomenon can be modelled to be implemented in the mechanical module.

Therefore, FRAPCON-3.4xt is being developed in several stages according to the mechanisms implemented: I) Creep, II) DHC and hydrogen embrittlement, and III) Hydride radial reorientation. The cladding material chosen is Zry-4, although the project is open to others light water reactor (LWR) zirconium alloys. The adaptations carried out in each stage are verified by

applying FRAPCON-3.4xt to fuel rod postulated scenarios, in which two burnup targets of around 50 and 70 GWd/tU are used, depending on verification needs.

This paper presents the work carried out in stages I and II. Particularly, hydrogen hardening effect has been included in a new creep law, based on experimental data found in the literature. Moreover, DHC initiation has been implemented, from a fracture mechanic point of view and supported by conservative assumptions. Finally, FRAPCON-3.4xt has been applied to postulated scenarios of a Zry-4 fuel rod not only to verify, but also to analyse these hydrogen related mechanisms.

2. FRAPON-3.4xt

2.1. Creep law

In order to derive a simple expression of creep with as few parameters as possible with the aim to make an easy implementation in the code, CIEMAT proposed a generic equation in terms of hoop creep rate [1,2]:

$$\dot{\varepsilon}_\theta = f_1(\sigma_\theta)\cdot f_2(T)\cdot f_3(\phi t)\cdot t^{-0.5} \tag{1}$$

where the independent variables are the average temperature of the cladding, T, the applied hoop stress, σ_θ, the fluence during the irradiation phase (hardening effect), ϕt, and the storage time, t. The functions f_i (i=1,3) are given elsewhere [1,2]. The expression (1) depends on a reasonable number of parameters (4) and assumes no cross-effect between variables (if existent, they should be second order). This equation form was convenient since it was quite like the one in FRAPCON-3.4 for in-reactor conditions.

In this study, based on data made available in the open literature [3,4,5,6], CIEMAT has proposed a correlation that encapsulated the hardening effect of the hydrogen concentration, H:

$$\dot{\varepsilon}_\theta = f_1(\sigma_\theta)\cdot f_2(T)\cdot f_3(\phi t)\cdot f_4(H)\cdot t^{-0.5} \tag{2}$$

where

$$f_1(\sigma_\theta) = \frac{a}{2}\sigma_\theta^b \tag{3}$$

$$f_2(T) = \exp\left(\frac{-c}{T+273}\right) \tag{4}$$

$$f_3(\phi t) = \exp(-d\cdot\phi t) \tag{5}$$

$$f_4(H) = \exp(-e\cdot H) \tag{6}$$

This correlation is valid throughout the individual variable margin at which the data used were measured (Table 1). Table 2 shows the values of the fitting parameters. By comparing to experimental data the correlation derived has shown relative errors below 20 %. Although a larger database is needed to obtain a sound model, the creep law obtained shows aspects of interest from the point of view of its use in a thermo-mechanical code: simplicity (few

parameters), completeness (more effects considered) and acceptable accuracy (mentioned above). Therefore, this CIEMAT creep law has been implemented in the FRAPCON-3.4 mechanical module.

Table 1. CIEMAT creep law validity ranges

	Range
σ_θ (MPa)	75-250
T (ºC)	360-400
ϕt (n/cm^2)	(0-9)10^{21}
H (ppm)	0-600
t (h)	0-11000

Table 2. Parameters of CIEMAT creep law

Parameter	**T < 380 ºC or T ≥ 380 ºC σ_θ < 187 MPa**	**T ≥ 380 ºC σ_θ ≥ 187 MPa**
a	7.2·10^4	360
b	1.84	2.95
c	15000	
d	2.4·10^{-22}	
e	1.6·10^{-3}	

2.2. DHC initiation

From the fracture mechanics point of view, DHC can not occur in a cladding with a pre-existing crack if the stress intensity at the crack tip is not enough to initiate the defect propagation [7]. This argument is independent of the hydrogen content in the cladding. Thus, a control variable to DHC initiation is the stress intensity factor at the crack tip, K_I, which can be estimated by the expression [8]:

$$K_I = \sigma_\theta \left(\frac{1.2\pi}{Q} \cdot l \right)^{1/2} \quad (7)$$

where σ_θ is the applied hoop stress, l is the crack depth and Q is a shape factor (1 and 1.5 for longitudinal and elliptical defects, respectively). The threshold of K_I to initiate crack propagation is the DHC fracture toughness, K_{IH}, with a minimum value of the order of 5 MPa·m$^{0.5}$ [8].

With the aim to extend FRAPCON-3.4 capabilities to DHC initiation analysis in dry storage, a criterion based on the K_I/K_{IH} ratio has been implemented in the mechanical module. To do that, pre-existing cracks have to be assumed. It has been used conservative initial size values of 28% of the cladding thickness [7]. Moreover, longitudinal cracks (Q=1) have been assumed.

3. Applications

3.1. Hydrogen effect on creep strain

In order to check FRAPCON-3.4xt in terms of fuel rod creep response under dry storage conditions, the code has been used to simulate a postulated scenario of a Zry-4 fuel rod irradiated to 51 GWd/tU.

The fuel rod history has been simulated, from the in-reactor irradiation (4 cycles with an average linear power of 27.2 kW/m (shown in [2])) to dry storage cooling (100 years), going through a 5 years cooling in the fuel storage pool and even the anticipated heat-up during the 1 day dry out process. The temperature decay considered during dry storage is the conservative exponential decay found in the literature [9]. The initial temperature has been set to 400 ºC, which is the highest temperature allowed by ISG-11 rev. 3 [10].

The results obtained in the simulation are consistent: identical creep strain as in FRAPCON-3.4 simulation during the pre-dry storage period [2] and creep-out during dry storage. In order to show how significant hydrogen can be, Figure 1 displays the evolution of the maximum value of hoop strain obtained during dry storage, ε_θ^{max}, as a function of the oxide thickness layer, δ_{ox}. The plot demonstrates that when hydrogen effect is not considered, the hoop strain would increase almost linearly with the clad oxidation (this results from the clad thinning due to oxidation). However, as the hydrogen hardening effect is taken into account, it is noted that the previous growing trend is not just smoothed, but reversed with a much steeper slope. In other words, the enhancement of hydrogen absorption by the clad oxidation dominates over the increase of the stress due to resulting clad thinning.

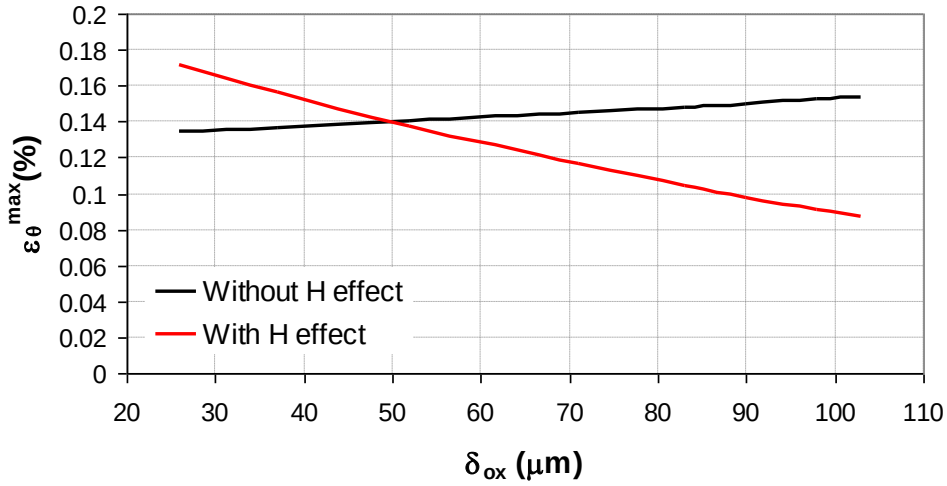


Fig. 1. Maximum hoop creep strain during dry storage vs oxide thickness

It is worth to note that, in spite of the hydrogen positive effect on cladding creep failure, it also results in material embrittlement. Thus, the determination of a critical plastic strain dependent on hydrogen concentration, among other variables, would be merit consideration in the dry storage of spent fuel rods irradiated to high burnup.

3.2. Initial gap pressure effect on DHC

According to the equation (7), DHC initiation depends on the stress at which the cladding is submitted along dry storage. This stress is substantially affected by fission gas released during in reactor period and the initial fill gas pressure, P_i. This study has been focused on analysing initial gap pressure effect on DHC initiation. To do that, FRAPCON-3.4xt has been used to simulate two postulated scenarios of a Zry-4 fuel rod irradiated to 67 GWd/tU with different fill gas pressure: a) 2 MPa and b) 3.45 MPa (minimum and maximum values found in the literature (Rashid, 2007)). The fuel rod history has been simulated as in the previous section, except the irradiation phase (5 cycles with an average linear power of 29.9 kW/m (shown in [2])), which is the most challenging scenario used.

Table 3 shows the results related to the maximum hoop stress obtained in each case. It can be seen that even with the higher initial gap pressure, DHC onset does not occur under the simulated conditions (the applied stress is not enough). If both scenarios are compared, around 1.2 times higher stress intensity at the crack tip is obtained as the initial gap pressure is increased by 75 %. Therefore, DHC initiation shows low sensitivity to the initial fill gas pressure.

Table 3. DHC initiation analysis in different scenarios (K_{IH} = 5 MPa·$m^{0.5}$)

Scenario	l (μm)	σ_θ (MPa)	K_I/K_{IH}
a)	174.1	91.5	0.48
b)	174.6	111.3	0.58

4. Final remarks

The present work has been focused on stages I and II of a project launched to obtain a tool, based in FRAPCON-3.4, to predict fuel rod thermo-mechanical behaviour during dry storage, so called FRAPCON-3.4xt. The new adaptation of the code has the capability to assess mechanisms related to hydrogen as hardening on cladding creep strain and DHC.

The FRAPCON-3.4xt application to Zry-4 fuel rod postulated scenarios has highlighted two interesting aspects. On the one hand, the importance of considering the hydrogen hardening effect in dry storage creep laws, particularly at high burnups. On the other hand, the limited effect on DHC onset of the fuel rod initial gap pressure.

Otherwise, the modelling of hydride reorientation effect and its implementation in FRAPCON-3.4xt will be treated in further work (stage III of the project). Moreover, further work is foreseen to keep on improving CIEMAT creep law (database extension, other alloys, annealing).

5. References

[1] Herranz, L.E., Feria, F., 2010. Extension of the FRAPCON-3.3 creep model to dry storage conditions. Progress in Nuclear Energy, 52, 634-639.

[2] Feria, F., Herranz, L.E., 2011. Creep assessment of Zry-4 cladded high burnup fuel under dry storage. Progress in Nuclear Energy, 53, 395-400.

[3] Bouffioux, P., Rupa, N., 2000. Impact of hydrogen on plasticity and creep of unirradiated Zircaloy-4 cladding tubes. In: G.P. Gabol, G.D. Moan (Eds.), Zirconium in the Nuclear Industry 12th International Symposium, ASTM STP 1354, 399-424.

[4] Bouffioux, P., Leclercq, S., 2001. Interim dry storage of PWR spent fuel assemblies - development of a long term creep law to assess the fuel cladding integrity. 8th International Conference on Radioactive Waste Management and Environmental Remediation.

[5] Ito, K., Kamimura, K., Tsukuda, Y., 2004. Evaluation of irradiation effect on spent fuel cladding creep properties. Proceedings of the 2004 International Meeting on LWR Fuel Performance, 19-22.

[6] Bouffioux, P., 2005. Transportation and interim dry storage of PWR' s spent fuel. EDF report, HT25-C2005-192/PBF.

[7] Rashid, Y.R., Sunderland, D.J., Montgomery, R.O., 2000. Creep as the Limiting Mechanism for Spent Fuel Dry Storage. EPRI Report 1001207.

[8] Coleman, C.E., 2011. Delayed Hydride Cracking Considerations Relevant to Spent Nuclear Fuel Storage. EPRI Report 1022921.

[9] Rashid, Y.R., 2007. Spent fuel transportation applications - Assessment of cladding performance: A synthesis report. EPRI Report 1015048.

[10] USNRC, Spent Fuel Project Office, Interim Staff Guidance Nº 11, Revision 3 (ISG-11, rev 3), Nov 2003.

STUDSVIK CMS CAPABILITY FOR SPENT NUCLEAR FUEL

G. GRANDI
Studsvik Scandpower, Inc.
504 Shoup Avenue, Suite 201, Idaho Falls, ID 83402 – USA

T. SIMEONOV
Studsvik Scandpower, GmbH
Rathausallee 28, DE-22846 Norderstedt - Germany

ABSTRACT

A number of spent fuel pools around the world are close to reaching their filled capacity and thus alternative storage has to be found. To resolve this situation many utilities use dry storage containers. An accurate prediction of the spent fuel characteristics is mandatory for safe and economic solutions. So, it seems advantageous to couple the in-core and the back-end methods.

The purpose of this paper is to show the Studsvik Core Management System (CMS) capability to simulate spent nuclear fuel in dry storage starting from the In-Core Fuel Management (ICFM) calculations, followed by the evaluation of the residual heat sources using the SNF code, and finally, followed by the evaluation of safety related parameters (maximum creep strain, maximum clad stress, etc.) by ENIGMA.

1. Introduction

Knowing the isotopic composition of the fuel at the time of the discharge from the reactor is a prerequisite for back-end fuel cycle calculations with applications to storage, transportation and disposal of spent fuel. Solving the isotopic decay chains allows the calculation of the radiation as well as the decay heat sources at any time after the discharge from the reactor.

The CMS has the capability to simulate spent nuclear fuel in dry storage starting from the ICFM calculations using the CASMO5 and SIMULATE5 codes, followed by the evaluation of the residual heat sources using the SNF code, and finally, by the evaluation of the thermo-mechanical behaviour by ENIGMA.

The ENDF/B-VII data library used by CASMO5 includes a large number of actinides and fission products that are individually represented and tracked per fuel pin in ordinary ICFM depletion calculations. The operating history of a given fuel assembly is obtained from the ICFM 3-D simulations using SIMULATE5 core tracking calculations. The SNF program, developed by Studsvik, computes the residual heat and the radiation sources by the isotopic summation method. The isotopic concentrations required by the SNF code can be evaluated at the nodal level using the final burnup, spectrum history and power density history at each axial level for any given fuel assembly. Thus, the CMS state-of-the-art capabilities for ICFM are seamlessly integrated into a consistent back-end capability. The relatively high decay heat and low heat transfer coefficients during drying and dry storage may result in relatively high clad temperatures and high clad stresses that must be evaluated as part of the safety assessment. The ENIGMA code, developed by the UK National Nuclear Laboratory and recently added to the CMS code suite, is capable of predicting the thermo-mechanical behaviour of fuel rods during transient conditions as well as steady-state conditions of irradiation in Light Water Reactors (LWRs). Moreover, recent developments of the ENIGMA code extended its capabilities to pool cooling, drying and dry storage.

Section 2 provides a brief description of the CMS codes, namely: CASMO5, SIMULATE5, SNF and ENIGMA. Details are provided in the references at the end of this paper. Section 3 shows an example of the back-end calculations for a PWR rod irradiated close to 30 MWd/kgU and cooled for five years in the fuel pool.

2. Studsvik CMS code package

2.1 CASMO5

CASMO5 [1] is Studsvik's next generation LWR lattice physics code. It has many new features compared with its predecessors. Among them, it is important to mention in the context of back-end fuel calculations: (1) the generalized fuel storage racks simulation capability and (2) the generation of data for the code SNF. The data for SNF is comprised of isotopic number densities for important actinides, fission products and activation products. All data is provided by the ENDF/B-VII or JENDL-4 advanced data libraries [2]. These advanced libraries include additional isotopes (Th-230, Pu-236, Te-132, Te-127m, Te-129m, Tb-160, Ni-58, Ni-62, Te-127, Fe-54, and Fe-55) that are important for spent fuel characterization. Table 1 provides a complete list of the SNF isotopes available from CASMO5.

Actinides	Fission Products		Activation Products
U-234	Se-79	Te-132	Natural Fe
U-235	Kr-85	Xe-133	Co-60
U-236	Sr-89	Cs-134	Natural Ni
U-237	Sr-90	I-135	
U-238	Y-90	Xe-135	
U-239	Y-91	Cs-135	
Np-237	Zr-93	Cs-136	
Np-238	Zr-95	Cs-137	
Np-239	Nb-95	Ba-140	
Pu-236	Zr-97	La-140	
Pu-238	Nb-97	Ce-141	
Pu-239	Mo-99	Ce-143	
Pu-240	Tc-99	Pr-143	
Pu-241	Ru-103	Ce-144	
Pu-242	Ru-105	Pr-144	
Pu-243	Rh-105	Nd-147	
Am-241	Ru-106	Nd-148	
Am-242	Rh-106	Pm-147	
Am-242m	Ag-110M	Pm-148M	
Am-243	Ag-111	Pm-148	
Am-244	Sb-125	Pm-149	
Cm-242	Sn-126	Pm-151	
Cm-243	Sb-126	Sm-147	
Cm-244	Sb-127	Sm-151	
Cm-245	Te-127M	Sm-153	
Cm-246	Te-127	Eu-154	
Cm-248	Te-129M	Eu-155	
Cf-252	Te-129	Eu-156	
I-129	Tb-160		
I-131			

Table 1: CASMO5 SNF isotopes

2.2 SIMULATE5

SIMULATE5 [3] is Studsvik Scandpower's next generation nodal code developed to address deficiencies of existing reactor physics tools for today's aggressive core designs. Multi-group cross sections and other data for SIMULATE5 are generated from CASMO5. In the context of back-end fuel calculations, SIMULATE5 core follow calculations provide:

- operating data for fuel assemblies that will be analyzed with SNF
- thermal power densities for the rod(s) that will be analyzed with ENIGMA

The operating history from core follow calculations is transferred to SNF code through the SIMULATE5 restart file(s). A restart file written at the end of cycle depletion provides detailed assembly burnup and history distributions for each fuel assembly in the core. The final

isotopic concentrations for some actinides and fission products with short half-lives are quite sensitive to the power density history. A user can provide detailed power history to SNF by requesting SIMULATE5 to write a restart file at various exposure points that reflects the operation history of the cycle. Cycle start-up and shutdown dates are available to the SNF program through SIMULATE5 restart files. The fuel rod(s) irradiation history from core follow calculations is transferred to ENIGMA through the SIMULATE5 pin file(s).

2.3 SNF

SNF [4] calculates the 3-D distributions of isotopic concentrations, decay heat and neutron and radiation source terms in a spent fuel assembly for cooling times up to 100,000 years. The SNF isotopic depletion method utilizes the accurate, local neutron spectrum based depletion calculations of the 2-D lattice physics code (CASMO5) as well as nodal operating history data from the 3-D nodal simulator (SIMULATE5).

The SNF power history model accounts for the influence of the actual power density history on the isotopic concentrations generated by the lattice code at a constant power density. The End-of-Life (EOL) concentration of a given isotope in a given node is obtained by integration of the relevant isotopic build-up/decay chains through the entire in-core lifetime. Short-lived fission product isotopes are obviously very sensitive to the local power density; however, isotopes also generated by neutron capture in fission products and many actinides depend quite strongly on the power density history.

The isotopic concentrations at discharge are used as initial conditions for solving the isotopic decay chains. A number of isotopes not present in the lattice physics codes are added in this process. The final isotopic concentrations provide the required basis for calculation of radioactivity, decay heat, gamma heat, spontaneous fission and (α,n) - reaction source neutrons as well as photon release rates and spectra.

SNF applications include analyses of radioactivity, decay heat and neutron emission rates of spent fuel assemblies to be loaded into transport/storage casks as well as full core or fuel pool decay heat calculations required for demonstration of compliance with cooling capacity limitations.

2.4 ENIGMA

The ENIGMA fuel performance code [5], developed by UK National Nuclear Laboratory, has recently been added to the CMS suite. ENIGMA calculates the thermo-mechanical behaviour of an LWR fuel rod in both steady-state and transient conditions. The active stack length region of the fuel rod is represented by a series of axial zones consistent with the axial discretization in the SIMULATE5 model. In each axial zone the fuel is divided into radial annuli. The free volumes associated with the fuel-clad gap, pellet dishes and chamfers, pellet cracks, the pellet bore and upper and lower plena are also modelled. Only radial, i.e. no axial or circumferential, heat flow is assumed and the fuel annuli are all considered to be subject to the same axial strain. The effects of shear stresses are approximated using models for axial extrusion and dish filling and for pellet hour-glassing which feed calculated strain increments back into the main solution scheme. Coupling between the axial zones is restricted to the coolant enthalpy, rod internal pressure and gas transport. Recently, ENIGMA has been extended for modelling dry storage scenarios including the analysis of pool cooling, and drying. This involves: (1) incorporating an out-of-pile clad creep model; (2) including the ability to simulate annealing of the clad irradiation damage; (3) suppression of the clad corrosion modelling during out-of-pile conditions. Rossiter [5] provides a detailed account of these modifications.

ENIGMA is validated against a large database of LWR fuel rod irradiations in both commercial and test reactors. In total, over 500 rod irradiations with burnups up to 90 MWd/kgHM are included in the validation database. ENIGMA's out-of-pile clad creep model has been independently validated against experimental data and so that the integral clad

strain predictions are in good agreement with measurements for the creep testing of dry stored rod segments performed as part of the Dry Cask Storage Characterisation (DCSC) Project [7].

3 Example

An SNF / ENIGMA dry storage assessment has been performed for a UO_2 fuel rod irradiated in a PWR core to approximately 30 MWd/kgU and then cooled in a fuel pool for five years. In the example that follows, the analysis was performed assuming one assembly per cask. Furthermore, the spent fuel was dried for a period of 24 hours using the vacuum drying method before it is transferred to the cask. The spent fuel is assumed to be covered with helium during its cask storage.

The in-pile irradiation history was calculated using CASMO5/SIMULATE5. Fig. 1 shows the rod average rating as a function of rod average burnup for the selected rod.

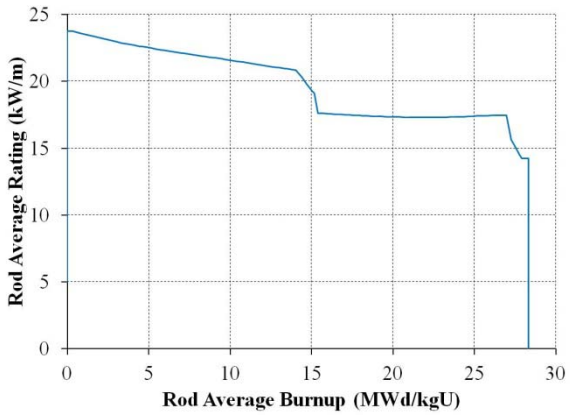


Fig. 1: In-pile irradiation. Rod average linear heat generation rate vs. rod average burnup.

Clad surface temperatures during in-pile irradiation and pool cooling were computed by ENIGMA and set to 50°C respectively. The calculation of the decay heat output, performed with SNF, was used to compute the clad surface temperatures during drying and cask storage. Clad surface temperatures were evaluated using the correlations provided by Manteufel and Todreas [6]. Fig. 2 shows the maximum axial rod surface temperature during and after the in-pile irradiation.

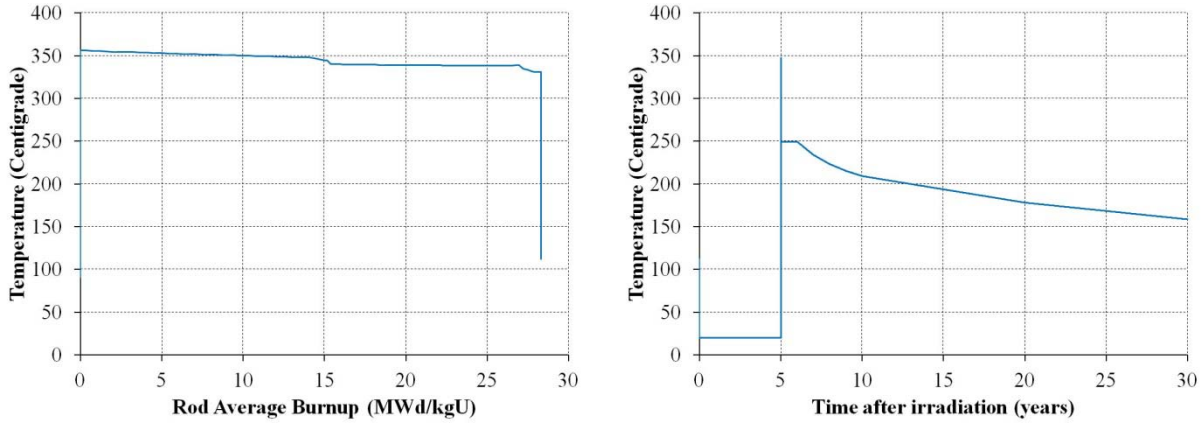
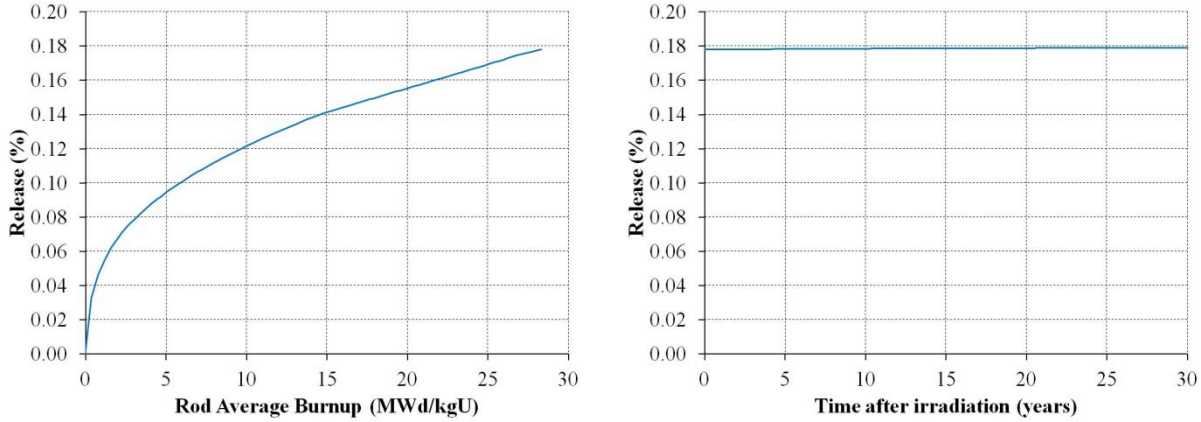


Fig. 2: Maximum axial rod surface temperature: (a) during irradiation, (b) after irradiation.

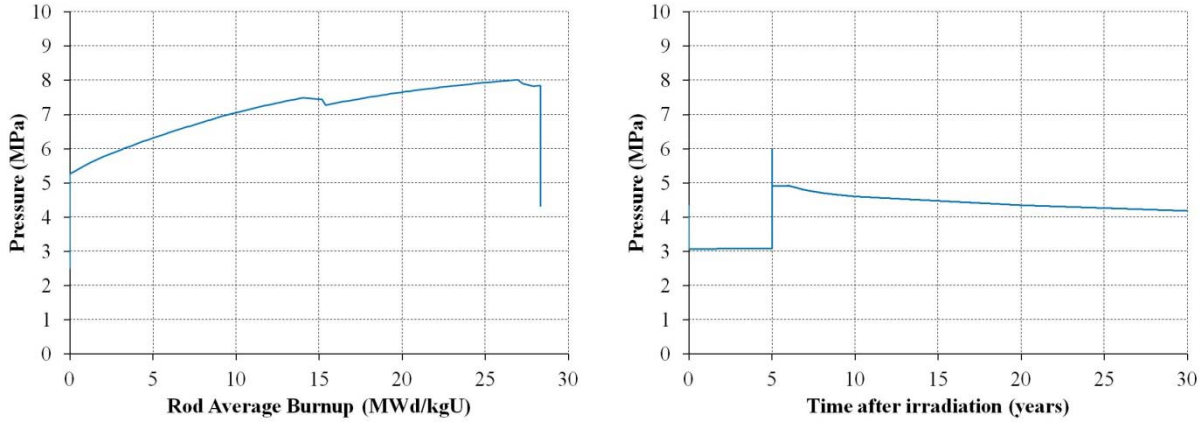
During in-pile irradiation, fission gas release may lead to the over-pressurisation of the rod. The waterside corrosion may reduce the cladding wall thickness. Both phenomena reduce the performance of the cladding. Therefore, fission gas release, rod internal pressure and the

maximum axial clad oxide thickness were evaluated with ENIGMA. They are shown in Fig. 3-a, Fig. 4-a, and Fig. 5-a respectively.



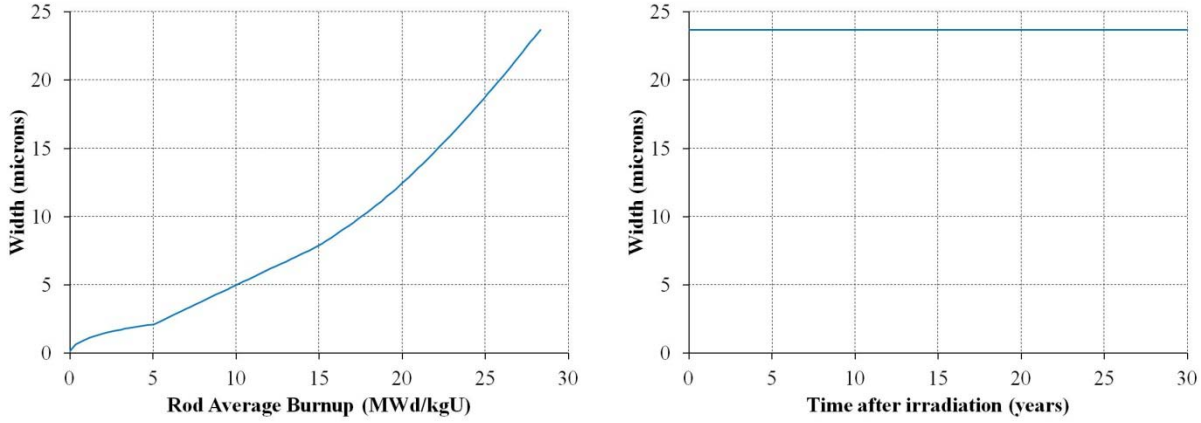
(a) (b)

Fig. 3: Fission gas release: (a) during irradiation, (b) after irradiation.



(a) (b)

Fig. 4: Rod internal pressure: (a) during irradiation, (b) after irradiation.

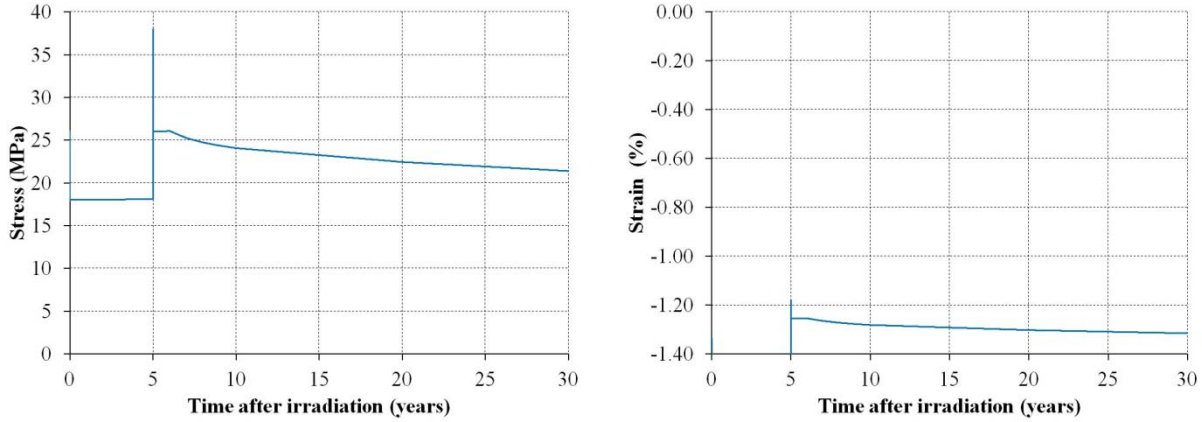


(a) (b)

Fig. 5: Maximum axial oxide thickness: (a) during irradiation, (b) after irradiation.

It is clear from the previous figures that the mild irradiation of this rod did not result in either rod over-pressurisation or excessive (>10%) thinning of the clad wall. Therefore, it is to be expected that during cask storage the maximum clad hoop stress and strain are within the safety limits as clearly shown in Fig. 6. For completeness, Fig. 3-b, Fig. 4-b, and Fig. 5-b

show the fission gas release, the rod internal pressure and the oxide thickness after the in-pile irradiation.



(a) (b)

Fig. 6: (a) Clad hoop stress, (b) Clad hoop strain.

Summary

The Studsvik CMS capability to simulate spent nuclear fuel in dry storage starting from ICFM calculations, followed by SNF and ENIGMA evaluations of safety related parameters has been described and exemplified for a fuel rod irradiated in a PWR core.

Acknowledgments

The authors want to thank Glyn Rossiter (UK National Nuclear Laboratory) for his support.

References

1. J. D. Rhodes, et al., “CASMO-5 development and applications,” PHYSOR 2006, Vancouver, Canada, 2006.

2. J. D. Rhodes, et al., “CASMO5 JENDL 4.0 and ENDF/B-VII.1 beta4 Libraries,” PHYSOR 2012, Knoxville, USA, 2012.

3. T. Bahadir, S-Ö. Lindahl, “Studsvik’s Next Generation Nodal Code SIMULATE5,” Advances in Nuclear Fuel Management IV (ANFM 2009), Hilton Head Island, USA, 2009.

4. A. Becker, et al., “SNF: Spent Fuel Analyses Based on CASMO/SIMULATE In-Core Fuel Management,” Advances in Nuclear Fuel Management IV (ANFM 2009), Hilton Head Island, USA, 2009.

5. G. Rossiter, “Development of the ENIGMA Fuel Performance Code for Whole Core Analysis and Dry Storage Assessments,” *Nuclear Engineering and Technology*, Vol. 43, No 6, December 2011.

6. R. Manteufel and N. E. Todreas, “Effective Thermal Conductivity and Edge Conductance Model for a Spent-fuel Assembly,” *Nuclear Technology*, Vol. 105, March 1994.

7. R. E. Einziger, H. Tsai, M. C. Billone and B. A. Hilton, "Examination of spent PWR fuel rods after 15 years in dry storage," NUREG/CR-6831, September 2003.

CRITICALITY SAFETY ANALYSIS OF FRESH AND SPENT FUEL STORAGE AND HANDLING FOR VVER REACTOR UNIT USING MCNP5

Vladimír Slugeň[1], Gabriel Farkas[1], Ján Haščík[1], Branislav Vrban[1], Jakub Lüley[1], Martin Petriska[1], Peter Urban[2]

[1] *Slovak University of Technology, Faculty of Electrical Engineering and Information Technology, Institute of Nuclear and Physical Engineering, Ilkovicova 3, 812 19 Bratislava, Slovakia*

[2] *SE a.s., NPP Mochovce, Slovakia*

gabriel.farkas@stuba.sk, vladimir.slugen@stuba.sk

ABSTRACT

The paper presents results of nuclear criticality safety analysis of fresh and spent fuel storage and handling for VVER reactor unit. Fresh fuel storage with transport containers, spent fuel storage pool (compact and reserve grid) and T-12 transport cask were modeled using the Monte Carlo code MCNP5. Conservative approach was applied and calculation of k_{eff} values was performed for normal and various postulated emergency conditions in order to evaluate the final maximal k_{eff} values. The requirement of current safety regulations to ensure 5% subcriticality was met except some especially conservative cases.

1 INTRODUCTION

Criticality safety associated with the packaging of spent nuclear fuel is a challenging issue for the scientific and legislative communities involved in efforts to prevent criticality accidents [3]. Safety issues associated with criticality accidents are assessed through appropriate nuclear criticality calculations which are usually performed on the assumption that the spent nuclear fuel is represented by its fresh composition. This is a simple approach, doing unnecessary any knowledge of the fuel irradiation history. However, it overlooks any possible decrease in the fuel reactivity due to the changes in fuel nuclide composition. Some of these nuclides are responsible for the decrease in the reactivity of the spent fuel. Therefore, the inclusion of these nuclides may result in a considerable improvement regarding criticality safety [6]. In this work criticality safety analysis of the spent fuel storage pool (both compact and reserve grid) was performed. Two basic loading scenarios were considered for the analysis - full loading with fresh Gd-II fuel assemblies (enrichment of 4.87 %) and partial loading with fresh and highly burned (45 and 50 MWd/kg) fuel assemblies.

2 ANALYSIS CODE AND VALIDATION

The criticality safety analysis was based on the determination of the effective neutron multiplication factor (k_{eff}) which is a key parameter for criticality safety. The continuous-

energy Monte Carlo Code MCNP5, version 1.40 and continuous-energy neutron cross section data ENDF/B-VII.0 were used [7]. Additionally, S(α,β) thermal scattering data for hydrogen in light water was applied to water and concrete. Code validation was conducted analyzing the BaW XI (2) case of the Criticality Safety Validation Suite [4, 5]. Based on this validation, bias and its uncertainty to be taken into consideration for criticality safety analysis are 0.0001 and 0.00142 respectively. The MCNP5 validation calculation was run with 200 active cycles. This number of active cycles was sufficient to rend the computation uncertainty from the MCNP5 calculation essentially negligible relative to the given benchmark uncertainty.

Table 1: MCNP5 result for BaW XI (2) case of the Criticality Safety Validation Set.

Case	**Benchmark k_{eff}**	**Calculated k_{eff} ENDF/B-VII.0**
BaW XI (2)	1.0007 ± 0.0012	1.0006 ± 0.00076

The bias and its uncertainty were calculated according formulas:

$$\Delta_{bias} = k_{eff}^{bench} - k_{eff}^{calcul}$$

$$\sigma_{bias} = \sqrt{\sigma_{bench}^2 - \sigma_{calcul}^2}$$

where:

Δ_{bias}	is	the bias,
σ_{bias}	is	the bias uncertainty derived from the code validation,
k_{eff}^{bench}	-	the benchmark (experimental) k_{eff},
k_{eff}^{calcul}	-	the calculated k_{eff},
σ_{bench}	-	uncertainty of the benchmark k_{eff} value,
σ_{calcul}	-	uncertainty of the calculated k_{eff} value.

3 EVALUATION METHOD

Conservative approach was applied and calculation of k_{eff} values was performed for normal and various postulated emergency conditions in order to evaluate the final maximal k_{eff} values. All conditions improving neutron multiplication in the storage pool were taken into account. Selected conservative parameters are listed in Table 2.

Table 2: Selected conservative parameters

No	**Parameter**	**Nominal value**	**Tolerance**	**Conservative value**
1	Lattice pitch of fuel pins	12.3 mm	± 0.12 mm	12.42 mm
2	Lattice pitch of absorption tubes of the compact grid	162 mm	± 0.842 mm	161.158 mm
3	Lattice pitch of the reserve grid	225 mm	± 0.842 mm	224.158 mm

4	Lattice pitch of hermetic tubes of the compact grid	230 mm		230 mm
5	Average fuel enrichment of fresh Gd-II fuel assembly	4.87 w%	± 0.05 w%	4.92 w%
6	Gd_2O_3 ratio in the fuel	3.35 w%	± 0.15 w%	3.2 w%
7	Uranium mass in the FA	126.3 kg	± 1.9 kg	128.2 kg
8	Boron content of NEUTRONIT steel of the compact grid	1.05 – 1.2 %		1.05 %
9	Coolant temperature in the compact grid of the storage pool	50 °C		4 °C
10	Coolant temperature in the reserve grid of the storage pool	50 °C		100 °C

The maximal effective multiplication factor $k_{eff}^{\max}$ was evaluated as a sum of the calculated conservative k_{eff}, the systematic error Δ_{bias}, and the combined uncertainty multiplied by 1.645 which is the one-sided tolerance limit factor for a normal distribution at 95% probability.

$$k_{eff}^{\max} = k_{eff}^{conser} + \Delta_{bias} + 1.645\sqrt{\sigma_{bench}^2 + \sigma_{calcul}^2 + \sigma_{conser}^2}$$

where:

k_{eff}^{conser} is the calculated conservative k_{eff},

σ_{conser} - uncertainty of the calculated conservative k_{eff}.

4 CALCULATION MODEL

The following detailed models were developed in the MCNP5 for criticality safety analysis:

- model of compact grid of the spent nuclear fuel storage pool, Figure 1 – 5,
- model of reserve grid of the spent nuclear fuel storage pool, Figure 6.

MCNP model of the compact grid consists of 603 hexagonal absorption tubes filled with profiled Gd-II fuel assemblies with the enrichment of 4.87 %, 54 hermetic tubes, supporting plate, and concrete well. Nominal lattice pitch of the absorption tubes represents 162 mm and 230 mm for the hermetic tubes. The reserve grid model consists of 296 fuel assemblies, 54 hermetic tubes, supporting plate, and concrete well.

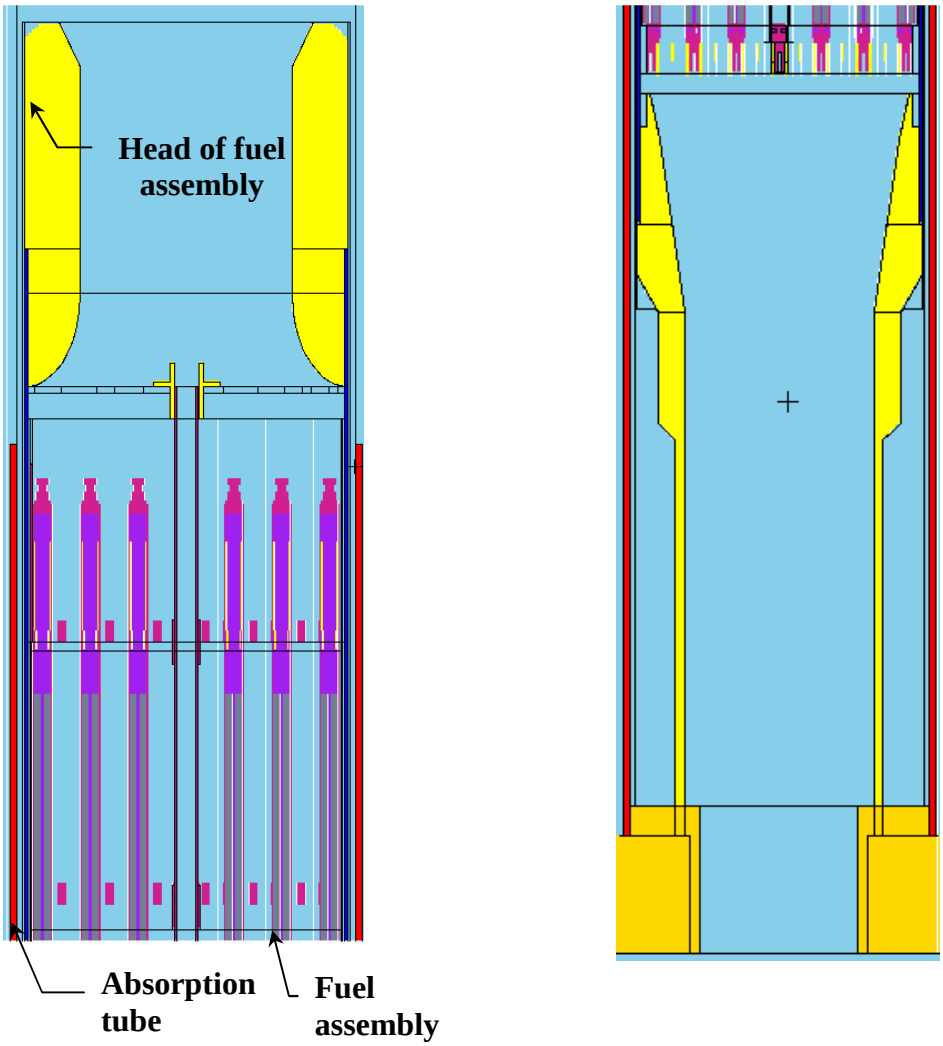


Figure 1: MCNP model of the hexagonal absorption tube filled with FA – vertical cross section.

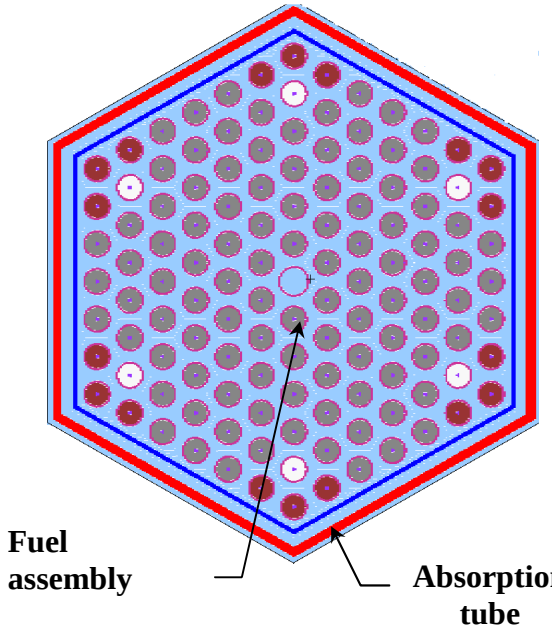


Figure 2: MCNP model of the hexagonal absorption tube filled with FA – horizontal cross section.

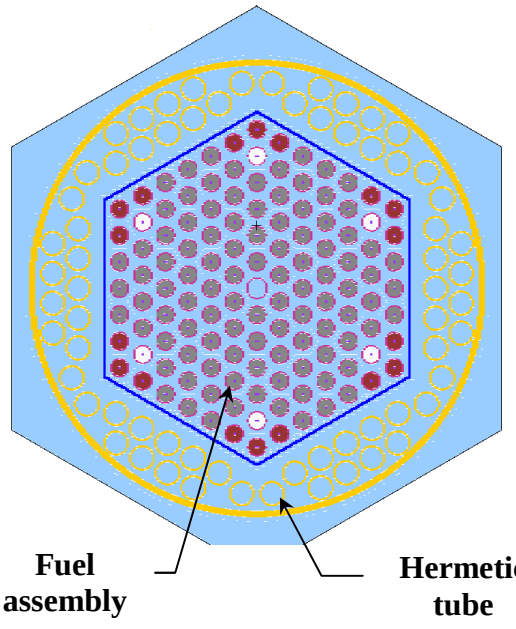


Figure 3: MCNP model of the hermetic tube filled with FA - horizontal cross section.

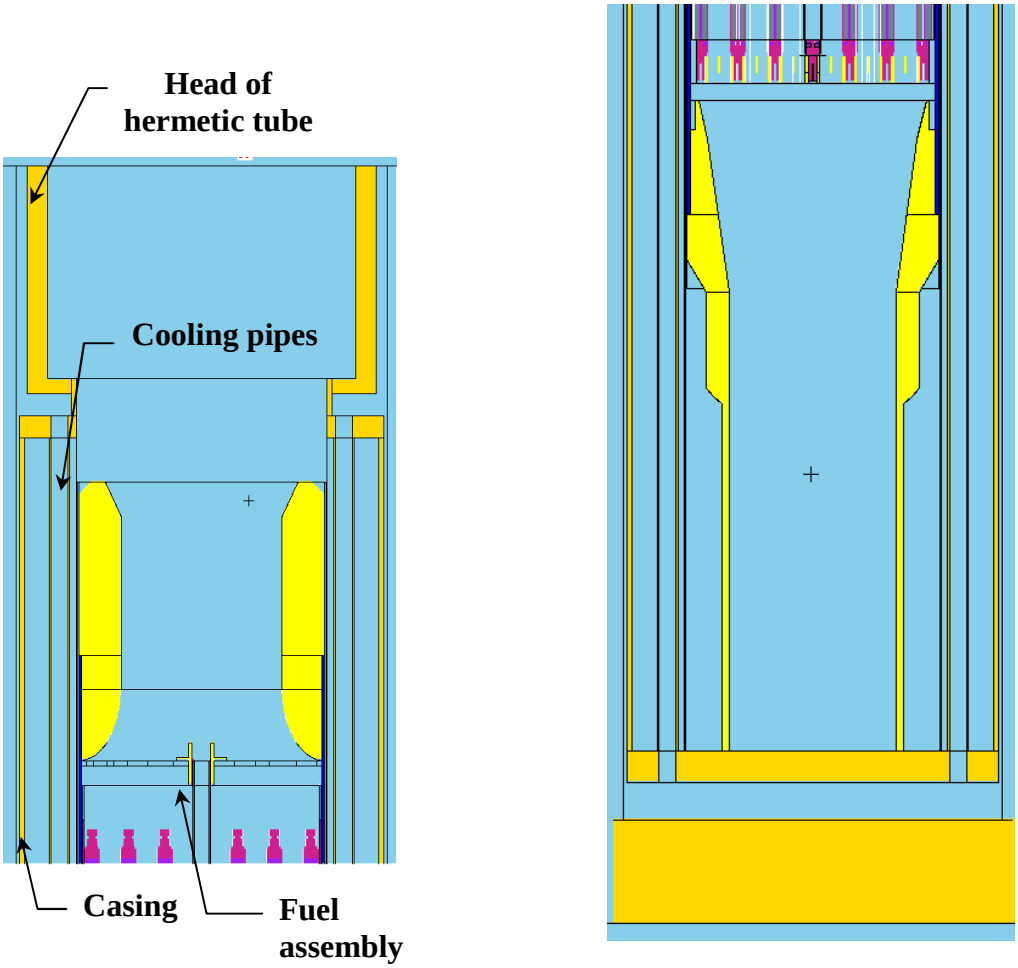


Figure 4: MCNP model of the hermetic tube with FA – vertical cross section.

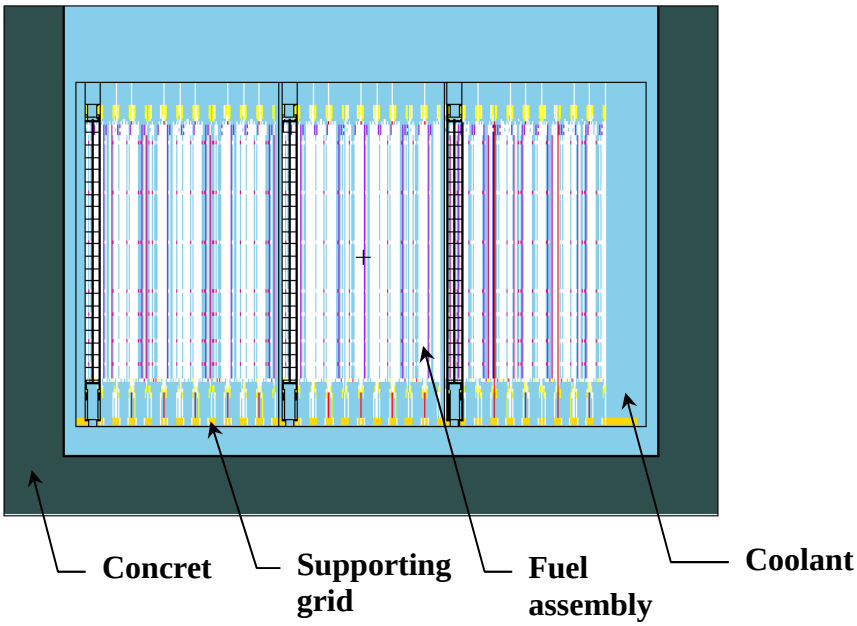


Figure 5: MCNP model of compact grid of the spent fuel storage pool – vertical cross section.

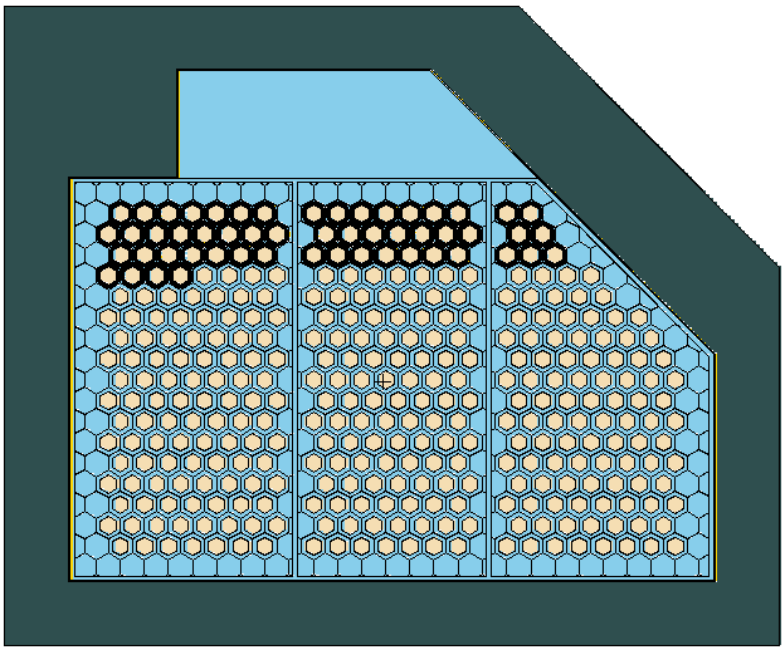


Figure 6: MCNP model of reserve grid of the spent fuel storage pool – horizontal cross section. (Variant R1– full loading with 4.87 % enriched FAs, variant R2 – full loading with 45 MWd/kg burned FAs).

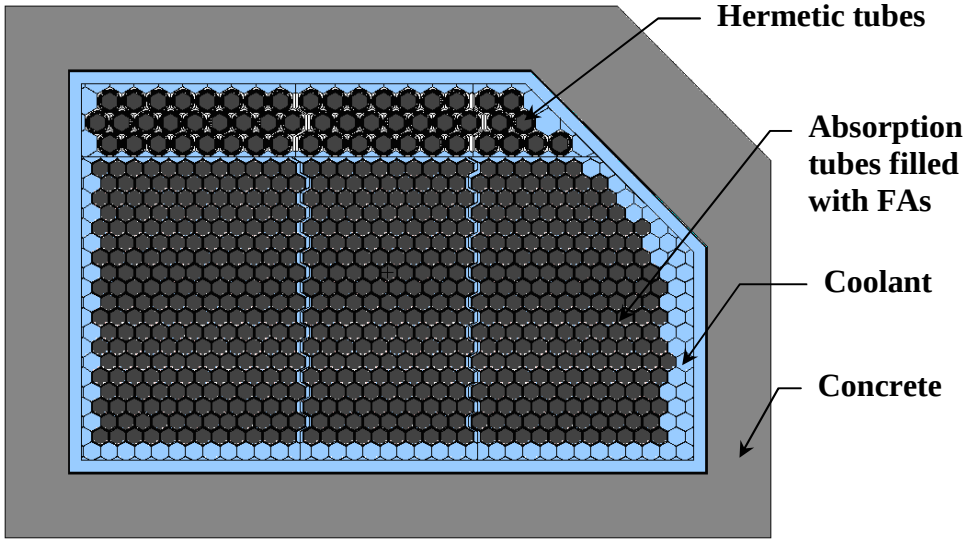


Figure 7: MCNP model of compact grid of the spent fuel storage pool – horizontal cross section. (Variant A – all positions loaded with 4.87 % enriched fresh FAs.)

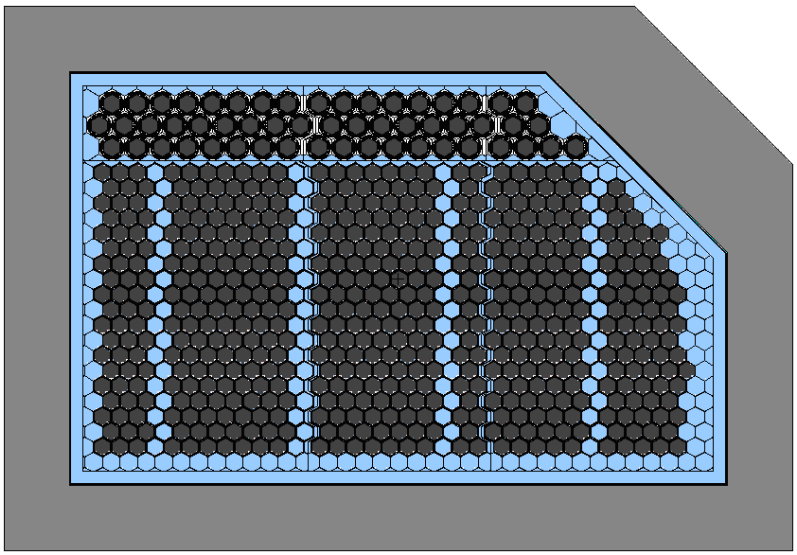


Figure 8: MCNP model of compact grid of the spent fuel storage pool – horizontal cross section. (Variant B – loading with 4.87 % enriched fresh FAs and four empty rows.)

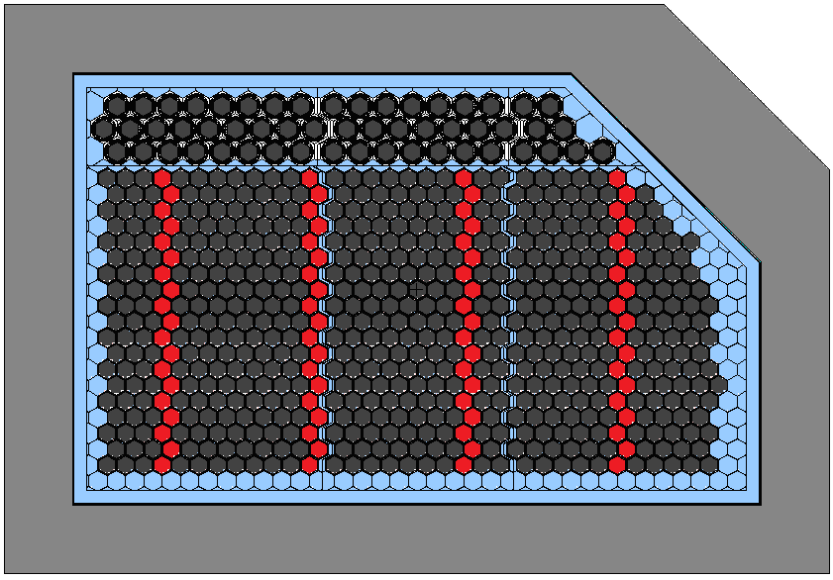


Figure 9: MCNP model of compact grid of the spent fuel storage pool – horizontal cross section. (Variant C1 (D1) – loading with 4.87 % enriched FAs and four rows of 45 MWd/kg (50 MWd/kg) burned FAs.)

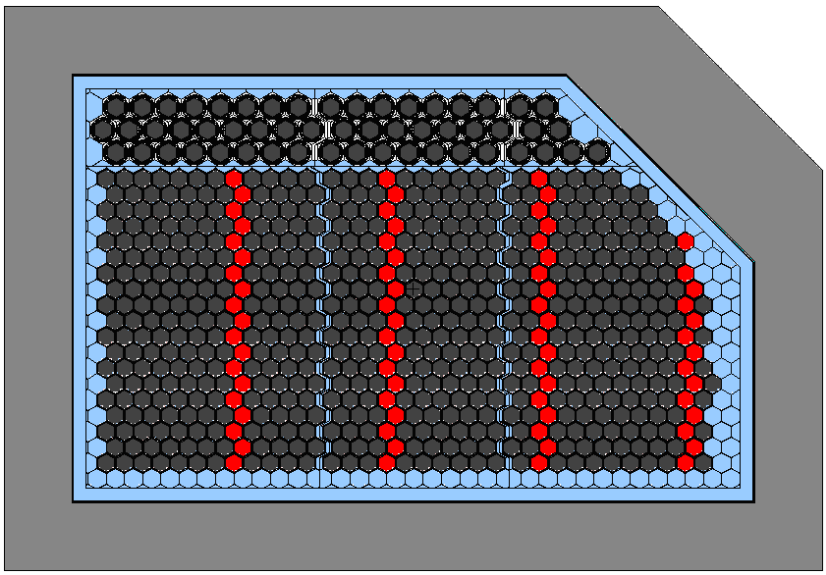


Figure 10: MCNP model of compact grid of the spent fuel storage pool – horizontal cross section. (Variant C2 (D2) – loading with 4.87 % enriched FAs and four rows of 45 MWd/kg (50 MWd/kg) burned FAs.)

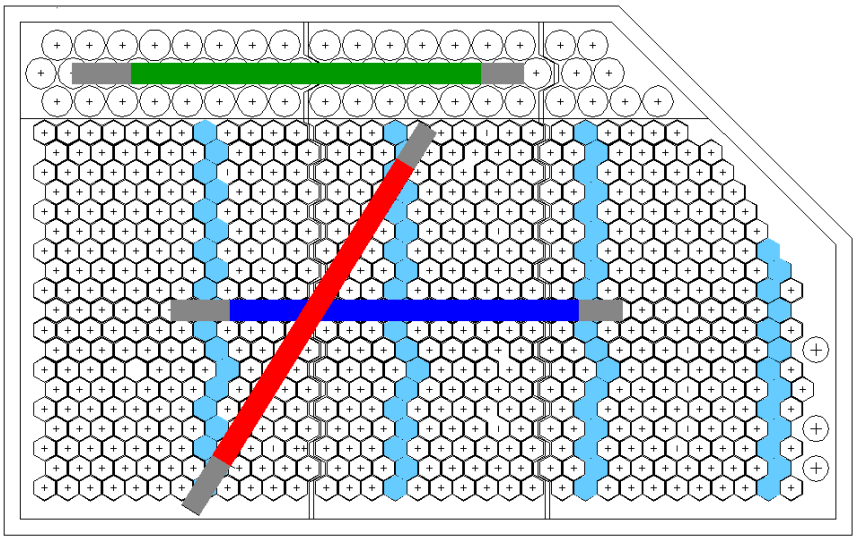
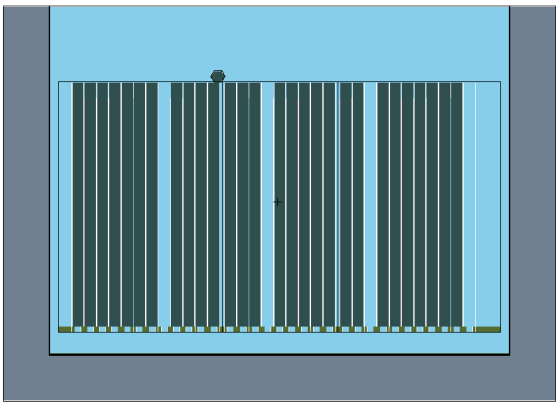


Figure 11: Compact grid with positions of fallen fuel assembly.
(red – diagonal downfall, blue – longitudinal downfall, grenn – downfall on the hermetic tubes)



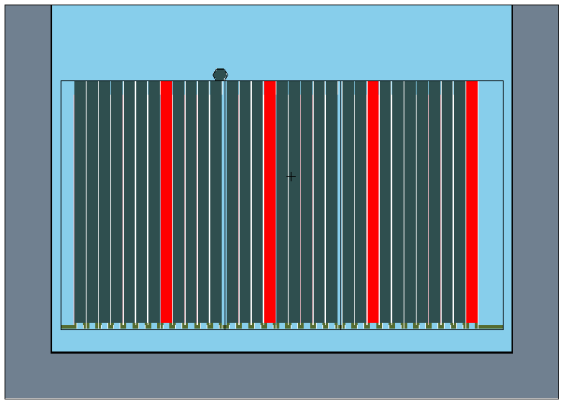


Figure 12: MCNP model of fully loaded compact grid with fresh FAs, four empty rows, and diagonally fallen fresh fuel assembly – variant E11.

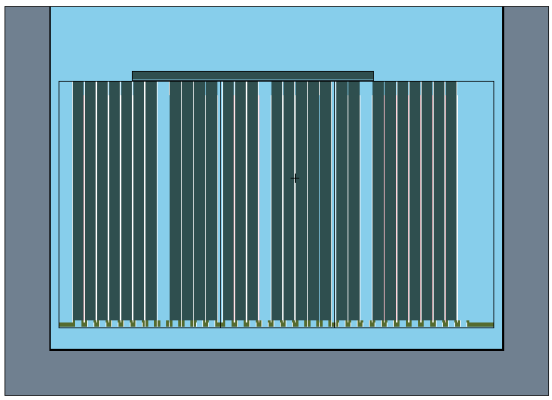


Figure 13: MCNP model of fully loaded compact grid with fresh FAs, four rows of 45 MWd/kg burned FAs, and diagonally fallen fresh fuel assembly – variant E12.

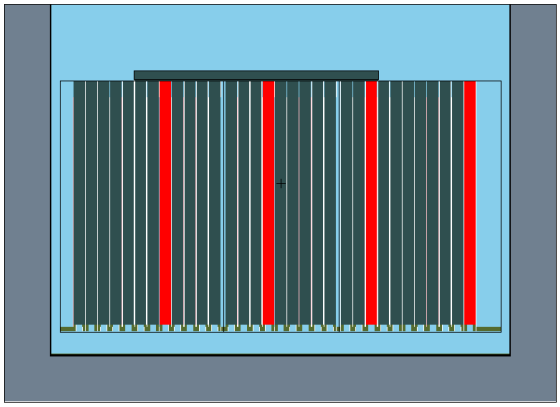


Figure 14: MCNP model of fully loaded compact grid with fresh FAs, four empty rows, and longitudinally fallen fresh fuel assembly – variant E21.

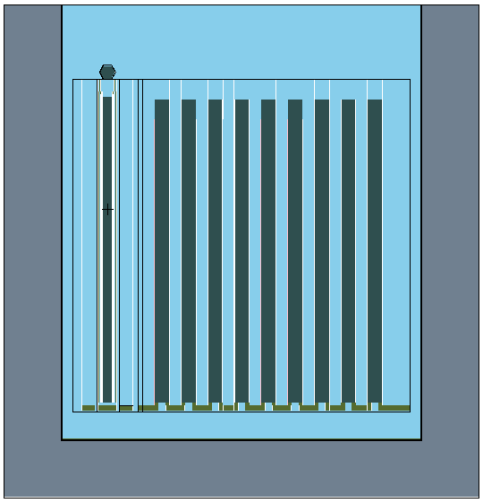


Figure 15: MCNP model of fully loaded compact grid with fresh FAs, four rows of 45 MWd/kg burned FAs, and longitudinally fallen fresh fuel assembly – variant E22.

Figure 16: MCNP model of the compact grid, and longitudinally fallen fresh fuel assembly on the hermetic tubes – variant E31 and E32.

5 RESULTS

Concerning criticality safety analysis of the compact grid, the following basic variants of storage pool loading were investigated:

- variant A - all positions loaded with 4.87 % enriched fresh FAs (Figure 7),
- variant B - loading with 4.87 % enriched fresh FAs and four empty rows (Figure 8),
- variant C1 - loading with 4.87 % enriched FAs and four rows of 45 MWd/kg burned FAs (Figure 9),
- variant C2 - loading with 4.87 % enriched FAs and four rows of 45 MWd/kg burned

FAs (Figure 10),

- variant D1 - loading with 4.87 % enriched FAs and four rows of 50 MWd/kg burned FAs (Figure 9).
- variant D2 - loading with 4.87 % enriched FAs and four rows of 50 MWd/kg burned FAs (Figure 10).

Table 3: Results of the criticality safety analysis for compact grid (basic variants).

Variant	k_{eff}^{conser}	σ_{conser}	$k_{eff}^{\max}$
A	0.95136	0.00004	**0.95520**
B	0.93672	0.00004	**0.94056**
C1	0.94152	0.00007	**0.94397**
C2	0.94064	0.00007	**0.94309**
D1	0.94042	0.00007	**0.94287**
D2	0.93949	0.00007	**0.94194**

In terms of reserve grid of the spent fuel storage pool, two variants were taken into consideration:

- variant R1 - all positions loaded with 4.87 % enriched fresh FAs (Figure 6),
- variant R2 - all positions loaded with 45 MWd/kg burned FAs (Figure 6).

Table 4: Results of the criticality safety analysis for reserve grid.

Variant	k_{eff}^{conser}	σ_{conser}	$k_{eff}^{\max}$
R1	0.90877	0.00003	**0.91261**
R2	0.73597	0.00005	**0.73842**

In order to evaluate the influence of fallen fresh FA into the compact grid on $k_{eff}^{\max}$ value, the following six emergency conditions were investigated:

- variant E11 - loading with 4.87 % enriched FAs, four empty rows, and diagonally fallen fresh FA (Figure 12),
- variant E12 - loading with 4.87 % enriched FAs, four rows of 45 MWd/kg burned FAs, and diagonally fallen fresh FA (Figure 13),
- variant E21 - loading with 4.87 % enriched FAs, four empty rows, and longitudinally fallen fresh FA (Figure 14),
- variant E22 - loading with 4.87 % enriched FAs, four rows of 45 MWd/kg burned FAs, and longitudinally fallen fresh FA (Figure 15),
- variant E31 - loading with 4.87 % enriched FAs, four empty rows, and longitudinally fallen fresh FA on the hermetic tubes (Figure 16),
- variant E32 - loading with 4.87 % enriched FAs, four rows of 45 MWd/kg burned FAs, and longitudinally fallen fresh FA on the hermetic tubes (Figure 16).

Table 5: Results of the criticality safety analysis for compact grid (emergency variants).

Variant	k_{eff}^{conser}	σ_{conser}	$k_{eff}^{\max}$
E11	0.93640	0.00007	**0.93885**
E12	0.94069	0.00007	**0.94314**

E21	0.93631	0.00010	**0.93877**
E22	0.94057	0.00010	**0.94303**
E31	0.93645	0.00007	**0.93890**
E32	0.94046	0.00007	**0.94291**

6 CONCLUSION

Criticality issues associated with compact and reserve grid of the spent fuel storage pool, located in the NPP Mochovce 1 and 2, were investigated using MCNP5. The criticality safety analysis focused on the evaluation of maximal neutron multiplication factor values at normal and some emergency conditions applying conservative approach. The outcomes of the investigations showed that the requirement of current safety regulations to ensure 5 % subcriticality was met (including postulated emergency conditions), except one especially conservative case of the fully loaded compact grid with fresh 4.87 % enriched Gd-II FAs. Only in this case, the calculated $k_{eff}^{\max}$ value exceeded the required subcriticality limit of 0.95 by 0.55 %. Except this one scenario the analyses showed that nuclear criticality safety criteria in terms of the spent fuel storage pool are satisfied.

REFERENCES

[1] F. B. Brown, "Theory & Practice of Criticality Calculation with MCNP5", LA-UR-08-0849, Los Alamos National Laboratory (2008)

[2] F. B. Brown, B. Nease, & J. Cheatham, "Convergence Testing for MCNP5 Monte Carlo Eigenvalue Calculations", M&C+SNA-2007, ANS Mathematics & Computation Topical Meeting Monterey, CA, 15-19 April 2007, LA-UR-07-1123 (2007)

[3] Implementation of Burnup Credit in Spent Fuel Management Systems. International Atomic Energy Agency, IAEA-TECDOC-1241, Vienna (2000)

[4] International Handbook of Evaluated Criticality Safety Benchmark Experiments, NEA/NSC/DOC(95)3, OECD Nuclear Energy Agency, ISBN 978-92-64-99140-8 (2006)

[5] R. D. Mosteller, "ENDF/B-VII.0, ENDF/B-VI, JEFF-3.1, and JENDL-3.3 Results for the MCNP Criticality Validation Suite and Other Criticality Benchmarks", LA-UR-07-6284, Los Alamos National Laboratory (2007)

[6] G. Nicolaou, N. Tsagas, "Criticality Safety of Spent Nuclear Fuel Assemblies from the Transmutation of Minor Actinides in Fast Reactors", Annals of Nuclear Energy, 33 (2006), pp. 305 – 309

[7] X-5 Monte Carlo Team, "MCNP – A general N-Particle Transport Code, Version 5 – Volume I: Overview and Theory", LA-UR-03-1987, Los Alamos National Laboratory (April 2003)

Top Fuel 2012 organised by ENS
2-6 September 2012 in Manchester, UK

IAEA's cooperation in supporting long term interim storage and back-end management of research reactor fuel

S. Tozser, P. Adelfang, E. Bradley
International Atomic Energy Agency,
PO Box 100, 1400 Vienna – Austria

ABSTRACT

The IAEA has always striven to give appropriate attention for each stage of the fuel cycle, including back-end management of research reactor (RR) spent fuel. Although the international activities in the back-end of the RR nuclear fuel cycle have so far been dominated by the acceptance of RR spent nuclear fuel (SNF) programmes by the country where it was originally enriched. However the collection and dissemination of practices on long term interim storage of RR SNF, as well as back-end options on a commercial basis have remained underlined objectives and hence a continuous effort on IAEA activities.

The paper gives an overview of IAEA's cooperation with take-back programmes made under the auspices of GTRI. This in turn involves the achievements made by both separate take-back programmes for US and Russian origin fuels with special emphasis on the IAEA's contribution. It also presents on-going activities for the collection and dissemination of best practices and lessons learned on long term interim storage as well as on back-end solution of RR SNF, harnessing the knowledge and practices that the international community gained by accomplishing the SNF take-back programmes.

1. Introduction

After fuel is discharged from the core, RR Spent Nuclear Fuel (RRSNF) is stored under water, usually at-reactor (AR) facilities for a period of time for cooling. This wet storage can be extended, in AR or away-from-reactor (AFR) facilities, in some cases over long periods (~50 years), or the RRSNF may be transferred to dry storage (AR or AFR) sites and stored dry for even longer periods [1].

However storage cannot be the end point for RRSNF. At present, some countries, with fuel of US or Russian origin, are taking advantage of the international research reactor (RR) spent fuel take-back programmes, the USA Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) acceptance programme and the Russian Research Reactor Fuel Return (RRRFR) programme. The major goal of the take-back programmes supported by the US DOE NNSA GRTRI is to eliminate inventories of Highly Enriched Uranium (HEU) [2].

The IAEA has always striven to give appropriate attention for each stage of the fuel cycle, including back-end management of RR spent fuel. Although the international activities in the back-end of the RR nuclear fuel cycle have so far been dominated by the acceptance of RR spent nuclear fuel (SNF) programmes by the country where it was originally enriched; the collection and dissemination of practices on long term interim storage of RR SNF, as well as back-end options on a commercial basis have remained underlined objectives and hence a continuous effort on IAEA activities.

2. Back-end option for RRSNF

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

Reprocessing is a proven technology for dealing with standard types of MTR fuels, and some countries have taken advantage of this option. However, it has not been used for small quantities of some experimental spent fuels, and even for some standard types of fuel, e.g. TRIGA type fuel, where it has only been carried out on a demonstration scale.

Return of the fuel to the country where it was originally enriched, has been selected as the preferred option by countries that are eligible for taking advantage of the two international RRSNF take-back programmes; the USA Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) acceptance programme and the Russian Research Reactor Fuel Return (RRRFR) programme. Many countries are taking advantage of these programmes [3-4]. However, these programmes are only available for USA and Russian origin fuels, and some additional fuels classified as "gap material".

Final disposal is a very expensive solution that can be afforded by only a few countries. For countries with no nuclear power programme, the construction of geological repositories for the relatively small amounts of spent fuel from one or two research reactors is obviously not practicable.

Extended interim wet storage of RRSNF. RRSNF can be stored, in AR or AFR wet facilities, for long periods. However, to keep the RRSNF in wet storage, it is essential to have a suitable storage structure and a very strong programme for maintenance of the water quality. Most research reactor storage pools were designed as service pools where the fuel would be kept only for about 10 years, considering that another option would be available after that time. They were typically not designed and constructed with sufficient support systems to facilitate long term interim storage, and consequently, many cases of poor water conditions in storage pools have been reported; in some cases causing severe degradation of the fuel cladding [5].

Interim dry storage includes the storage of spent fuel in vaults or in casks. Vaults consist of reinforced concrete buildings (or modulus) containing arrays of storage cavities suitable for containment of one or more fuel units. A cask is a sealed metal cylinder containing the spent nuclear fuel, and which may or may not be easily transported. Sometimes a hot cell is used for the loading and drying operation, and attention must be paid to the security of the RRSNF after the reactor reaches its lifetime.

Semi-dry storage means the fuel stored under water (interim dry storage) undergoes an encapsulation technology and the canned fuel, as a package is taken back mostly in the same water pool. The encapsulation technology typically uses a tube-type capsule made of the same material as the fuel cladding. The capsule is capable of accommodating one or, sometimes more fuel assemblies that are placed into the capsule "as they are". Prior to closing the capsule, the package (capsule with fuel assemblies) is dried, then filled with slightly pressurized inert gas (e.g. with dry nitrogen gas) and the package is closed hermetically. As the facilities typically do not have dry storage capability for SNF, as a practical solution the packages are taken back by the same AFR pool where the fuel was stored before encapsulation (this is why the storage mode is named as semi-dry). Facilities applying encapsulation claim that this storage mode ensures a safe extended storage of SNF for another 50 years even for the fuel that previously exhibited significant corrosion.

3. Support to the Research Reactor Fuel Take Back Programmes

The IAEA is an active supporter of the effort(s) to return RR fuel to the country of origin [1].

3.1. U.S. Take Back Programme

The US. take back programme was re-started in 1996 under the name of Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Acceptance Programme [1]. The strategic goal was to remove 1364 kilograms of HEU and LEU to the USA for disposition by 2013. Until now 57 shipments from 30 countries have been completed. The total amount of the removed fuel was 1244 kilogram (91 % of the targeted volume is completed) [6].

The Agency did not ensure a direct support activity in FRRSNF, instead began a number of activities to assist member states eligible to ship spent RR fuels back to the U.S. The IAEA convened experts to

develop a document with guidelines for Member States in this regard, which resulted in the "Guidelines Document on Preparatory Work Prior to Return of Spent Fuel of US-Origin from Foreign Research Reactors" [7]. The IAEA TECDOC 1593, "Return of research reactor spent fuel to the country of origin: national experiences and requirements for technical and administrative preparations" contains much about good practices from operators' participation in shipment preparations [8].

3.2. Russian Research Reactor Fuel Return (RRRFR) Programme

The preparation for the Russian Research Reactor Fuel Return (RRRFR) programme started in December 1999 [9], when at the IAEA General Conference in September, the U.S. Energy Secretary, Bill Richardson announced that the US was prepared to work with Russia and the IAEA to manage and dispose of Russian-origin HEU RR fuel remaining in a number of countries. Then in October 2000 the IAEA's Director General sent a letter to the governments of relevant countries for the elimination of HEU fuel from Soviet RRs. Fourteen out of sixteen responses were favourable, concerning 20 RRs, which led to the launching of the RRRFR programme in 2001.

3.2.1. Shipments accomplished

Since the first shipment made in August 2002, the RRRFR programme successfully completed 45 shipments of more than 1.7 tons of fresh and spent HEU fuel from different countries using Russian fuelled research reactors to the country of origin.

In the case of fresh shipments from 2002 to 2010 under contract agreements by the IAEA, 22 shipments representing a total amount of about 712 kilograms of fresh HEU were returned safely to the Russian Federation. The shipments are listed in Table 1 in a chronological order.

Table 1. Fresh RR HEU fuel returned to Russia under IAEA contracts

No.	Country	Facility	Container used	Mode of transport	U-mass [kg]	Actual Finish
1	Serbia	RA, Vinča	TK-S16	Air transport	48.0	2002-08-08
2	Romania	WWR-S Magurela	TK-S16	Air transport	14.0	2003-09-30
3	Bulgaria	IRT-2000, Sofia	TK-S16	Air transport	17.0	2003-12-23
4	Libya	IRT-1 Tajura	TK-S16	Air transport	17.0	2004-03-07
5	Uzbekistan	WWR-SM Tashkent	TK-S16	Air transport	3.0	2004-09-09
6	Czech Republic	LWR-15, Rez	TK-S16	Air transport	6.0	2004-12-21
7	Latvia	IRT-M, Salaspils	TK-S16	Air transport	3.0	2005-05-25
8	Czech Republic	CA, CTU Prague	TK-S16	Air transport	14.0	2005-09-27
9	Libya	IRT-1 Tajura	TK-S16	Air transport	3.0	2006-07-25
10	Poland	MARIA	TK-S16	Air transport	39.8	2006-08-10
11	Czech Republic	Rez	TK-S16	Air transport	0.2	2006-10-15
12	Germany	RRR	TK-S16	Air transport	268.0	2006-12-18
13	Poland	MARIA	TK-S16	Air transport	8.8	2007-08-28
14	Vietnam	Dalat	TK-S16	Air transport	4.0	2007-09-17
15	Romania	Pitesti	TK-S16	Air transport	30.0	2009-06-28
16	Hungary	BRR	TK-S16	Air transport	18.6	2009-07-06
17	Czech Republic	Rez	TK-S16	Air transport	12.2	2010-06-18
18	Belarus	Minsk. Pamir fuel	TK-S16	Air transport	46.7	2010.11.29
19	Ukraine	Sevastopol	TK-S16	Air transport	25.1	2010.12.29
20	Ukraine	KINR	TK-S16	Air transport	9.8	2010.12.29
21	Ukraine	Kharkov 1st	TK-S16	Air transport	15.7	2010.12.29
22	Ukraine	Kharkov 2nd	TK-S16	Air transport	108.6	2012.03.21
Last update: 2012-05-09				**TOTAL**	**712.3**	

Table 2 shows shipments of RRSNF carried out under the RRRFR programme in chronological order. Since 2006, altogether 23 shipments from RR sites to RF were safely and successfully accomplished - a total amount of about 1006.4 kg HEU SNF removal. The first, so called "pilot shipment" was accomplished in January 2006 followed by three other SNF transports from Uzbekistan. Russian type TUK-19 casks were used for the first four shipments, while later the newly developed Skoda type VPVR/M casks were also used for the shipments.

Table 2. Spent RR HEU fuel returned to Russia

No.	Country	Facility	Container used	Mode of transport	U-mass [kg]	Actual Finish
1	Uzbekistan	WWR-SM Tashkent	TUK-19	RW	10.0	2006-01-10
2	Uzbekistan	WWR-SM Tashkent	TUK-19	RW	13.0	2006-02-14
3	Uzbekistan	WWR-SM Tashkent	TUK-19	RW	14.0	2006-03-20
4	Uzbekistan	WWR-SM Tashkent	TUK-19	RW	26.0	2006-04-15
5(*)	Czech Republic	Rez	VPVR/M	RW	80.0	2007-11-29
6	Latvia	Salaspils	TUK-19	RW	14.4	2008-05-12
7	Bulgaria	Sofia	VPVR/M	RW	6.3	2008-07-04
8	Hungary	BRR	VPVR/M	PR-RW-**SV**-RW	154.5	2008-10-10
9	Kazakhstan	Alatau	TUK-19	RW	17.3	2008-12-25
10	Kazakhstan	Alatau	TUK-19	RW	16.6	2009-03-01
11	Kazakhstan	Alatau	TUK-19	RW	18.8	2009-04-01
12	Kazakhstan	Alatau	TUK-19	RW	21.0	2009-05-01
13	Romania	Magurele	TUK-19	AT	23.7	2009-06-29
14(*)	Poland	EWA	VPVR/M	PR-RW-**SV**-RW	190.1	2009-09-13
15	Libya	Tripoli	TUK-19	AT	5.2	2009-12-21
16(*)	Poland	EWA, MARIA	TUK-19, VPVR/M	PR-RW-**SV**-RW	139.2	2010-03-18
17(*)	Poland	MARIA	TUK-19	PR-RW-**SV**-RW	49.5	2010-05-23
18(*)	Ukraine	KINR	VPVR/M	PR-**RW**	55.9	2010-05-25
19(*)	Poland	MARIA	TUK-19	PR-RW-**SV**-RW	38.6	2010-07-24
20(*)	Poland	MARIA	TUK-19	PR-RW-**SV**-RW	37.5	2010-10-10
21	Belarus	Minsk. Pamir	VPVR/M	RW	42.0	2010-10-24
22(*)	Serbia	Vinča RA	TUK-19, VPVR/M	PR-RW-**SV**-RW	13.2	2010-12-17
23)	Uraine	KINR	VPVR/M	PR-RW-**SV**-RW	19.6	2012-03-25
Last update: 2012-05-09				**TOTAL**	**1006.4**	

(*) = IAEA involvement; PR = Public Road (highway, truck); RW = railway; SV = seagoing vessel; AT = air transport

3.2.2. Programme specific technical cooperation

In general, the IAEA's role in supporting projects like the RRRFR programme is threefold: (1) verification made by Safeguards; (2) standardization ensured by IAEA Safety Standards (e.g. Nuclear Safety, Transport, Emergency preparedness, waste management standards, etc.); and (3) technical cooperation ensuring multidisciplinary backing for MSs throughout technical cooperation mechanisms. The first two supports are a continuous IAEA service for the MSs. From the RRRFR programme's viewpoint, the third group plays a significant role through which programme specific support is provided.

Within the framework of the IAEA's Technical Cooperation projects, two significant subject specific projects were launched: (1) Skoda VPVR/M cask procurement; (2) Vinča (Serbia) SNF return programme.

Skoda VPVR/M cask procurement. Transporting a large quantity of SNF stored at many of Russian origin RRs required to develop suitable new capacity packages for the RRRFR programme to haul all of the stored SNF with one shipment from some facilities (at the beginning 16 pcs. TUK-19 casks were available), and in addition ensure further transport package alternatives with an improved cask

loading technology to meet the needs of the different RR site and SNF conditions stored at a facility. To assist in meeting this requirement, the IAEA agreed to use its procurement system to send out a call for bid and procure enough casks to meet the foreseen shipment needs. After the bidding procedure the VPVR/M cask system of Skoda (Czech Republic) was selected from six international cask vendors [10].

The IAEA procured ten high capacity dual purpose (transport/storage) containers under a €4 million contract. The complex procurement and implementation included outlining the technical requirements, evaluation of bids, contracting, quality inspections, evaluation of the results of the so called "dry run" and "wet run" tests [11]. Thus, the programme now has 16 VPVR/M casks and 16 TUK-19 casks.

Vinča (Serbia) SNF return programme. Since 2004 the IAEA, the Nuclear Threat Initiative (NTI), the US-DOE and the European Union have provided funds to cover the Vinča RA Reactor SNF removal. With this financial support, upon the invitation of the IAEA, in May 2005 an international consultancy meeting was held in the Vinča Institute that resulted to issue an international tender for safe removal of Vinca fuel. Following the selection procedure, an RF consortium, Sosny-Mayak-Tenex, was selected and an international tripartite contract between IAEA, Sosny and Vinča Institute was signed in September 2006 for the safe removal of spent nuclear fuel (SNF) from the Vinča RA Research Reactor and return to the Russian Federation.

For the implementation of the tripartite contract, to be consistent with the TC management principle, a special PMO was appointed by the IAEA to coordinate the programme implementation. During the programme performance, 16 technical officers, and two technical experts were assigned to the Project Management Unit at Vinča site. Thus, the IAEA not only contracted, but provided a general coordinative managerial support, including an overall technical backing for the operating organisation and the officers of the regulatory body [12].

The project was completed as planned in December 2010: 8030 SNF was removed representing more than two-and-half tonnes of highly radioactive spent fuel [13]. This transport was the largest single shipment of SNF made under the RRRFR programme, and it became the largest and most complex TC project in the history of the IAEA with a total budget of over US$ 55 million.

3.2.3. Collecting and dissemination practices

Regional lessons learned workshops. As the first shipments were completed, in 2005, the IAEA in cooperation with the US DOE initiated a yearly regional workshop on "RRRFR Programme Lessons Learned". The primary objective was – and still is – to bring together the core players in the preparation and accomplishment of shipments, and sharing experiences on lessons learned so that others may benefit in the future. Table 4 shows the history of the Regional Workshops. Although the meeting indicated in the second row was a workshop on "International Legal Framework Applicable for Shipment of Russian-origin Research Reactor Spent Fuel to the Russian Federation", it replaced the annual regional workshop in 2007, but its main feature was gathering experience. Thus, altogether six workshops on lessons learned were organised.

Table 4. History of the Regional Workshops on RRRFR programme Lessons Learned"

No	Place	Date	Participants
1	Belgrade, Serbia	October 2006	75 participants from 15 countries
	Poina-Brasov, Romania(1)	April 2007	43 participants from 10 countries and EU
2	Rez, Czech Republic	May 2008	97 participants form 17 countries
3	Varna, Bulgaria	June 2009	88 participants form 17 countries
4	Poina-Brasov, Romania	May 2010	71 participants from 16 countries
5	Jackson, WY-USA	June 2011	95 participants from 17 countries
6	*Lake Balaton, Hungary*	*Scheduled for 2012*	

(1): It was a Workshop on "International Legal Framework Applicable for Shipment of Russian-origin Research Reactor Spent Fuel to the Russian Federation" organised by the IAEA in cooperation with the European Union.

IAEA-TECDOC booklets issued supporting the RRRFR programme objectives. The IAEA-TECDOC publications mean another effective tool to disseminate practical information and experiences. On the basis of the experience from the RRRFR programme's implementation, the IAEA issued four booklets to support the programme. They are:

- B. Yuldashev and J. Thomas: Technical and Administrative Preparation for Shipment of Russian-origin Research Reactor Spent Fuel to Russian Federation. IAEA Guideline document. Vienna, Austria. February 2007. This guideline document provides key information for the planning and return of Russian-origin SNF or materials containing HEU to the RF.
- IAEA-TECDOC-1593: Return of Research Reactor Spent Fuel to the Country of Origin: Requirements for Technical and Administrative Preparations and National Experiences. July 2008. This IAEA-TECDOC is a proceedings of technical meeting held in Vienna, August 2006 summarising shipment experiences 32 shipment preparation and operation experiences made under the umbrella of USA Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) acceptance programme and RRRFR programme.
- IAEA-TECDOC-1632: Experience of Shipping Russian-origin Research Reactor Spent Fuel to the Russian Federation. November 2009. This IAEA-TECDOC is an extended summary and account of the experience obtained from the completion of international projects on return SNF to the RF from RRs in Uzbekistan, Czech Republic, Latvia, Bulgaria and Hungary;
- Draft of IAEA-TECDOC: Legal Aspects of Spent Nuclear Fuel Repatriation to Russian Federation - Lessons Learned. The need for a multilateral approach to reviewing both national and international legal obligations connected with the international transport of the SNF was first raised in the context of LL workshop held in Belgrade 2006. The TECDOC focuses on the national and international legal aspects of SNF fuel to the RF from RRs located in a number of States in central and Eastern Europe.

3.2.4. Look into the Future

The Agency will continue to support RRRFR programme. This support encompasses the core conversion efforts of RRs since core conversion is mandatory for reactors to participate in the RRFR programme. A status review of the shipments foreseen to be completed under the umbrella of the RRRFR programme is shown in Table 5.

Table 5. Status review of the foreseen shipments

Location	Facility name	Type	Power	Core conversion		Due shipments
				Start	End	
Due HEU SNF shipments						
Kiev, Ukraine	WWR-M	Tank type	10MW	Jan 2011	2013	March 2012
Taschkent, Uzbekistan	WWR-SM	Pool type	10 MW	March 2008	Nov. 2009	December 2012
Rez, Czech Republic	LVR-15	Tank type	10 MW	Feb. 2010	Sept 2011	December 2012
Budapest, Hungary	BRR	Tank type	10 MW	Sept 2009	Dec. 2012	December 2014
Dalat, Vietnam	Dalat RR	Pool	500 kW	–	*2011*	December 2012
Svierk, Poland	MARIA	Pool type	30 MW	July 2012	Jan 2014	2016
Germany, Rossendorf	–	–	–	–	–	3 postponed SNF shipments (no scheduled date yet)
Due HEU fresh shipments						
Svierk, Poland	–	–	–	*MARIA fresh fuel 36.0 kg*		December 2012
Minsk, Belarus	–	–	–	-		No scheduled date yet
Tashkent, Uzbekistan	IIN-3M FOTON	Pulse-type	10 kW	aqueous solution of uranium-sulphate of 4.5 kg		End of 2013

As it can be seen, after the core conversion six RR SNF shipments are still due between 2012-2016. Three postponed shipments from Germany are pending, dates for which are not yet scheduled. Also three fresh HEU shipments are still due. The remaining fresh HEU at MARIA in Poland is scheduled for shipment by the end of 2012, but the fresh HEU from Minsk (Belarus) is still not scheduled.

4. IAEA activities in supporting solutions for RRSNF

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

In view of these concerns, the IAEA started a programme to help research reactor managers to deal with the spent fuel management problem. The activities included Technical Meetings (TM) and Coordinated Research Projects (CRP), which resulted in several publications available for all managers of research reactors and RRSNF storage facilities. The activity involved:

- Exchange of experience in storage options, procedures and practices [14],[15];
- Management of ageing materials in spent fuel storage facilities [16];
- Corrosion and other forms of material ageing leading to the degradation of mechanical and physical properties of RRSNF[17];
- Study of regional solutions for research reactors in Latin America [18];
- Exchange of experience in the two international RRSNF take-back programmes; the USA FRRSNF acceptance programme and the RRRFR programme [3].

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

Following the TMs and CRP, it was decided to organise a consultancy meeting with the purpose of (1) assembling operators and managers of research reactors and spent fuel storage facilities, to review their practices and plans for the management of RRSNF; (2) available commercial options for back-end management of RRSNF based on the experiences gained during the accomplishment of shipments under the auspices of back programmes.

4.1. Interim storage of RRSNF

In the preparation of guiding documents for interim storage of RRSNF, the IAEA developed a programme of activities to study and discuss how to safely maintain the integrity of the fuel, and to improve storage conditions until a final decision on the end-point is selected. Within this programme, a Technical Meeting was organized to discuss Good Practices for the Management and Storage of Research Reactor Spent Fuel. The meeting was held during October 2009, in Thurso, Scotland, and was attended by experts representing organizations with experience in handling and storing RRSNF. The information assimilated during the meeting was compiled in a Proceeding to be published to ensure good guidelines on interim storage of RRSNF for managers of research reactors and managers of RRSNF storage facilities [19].

The document will provide an overview on: (1) the review of standards and general criteria for spent fuel management and national programmes; (2) the wet storage practices and experience; (3) the dry storage practices and experience

The IAEA's Publication Committee has approved the manuscript for publication, thus the proceeding will be issued soon.

4.2. Available Commercial Option for Back-end Management of RRSNF

The needs of the nuclear community dictate that the majority of the research reactors continue to operate using LEU fuel in order to meet the varied mission objectives including science and research, education, isotope production etc. Consequently, inventories of LEU SNF will continue to be created during the life time of a research reactor with no obvious path to its disposal. Those countries with one or more RRs and no nuclear power programme may have to choose to either creating a national final disposition route for relatively small amounts of RR SNF, or, permanently shutting down their RRs before the termination of the SNF take-back programmes. Finding appropriate, sustainable and cost effective solutions for the management of the back end of the fuel cycle for these countries are critical to the continued use of research reactors in these countries.

A number of industrial entities in several countries offer international SNF management services on a commercial basis. These services may provide the basis for viable SNF management options for RRs, depending on their scope, technical compatibility, cost and accessibility.

In order to address this critical issue related to the long term viability of research reactors, the IAEA organized a Consultancy Meeting in Vienna from 3 to 5 May 2011, to discuss the feasibility, scope, acceptance criteria and accessibility of the available SNF management services.

The main objective of this consultancy meeting was to:

- Discuss the feasibility of and interest in a publication on Commercial Alternatives for Back-end Management of Research Reactor Spent Nuclear Fuel;
- Identify the possible management options and facilities;
- Define a service description template (SD) that would permit comparison of treatment and/or processing options. It is expected that the SDs should be consistent to ensure inter-comparability in all respects. The SD template should cover all issues (e.g.: accepted fuel type and conditions; packages and prerequisites for their handling at the RR, as well as package selection criteria; available transport modes and their applicability; licensing and regulatory matters; logistics support to be provided by the reprocessing facility; options for final disposal of the processing products; time frames and benchmark rates for cost estimations); a draft version of SD template is attached;
- Discuss the potential impact of research reactor common ventures, including coalitions and other cooperative arrangements on commercially available spent fuel management services, including cost, technical and administrative preparation and multiple shipments implementation.
- Establish the scope for a report documenting the options for the management of the back-end of the RR fuel cycle and a detailed plan and outline for the report.

At the meeting it was concluded that it is necessary to develop a document reviewing the status, issues and challenges of commercial solutions for the management of the back-end of the fuel cycle. The scope and draft Table of Contents for the document was finalized.

Conclusion

The IAEA has been involved for almost 30 years in supporting international nuclear non-proliferation efforts associated with reducing the amount of highly enriched uranium (HEU) in international commerce that includes support of SNF take back programmes. In order to cover the long term interim storage and all aspects of back-end solution of the RRSNF, the IAEA also involved some cases that played an initial role to collect experiences and best practices on interim storage of RRSNF, and available commercial options for back-end management of RRSNF. Regardless of how long the extended interim storage is drawn out; the resolution of the back-end problem will remain, and proliferation, safety and physical security concerns will continue to be a commitment of the RR

operating organization/Member State to ensure safe, secure and economic management (especially storage) of its own RRSNF.

It is expected that hundreds of RRs worldwide, both operational and obsolete, but not yet decommissioned, will continue storing RRSNF AR or AFR for a long time. Therefore, the continued safe, secure, reliable and economic handling, management and storage of RRSNF of all types, is a serious issue for almost all Member States with RRs. National strategies for long term storage of the RRSNF will need to be developed.

Thus the IAEA will continue contributing to the international non-proliferation efforts in connection with HEU minimization by supporting RR fuel return programs, and will continue its effort to cover the entire fuel back-end objective thus offering the Member States pertinent technology and best practices. The IAEA offers all mechanisms available through its Regular Agency Programme and Technical Cooperation Programme to assist the Member States in this matter.

References

[1] IAEA Proceeding of a technical meeting held in Thurso, October 19-22, 2009: Management and storage of research reactor spent nuclear fuel (the Proceeding is under preparation).

[2] P. Adelfang, I.N. Goldman, E. Bradley and D.O. Jincuk: IAEA Cooperation with RERTR and Spent Fuel Return Programmes: a Retrospective and Future Look. 30th International Meeting of RERTR, October 5-9, 2008. Washington D.C., USA.

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Return of Research Reactor Spent Fuel to the Country of Origin: Requirements for Technical and Administrative Preparations and National Experiences, Proceedings of a Technical Meeting held in Vienna 28-31 August , 2006, IAEA-TECDOC-1593, IAEA, Vienna (2008), available at: http://www-pub.iaea.org/MTCD/publications/PDF/te_1593_web.pdf

[4] ΝINTERNATIONAL ATIONAL ATOMIC ENERGY AGENCY, Experience of Shipping Russian-origin Research Reactor Spent Fuel to the Russian Federation, IAEA-TECDOC-1632, IAEA, Vienna (2009), available at: http://www-pub.iaea.org/MTCD/publications/PDF/te_1632_web.pdf

[5] PESIC, M., et al. Corrosion of aluminium-clad spent fuel in the RA research reactor storage pool (Serbia), on: INTERNATIONAL ATOMIC ENERGY AGENCY, Corrosion of Research Reactor Aluminium Clad Spent Fuel in Water, IAEA-TECDOC-1637, IAEA, Vienna (2009), available at: http://www-pub.iaea.org/MTCD/publications/PDF/te_1637_web.pdf

[6] MESSICK, C. GTRI's U.S.-origin and Gap Nuclear Removal Program. RERT-2011. Oct 23-27 2011, Santiago, Chile (PowerPoint Presentation).

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Guidelines document on preparatory work prior to return of spent fuel of U.S.-origin from foreign research reactors. Available at: http://www.rertr.anl.gov/FRRSNF/GUIDE.PDF

[8] IAEA-TECDOC-1593: Return of Research Reactor Spent Fuel to the Country of Origin: Requirements for Technical and Administrative Preparations and National Experiences. Proceedings of a technical meeting held in Vienna, August 28-31, 2006. Vienna, Austria, July 2008.

[9] TOZSER, S. et al. Ten Years of IAEA Cooperation with the Russian Research Reactor Fuel Return Programme. RRFM-IGORR 2012, European Research Reactor Conference. Prague, Czech Republic, 18-23 March 2012.

[10] BRADLEY, E. et al. Repatriation of Vinča RA reactor spent fuel. PATRAM (Packaging, Transport, Storage & Security of Radioactive Material) Journal, 2009, Vol 20, No 2, pp 55-59.

[11] TAYACK, M. et al. Development of new transportation/storage system for use by DOE Russian Research Reactor Fuel Return Program. Packing, Transport, Storage & Security of Radioactive Material. 2010 vol. 21. No. 2 pp 111-121.

[12] TOZSER, S. et al: IAEA's Technical Advisory Support of Vinca's SNF Repatriation in the Final Preparatory Stage of the Shipment. RRFM-2011, Rome, Italy, 20-24 March 2011.

[13] O. Šotić, R. Pešić: Nuclear Fuel Shipments for the Republic of Seerbia. RRFM-2011, Rome, Italy, 20-24 March 2011.

[14] INTERNATIONAL ATOMIC ENERGY AGENCY, Procedures and Techniques for the Management of Experimental Fuels from Research and Test Reactors, IAEA-TECDOC-1080, IAEA, Vienna (1999), available at: http://www-pub.iaea.org/MTCD/publications/PDF/te_1080_prn.pdf

[15] INTERNATIONAL ATOMIC ENERGY AGENCY, Experience with spent fuel storage at Research and Test Reactors, IAEA-TECDOC-786, IAEA, Vienna (1995) available at: http://www-pub.iaea.org/MTCD/publications/PDF/te_786_prn.pdf

[16] INTERNATIONAL ATOMIC ENERGY AGENCY, Understanding and Managing Ageing of Material in Spent Fuel Storage Facilities, IAEA Technical Reports Series – 443, IAEA, Vienna (2006), available at: http://www-pub.iaea.org/MTCD/publications/PDF/TRS443_web.pdf

[17] INTERNATIONAL ATOMIC ENERGY AGENCY, Corrosion of Research Reactor Aluminium Clad Spent Fuel in Water, IAEA-TECDOC-1637, IAEA, Vienna (2009), available at: http://www-pub.iaea.org/MTCD/publications/PDF/te_1637_web.pdf

[18] INTERNATIONAL ATOMIC ENERGY AGENCY, Spent Fuel Management Options for Research Reactors in Latin America, IAEA-TECDOC-1508, IAEA, Vienna, (2006), available at: http://www-pub.iaea.org/MTCD/publications/PDF/te_1508_web.pdf

[19] IAEA Proceeding: Management and storage of research reactor spent nuclear fuel. Proceeding of a technical meeting held in Thurso, October 19-22, 2009 (draft)

RETHINKING USED FUEL MANAGEMENT

A Post-Fukushima perspective

M.Chiguer, A. Froment and F. Lelièvre
AREVA
1, Place Jean Millier, 92084 Paris-La-Défense – France

ABSTRACT

This paper lists and briefly describes challenges and issues at stake that used Fuel Management will be facing at existing Spent Fuel Pool (SFP) at reactor in operation or permanently shutdown in the light of the accident of Fukushima Daiichi.
Proven solutions going along risk reduction strategies that are of value to adequate protection and appropriately balances defence-in-depth and risk considerations as far as Used Fuel Management is considered are presented from the unloading of used fuel at SFP, up to the disposal of final waste.
Based on this and to come back to a more orthodox Used Fuel Management, utilities as well as stakeholders should rethink Used Fuel Management Strategy and Options going along with starting at its very first stage, namely the storage configuration and requirements at SFP. This is one of the pre-requisite needed today to help provide more confidence for existing NPPs and for new builds to proceed.

1. Introduction

The Fukushima accident draws attention to a broader nuclear industry issue concerning used fuel management options and strategies, namely that used fuel was, and is still perceived as one of the crucial unresolved issues when referring to nuclear energy.
After being something of an afterthought in many national fuel cycle policies, Used Fuel Management Options are likely to be reconsidered in the aftermath of Fukushima Accident.
Whatever the changes that will be required by Safety Regulators across nuclear countries as well as new comers, oversight of Nuclear Power Plant safety performance will be strengthened by focusing more attention on defence-in-depth philosophy including deactivation or Spent Fuel Pool (SFP) at reactor [Ref. 1, 2 and 3], the very starting point of Used Fuel Management at Nuclear Power Plant.

2. Background and issues at stake…

….Before Fukushima and during decades, used fuel was perceived as one of the crucial unresolved issues when referring to nuclear energy. This conclusion, largely shared, was based on the results of several opinion surveys launched by governments, national and international institutions [Ref. 4, 5 and 6] and the nuclear industry across all nuclear countries.

Moreover, it was shown that Public Opinion main cause for reluctance to nuclear energy production during the Nuclear Renaissance is the large accumulation of waste and used nuclear fuel (Fig. 1) which remains, and by far, a very sensitive subject that crystallizes many fears and a worrisome issue in every country, regardless of their position on nuclear energy.
Up to Fukushima events, the above mentioned concerns have been nurtured and guided solely by the visible part of the used fuel inventories, while the hidden part of the iceberg, the majority of the inventories (more than 80%, Fig.1), were stored in SFPs pending decision on comprehensive used fuel management national policy.

Advanced opinion survey among “well informed” group of individuals familiar with the subject and “expert” group made up of local and national government representatives as well as

journalists across the US and six nuclear countries in Europe (Ref. 6) showed in 2010 that they are in favor of current and proven recycling technology that brings down used fuel inventories or stockpile figures instead of waiting for a “leap-frog” technology, for example Gen IV reactor and advanced recycling technology going along with, that might ultimately resolve all burden in the future!

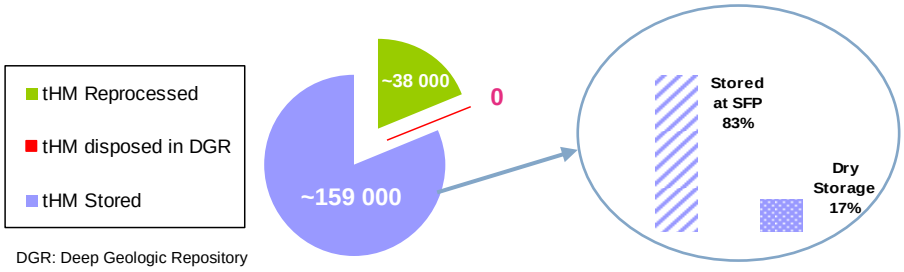


Figure 1. Worldwide Inventories of LWR’s discharged used fuel at the end of 2011, in tons of Heavy Metal (tHM)

Paradoxically and despite this clear feedback from public opinion and opinion leaders, a cornerstone in the development of nuclear energy, Used Fuel Management was something of an afterthought in many national fuel cycle policies, leading to a large spread of “wait & see” strategy across nuclear countries and “new comers”, the so called nations developing or aspiring to have civilian nuclear energy programs.

The interim solution to SFP increased inventories has then been a modification of spent fuel storage racks to further increase the ultimate discharge capacities of deactivation or reactor fuel pools at most reactor. This interim solution has since turned into a long term solution in most cases, if not a permanently solution in the case of shutdown reactors. This resulting in increasingly large inventories of used fuel and radionuclide being stored in reactor SFPs.

… After Fukushima, the natural disaster of March 2011, in Japan, followed by the accident at the Fukushima Daiichi NPP, it has become clear that SFPs were an aggravating factor in the difficult emergency situation faced by the operator and SFPs turned out to be far more vulnerable than initially assumed and have led to concerns of radioactive release [Ref.7 & 8]. Although the radiological consequences from Fukushima due to airborne releases have eventually far been predominately released from reactor cores, SFPs at Fukushima definitely presented a considerable potential threat given that there was no containment to prevent potential releases coupled with the large radionuclide inventories in SFPs [Ref.1 & 8].

Actually, while cooling down reactor core has been the first priority, the Operator quickly understood that keeping SFPs filled with water and adequately cooling used fuel were the other priorities at Fukushima following the earthquake and tsunami. Priorities needed to help:

- Maintain hall temperature below 80°C, threshold for human access in SFP hall needed for survey and control
- Maintain shielding to avoid release and dose intakes
- Protect the fuel structural integrity to avoid releases and re-criticality
- Prevent hydrogen formation due to the oxidation of the Zirconium cladding exposed to air, resulting in hydrogen generation, important only at elevated temperatures (>650°C), and possible risk of explosion. Though, clad structural failure could occur at temperatures as low as 650°C should thermal loading sustained fro several hours [Ref.13]
- Avoid fuel fires due to zirconium fire: the cladding ignition is about 850°C compared to the fuel melting point over 2800°C.

Five months later, the operator was still facing a difficult situation to restore normal condition in the SFPs, notably in unit 4, despite the 10-month cooling time of the latest discharged

used fuel batch. This explains the shock and surprise among the public, and broadly among stakeholders, fed further by the apparent inadequacy of contingency plan and preparedness at the plant during the Fukushima accident relayed heavily by the new information and communication media. Even to date, and after prompt reinforcement to existing SFP structures, Fukushima operator knows that he has not turned the corner yet and removing all used fuel assemblies off SFP is still the pre-requisite to enhance safety of Fukushima shutdown accident-damaged NPP, notably to face risk of new earthquake [Ref 3 & 9] when structures and equipments have already been heavily impacted during March, 2011 accident.

This served as a grim reminder that used fuel at most reactor site might become today one of the biggest risk at some plant as concluded by the Japanese Atomic Energy Commission (JAEC) and presented by its president to Japanese Prime on March 25th, 2011 [Ref.8]. This confirms that earlier conclusions stating that *"the risks associated with spent fuel storage [at reactor pool] were extremely small in comparison with accidents associated with the reactor core*" is today no more intrinsically valid and could be significantly attributed to used fuel management at SFP along with other earlier plant related design parameters, instrumentation and layout.

Consequently all pre-Fukushima used fuel management related critics and weakness underlined by stakeholders are rushing back and regaining stakeholders' confidence, and in particular public opinion, will be a long road and the Achilles heel of the civilian nuclear development.

3. Originally, a shared view, "no used fuel accumulation in deactivation pool"

Yet, as soon as the civil nuclear power age got underway, it became unthinkable to imagine generating nuclear electricity without reprocessing used nuclear fuel. In every country where this form of energy was being developed, construction programs involved not only Nuclear Power Plants, but also fuel cycle facilities, notably reprocessing spent fuel facilities [Ref.10]. This view has been extensively shared across European countries and overseas in the Fuel Cycle facilities development and implementation early 70s and 80s. We have been since witnessing the genuine join development of expertise and facility not only in the front-end of the fuel cycle (Enrichment), but also in the back-end of the fuel cycle with "Eurochemic" and "United Reprocessors Ltd" or other multi-lateral agreements [Ref. 10].

Unfortunately, the postponement of Back-end strategy implementation or the spread of the Wait & See strategy have meanwhile both led to the unavailability of off-site routes for used fuel discharged in the deactivation pool (SFP).

Consequently, more and more NPP has been challenged by limited storage space in SFPs at reactor, absolutely necessary for NPP operations (planned outages for loading/unloading batch operation, or for preventive maintenance and curative maintenance with a mandatory space for a Full Core Reserve, FCR). In order to accommodate this unanticipated upcoming larger used fuel inventories and to help Nuclear Power Plants across the world to continue their operations, high density racking in pools has been designed and ultimately licensed allowing maximizing SFP storage capacity.

This led storage rack design in SFPs to evolve from open and widely spaced structures, and consequently neutron flux "trap" design, to a closed and tightly packed frame steel containers relying heavily on permanently installed neutron absorbers to maintain criticality requirements (Fig.2).

This new rack design has been further spread and widely shared among NPPs with its corollaries, a fundamental design change in rack design and configuration as well as used fuel management option and storage requirements in SFP going along with. Both impacted

the existing conservative margins in SFP safety analysis and, ultimately, accident risk analysis, in particular by decreasing drastically available grace period in case of loss of water inventory or cooling.

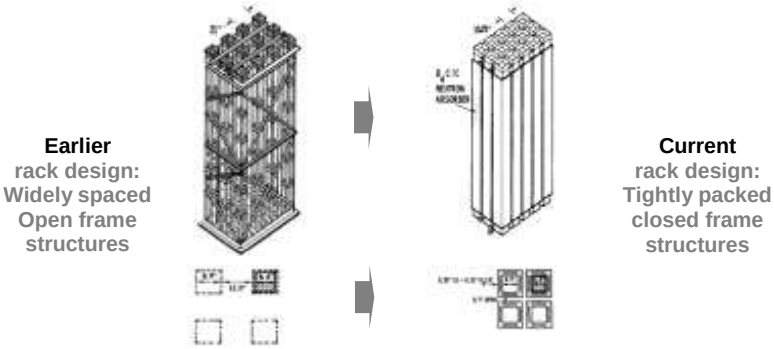


Figure 2 – Higher density racking to accommodate the larger used fuel inventory at SFP [11], PWR type case

Accident risk analysis associated with SFP was impacted further with refuelling outage duration reduction, reactor lifespan extension and increase in discharged fuel burnup.
In addition, fuel assemblies (FAs) themselves have become more reactive with increased ^{235}U enrichment an obvious example along with other more subtle changes: increased fuel pellet diameter and density, increased rod number per FA, increased use of fixed and integral burnable absorbers, and even changes to core operating parameters due to power uprates result in more reactive FAs to be stored in the SFP [Ref. 13].
Confluence of the various factors above mentioned coupled with the first lessons learned from the Fukushima Daiichi nuclear accident as well as from post-accident management will cause SFP safety margin assessments, accident risk analysis and storage requirements to definitely become far more complex with regard of today's and complete changes will take time before being defined and implemented.

4. Safe and Optimized Used Fuel Management

Whatever the changes that will be required by Safety Regulators, oversight of safety performance will be strengthened by focusing more attention on defence-in-depth philosophy including SFPs, the very starting point of used fuel Management. The effectiveness of the response of existing as well as upcoming SFPs, as part of the new build, and the effectiveness of the preventive safety strategy are related to the enhancement of safety margins and prevention of environmental damage, notably in extreme situations. In the following a three-pillar approach for a safe and optimized used fuel management is described.

4.1 The starting point, SFP hardening, with its corollary, the instrumentation upgrades, namely providing SFP with safety related instrumentations able to withstand design-basis natural phenomena to remotely monitor basics, though key parameters, such as area radiation, temperature and water levels. In most cases, those upgrades have been determined that should be started without unnecessary delay.
Hardening the pools, strictly speaking, consists mainly, in reinforcement of the robustness of cooling functions notably against some potential effect of severe natural events (i.e. loss of ultimate heat sink, loss of power supply and prolonged SBO conditions) and, if needed, reinforcement of structural integrity of the SFP to withstand a new safety requirements.
These upgrades are mainly consequences of stress-tests, complementary safety assessments or prioritized recommendations and are plant related.

4.2 The second and obvious measure, Used Fuel Inventory Reduction and Segregation, is under assessment in many nuclear countries following stress tests and prior to potential prioritization along with other measures of risk reduction [Ref. 23 & 24].
Should the decision be taken, this will definitely lower the inventory, and therefore the residual heat in the pool, enlarging the time to reach the boiling point and the time to boil down to the top of the fuel and consequently enlarging the grace period. As Fukushima lessons learned clearly showed is that a simple action as SFP water refilling could be very difficult in extreme conditions and in that situation, comfortable grace period is essential to succeed.

Additionally, reducing used fuel at SFP introduces a clear segregation between various used fuel inventories at NPP and issues going along with, notably during refuelling outages. Both reduction and segregation enhance key parameters that govern likelihood and consequences of various SFP or plant accidents radionuclide inventories releases (see section 2). And again this is in line with the closing remarks of the1st International Experts' Meeting on "reactor and spent fuel safety in the light of the accident of Fukushima Daiichi" organized by the IAEA on March 2012 [Ref.16 & 25], where emphasis have been put by nuclear experts from more than 40 country members and 5 International Organizations on efforts to place priority not only on preventing accidents, but also mitigating them.

With four decades of excellent track records in Reprocessing and Recycling used nuclear fuel transferred from SFPs, utilities as well as stakeholders should take benefits of this proven solution that helps reduce drastically used fuel inventory in SFP [Ref. 17 & 19], including those recently discharged as early as one year after its off-load from reactor core [Fig.3] when re-assessing and re-thinking their used fuel management strategy. This duration figure could be adapted depending on core management, fuel type and performance and capacity of logistics means.

In addition to reducing accident prevention and mitigation, recycling used fuel currently stored in SFP and using current proven technology also alleviates the need for fresh natural uranium to be mined thanks to the recovery of 96% of valuable materials still encased in used fuel, namely uranium and plutonium and transforming them into ERU fuel (Enriched Reprocessed Uranium fuel) and MOX fuel (Mixed Oxide fuel) respectively. The remaining 4% is actual high level waste (HLW) which contains practically no remaining fissile material, consequently IAEA safeguards constraints free, and no energy value for the current and mid-term generation of reactors and fuel cycle facilities.

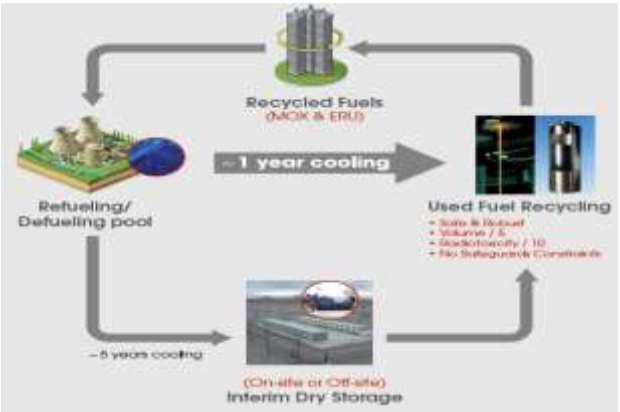


Figure 3 – Safe and Optimized Used Fuel Management Strategy
Recycling or interim storage flow diagram [Ref. 15]

Thus, recycling makes a positive contribution to the security of supply and market predictability [Ref.17] by saving up to 25 % of natural uranium reactor requirements that are subject to volatility of uranium market.

In conjunction with final disposal and for a long time now, recycling of used nuclear fuel has been used also to reduce drastically disposed waste volume (by a factor of 10) and radiotoxicity (by a factor of 5) as well as heat residual heat, a cornerstone in long term safe storage as well as of design and construction of the Deep Geological Repository.
Finally, and should recycling implementation not foreseen in the near term, the existing and proven dry storage solutions present an interim off-site route (Fig.3) for older used fuel, for example with 5 years of cooling time, that will help utilities to early transferring used fuel from reactor pools to dry storage [Ref. 21, 22 and 23] and Nuclear Waste National Agencies to choose an interim outcome to the accumulation of the used fuels stored or discharged currently in reactor pools awaiting for more broader strategic decisions to be taken [Ref 20] and, if so, to be timely and genuinely implemented. This latest time figure, namely 5 years of cooling time prior to dry storage is an average figure that depends on several parameters such as initial enrichment, burn-up, cask design, safety cases and licensing limits or economics considerations

4.3 The ultimate measure, SFP's racking configurations enhancement

After removing older used fuel currently stored at SFP as described above, utilities will be in position to enhance further the intrinsic safety of used fuel management at SFP by reviewing Fuel Assembly holder/basket configurations and promoting further the design of storage racks that enhance heat transfer for a drained SFP and large fuel spacing for criticality control giving more robustness to face beyond design basis accident and additional grace period.

5. Conclusion

To come back to a more orthodox Used Fuel Management, utilities, Safety regulators and stakeholders should rethink Used Fuel Management Strategy and Options going along with, starting at its very first stage, namely the storage configuration and requirements at SFP, including maximum cooling time of used fuel after its discharge SFP, through off-site routes up to disposal of final high level waste. This is one of the pre-requisite needed today to help provide more confidence for existing NPPs and for new builds to proceed.

By removing discharged used fuel and reducing their inventory, and consequently radionuclide inventory, in reactor pools at the earliest, recycling will help countries and utilities that have previously postponed used fuel management strategy to get back rapidly to a safer and optimized used fuel management.

In choosing to recycle used fuel and consequently reduce radionuclide inventories in reactor pools since its inception as well as managing ultimate waste responsibly and optimally, civil nuclear power in Europe has prepared its renaissance and proven its maturity, notably to meet the challenges of the 21st century.

REFERENCES

[1] US NRC, Recommendations for enhancing reactor safety in the 21st century July, 2011
[2] WENRA Task Force "Stress tests specifications" April, 2011
[3] Report of Japanese Government to the IAEA, June 2011 at www.iaea.org
[4] EU country members' survey, Eurombarometer 2005 & 2008 at http://ec.europa.eu/public_opinion
[5] US DOE/NEI nationwide Opinion Survey conducted by Bisconti Research Inc at www.nei.org
[6] Nuclear Energy and Recycling opportunities: the perceptions in Europe & the US at www.tns-sofres.com
[7] UK-ONR Report on implications of Fukushima for the UK nuclear industry Interim report, p.27/106-May 2011
[8] JAEC, "Contingency scenario outline of Fukushima Daiichi NPP" JAEC's report to Prime Minister of Japan on March 25th 2011.
[9] TEPCO, "General Plan of the cover for fuel removal of Unit 4 in Fukushima Daiichi NPP" Attachment to Tepco release on April 16, 2012 at www.tepco.co.jp
[10] WNA annual symposium "recycling, back to the future"; M. Chiguer and al Sep. 2008
[11] US NRC-NUREG/CR-0649 "Spent Fuel Heatup Following Loss of Water during Storage"
[12] US-NUREG WASH- 1400 Reactor Safety Study- An assessment of accident in US Commercial Nuclear Power Plants, October 1975
[13] US NUREG - Severe accidents in SFP in support of Generic Safety issue 2- August 1987
[14] US NRR action Plan "On Site Spent Fuel Criticality Analyses, TAC- n° ME0372, May 2010
[15] IAEA International experts' meeting on reactor and spent fuel safety in the light of the accident at the Fukushima Daiichi Accident, M. Chiguer March 2012
[16] IAEA International experts' meeting on reactor and spent fuel safety in the light of the accident at the Fukushima Daiichi Accident, Chairman's summary R. A. Meserve, March 2012
[17] Recycling as an option of used fuel Management strategy, Nuclear Engineering & Design V.241 p.1238-1242, T. Zagar, M. Chiguer and al. 2011
[18] UK-ONR, Japanese earthquake and tsunami: Implications for the UK nuclear industry, Final report, pp. 60/288 Sep. 2011
[19] ENC 2010, Recycling as an option of used fuel management strategy for Europe, M. Chiguer and al. June 2010
[20] US-NRC Prioritization of Recommended actions to be taken in response to Fukushima Lessons Learned, Policy Issue, Oct. 2011
[21] "Tennessee Valley Authority, TVA, considering moving additional spent fuel to dry storage" as reported by Spent Fuel Vol.19 n° 908 May 2012
[22] US NRC Fukushima Lessons Learned Actions » presentation by W. Borchardt at the International Experts' Meeting organized by the IAEA on March 19-22, 2012
[23] US NRC on "NTTF identified the early transfer of spent fuel from pools to dry storage as an aea to examine » as reported by Spent Fuel Vol.19 n° 909, May 2012
[24] CSN, the Spanish Nuclear Safety Regulator, Acto del pleno del CSN n°1-208, Madrid October 13, 2011. In addition to Stress tests that could be viewed at www.ensreg.eu
[25] IAEA International Experts' Meeting- Fukushima Daiichi Accident Spent Fuel Pools – Facts, actions and issues, H. Abe and all, Japan Nuclear Energy Safety Organization, March 2012

REVIEW OF QUALIFICATIONS FOR FUEL ASSEMBLY FABRICATION

DAN SLABU, MARTIN ZEMEK, CHRISTIAN HELLWIG
Axpo AG
5401 Baden – Switzerland

ABSTRACT

The required quality of nuclear fuel in industrial production can only be assured by applying processes in fabrication and inspection, which are well mastered and have been proven by an appropriate qualification. The present contribution shows the understanding and experiences of Axpo with respect to qualifications in the frame of nuclear fuel manufacturing and reflects some related expectations of the operator.

1. Introduction

Fuel Assemblies (FA) fabrication monitoring and inspection is a major part of Axpo's activities in the procurement of nuclear fuel for the power plants Leibstadt (BWR), Beznau I and II (both PWR) [1] and is performed according to the requirements of regulatory guidelines [2]. The review of qualification activities performed by fuel suppliers is an important aspect of the Axpo monitoring activities already mentioned in [1].
The present contribution shall focus on our efforts regarding monitoring and review of qualifications and give a broader insight into this topic.
During the last 32 years Axpo has received FA with normal UO_2 fuel, enriched reprocessed uranium (ERU), and mixed-oxide (MOX) from more than a dozen companies and fabrication sites spread over three continents. This broad spectrum of contractors and fuel types (including various Lead Use Assembly programs) forced Axpo to develop, apply and steadily improve a procedure to deal with new fabrication procedures and/or new types of FA, which have to be subject to qualification.
Qualification was first mentioned by Platon, in Greek language: "poiotes" meant the basic appearance of something. He was the first to create and give a sense to this unusual word at that time [3]. Cicero translated it to Latin in two parts: "poios" to "qualis" and "-tes" to "-tas". Thus, the term "qualitas" was born; today known as "quality" [4].
In the middle ages "qualis" was combined with "facere" (to make) [5], resulting in "qualificare", and the term of "qualification" was finally born. During the years, the term "qualification" became more and more important.
According to literature the meaning of qualification is the capacity, knowledge, or skill that matches or suits an occasion, or makes someone eligible for a duty, office, position, privilege, or status. Qualification denotes fitness for purpose through fulfillment of necessary conditions such as attainment of a certain age, taking of an oath, completion of required schooling or training, or acquisition of a degree or diploma [6].
In the technical meaning "quality" corresponds to the ability of a product or equipment to fulfill the specified criteria or to provide the required functionality. The today's technical understanding of "quality" is based on standards first defined for the production of military goods [7]. They were then further developed, modified and are applied today in almost all main industry branches like: car- and airplane-industry, medicine and pharmacy. These industries have established a specific qualification culture for their industry during the years.
In the nuclear industry the situation is somewhat different. As international norms are largely missing and guidelines mostly stay vague, different companies developed their own qualification culture and different ways to perform a qualification are applied [8].

2. Qualifications Applied in the Nuclear Fuel Manufacturing

In a mass production, as it is the case for fabrication of nuclear fuel or at least for a part of its components, the standard quality control (QC) practice is the application of sampling inspection. In other words, a complete 100% inspection of the product is not always possible. Although most of the sampling inspections are based on well-established procedures, there are still processes relying only on mere inspection sampling. Experience shows, that this practice can be justified in terms of quality and economics as well. However, in this practice the quality of FA and/or its components cannot be guaranteed exclusively by a certificate based on the said sampling inspection as outliers cannot be always fully excluded. The required quality can only be assured by applying processes in fabrication and inspection, which are well mastered and have been proven by an appropriate qualification. For this reason it is imperative that the qualification programs have been audited at least once and that any changes in processing, inspection and personnel are monitored, to assure that the original qualification still applies or alternately, to decide that a revised qualification may be needed [9].

A good descriptions about the qualification practice and methods within the nuclear fuel fabrication can be found in e.g. IAEA guides [10], or in documents of public or private institutions [9,11]. The present paper shows the understanding of Axpo with respect to qualifications in the frame of nuclear fuel manufacturing and reflects some related expectations of the operator.

Several types of qualifications are applied in the FA manufacturing industry: plant/site qualification, installation qualification, process/operation qualification, product qualification, personal/human qualification.

However, the main target of each type of qualification is to assure and witness the capability to operate equipment or to produce a product with a stable quality level, which is well in the specified range at a high confidence level.

Plant Qualification: In case that a completely new production site is being taken into operation (maybe as a completely new manufacturer), a plant qualification is necessary. This qualification will consist of several steps including all the below listed qualifications.

Installation (Line) Qualification: This is a demonstration that the manufacturing or inspection equipment including all necessary tools and apparatus has been validated and is capable to generate/inspect products according to specifications and drawings.

Process/Operation Qualification: This is a demonstration that the process has been developed and tested consistently such that the results of the process or operation are fully corresponding to the requirements and that destructive testing is not necessary.

Product Qualification: This is an action of verifying that a certain product has passed performance or quality tests as required by the specifications or drawings.

Personal Qualification: This is the demonstration that the personal involved in the manufacturing/inspection has been adequately educated/verified and is capable to conduct the operations according to the requirements.

Qualification of suppliers: Suppliers for FA components must abide by the same qualification processes as noted above to provide products that consistently meet specifications and performance requirements. Suppliers of critical materials or process components have to be qualified to provide products that are compatible with the manufacturer's qualified process.

3. Steps of Qualification Development

While the specific nomenclature of the qualification and the specific content of individual documents vary from manufacturer to manufacturer, the total sum of the information is roughly equivalent.

A qualification consists generally of three parts [9,11]:

i Definition of Qualification Rules and Criteria

First, rules for the proposed qualification have to be fixed for each fabrication line, installation or process to be qualified. These rules are stated in the first quality assured document, generally referred to as **qualification program**.

The qualification program describes the subject and scope of the qualification, testing, sampling, inspection methods, and the acceptance criteria that have to be met, such that the requirements given in the respective process and product specifications are fulfilled. It may also define the actions for non-conformances and conditions under which re-qualification is required [11].

ii Performance of the Demonstration Runs and/or Tests
Depending on the qualification method, the demonstration fabrication runs, inspection measurements and/or tests are performed, i.e. the **qualification process** is conducted.

iii Issue a Quality-Assured Document Confirming the Qualification
Finally, a new document - e.g. a **qualification report** has to be issued.
The qualification report is a part of the quality assurance (QA) system and contains:
- the approval of the process,
- the list and description of the samples tested,
- the list of the fixed and checked key process parameters,
- the evaluation of inspections and measurements and
- the approved and fixed criteria

Parameter sheets, operation procedures, and instruction procedures are generated on the base of the qualification reports and are consequently used in the fabrication. An important part of this report is a statement about the conditions of validity of the qualification, which are clearly defined on the technological and/or time basis.
In both, qualification program and qualification report, attention has to be paid particularly to the equipment / line performance and process outcome. Inadequately high rejection rates due to quality deviations of the produced components or due to malfunctions of the equipment shall not be accepted for a qualified process (e.g. [12]).

4. Qualification Methods

Two methods for qualifications can be given as examples:
- the experience-based qualification using the historical data for qualification purposes,
- the demonstration run qualification, where a limited volume of parts is manufactured under intensive surveillance and tightened criteria [9,13].

4.1. Experience-based qualification

The experience-based qualification can be applied for plant or fabrication qualification in case that: i) it is already in operation and ii) the producer has already a long-year experience with nuclear fuel or fuel component fabrication for commercial LWR and can provide sufficient references.
Two types of experience-based qualification can be mentioned here:

i Requalification: A requalification has to be performed according to:
- time validity requirements
- significant time interruption of manufacturing
- change of hardware/maintenance
- change of the process requirements/yields

ii Extended Qualification: A qualification has to be extended in the case of:
- diversification of the product type
- change of the specified requirements

The qualification can be based on historical manufacturing data . In the qualification program, the description of: the fabrication line, relevant process parameters and their ranges, the specifications and the scope (time or volume based) of data for evaluation have to be defined. The corresponding information on the stability of the relevant process parameters and qualitative characteristics of the product are statistically treated by appropriate tools, are compared to the specifications and are therefore providing the required proof of the quality capability and long-term stability [13].

In the final report, the qualified equipment and parameters shall be described and the results of the said statistical evaluation shall be given.
After confirmation that the plant and/or line fulfills all criteria, a production campaign of demonstration FA is typically performed under extended final inspection of components and final product. The qualification might be finalized by the consequent irradiation test program with demonstration FA [13].

4.2. Qualification Based in Demonstration Runs

In most cases, the qualifications deal with new lines, equipment, processes or products without any previous large-scale experience. In this case, the purpose of a qualification program is to define the bounding parameters for a process that, if observed, will deliver a product that meets the specifications within a certain statistical reliability (see below). A fabrication or inspection process can be qualified by a short, statistically designed production campaign, within and extending beyond the proposed process limits. The product is then inspected thoroughly to determine whether it meets the specified product quality [9].
In the qualification program a description of the process to be qualified is given and the relevant process parameters are defined. It then outlines the test matrix to be used to qualify the particular product. It specifies the range of process parameters to be qualified, the number of tests to be carried out, and the acceptance criteria for the tests and the product.
Upon completion of the qualification tests as defined in the qualification program, a QA document summarizing the qualification is prepared. This document includes a description of the equipment and/or process being qualified, the date of testing, the particular qualification program defining the testing, the product being qualified, the tests and methods used for qualification, the results of the tests, acceptance criteria for the tests and product, the qualified parameters and possibly their acceptable ranges, non-conformances (if any) and their resolutions, and the specific steps to be taken to implement the qualified parameters into production [11]. A good example of a line qualification is given in Fig. 1 for the fuel pellet as a key sub-component of the FA.

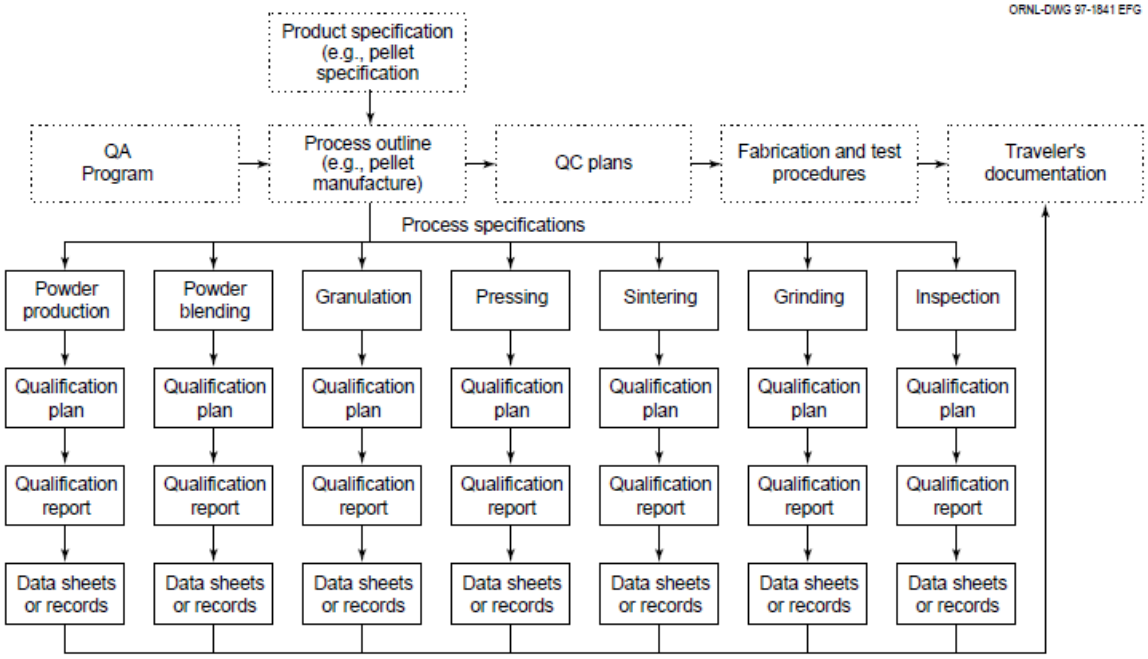


Fig. 1 Example of a line qualification process: fabrication, qualification, and QA documentation for pellets [11]

5. Qualification tools

5.1. Fuel Fabrication Processes and Products

As already discussed, the fabrication of nuclear fuel can be regarded as a mass production. This is the reason, why the statistical methods play a major role not only for the product QC during the standard fabrication, but also for the qualifications.
Depending on the specific characteristic, the QC in the nuclear fuel production applies a singular or statistical sampling inspection or a 100% inspection. For discontinuous processes, the sampling can occur on a volume basis or, for continuous processes, on a time basis [10]. Correspondingly, statistical sampling and evaluation plans are applied. However, for demonstration purposes in a qualification, other criteria have to be used.
A solid basis for the statistical treatment of process qualifications is provided e.g. in [14,15]. A graphical example of value distributions in quality capable and incapable processes is presented in Fig. 2. Control and long-term stability is required for a valid qualification, see Fig. 2.
The method for a process qualification can be the additional application of the process capability indices c_P and c_{PK} on the 6σ basis [13,16] for the statistical evaluation of samples. The process capability index c_P does provide information about the process scattering compared to the specification tolerance [17]:

$$c_P = \frac{min(USL - T; T - LSL)}{6 \cdot \sigma}$$

where *USL* and *LSL* are the upper and lower specification limits, *T* is the target value and *σ* is the standard deviation of the sample. However, this index does not answer the important question whether the process is "centered", i.e. if the average value of the process variation is near to the target value. To answer this question, a second capability index c_{PK} was developed [17]:

$$c_P = \frac{min(USL - T; T - LSL) - |\mu - T|}{3 \cdot \sigma}$$

where *μ* is the average value of the sample. The graphical representation of the c_P and c_{PK} indices is given in Fig. 3. For demonstration of a high process stability and control, the required limit values of c_P and c_{PK} should be properly defined. Typical minimum values can be between 1 and 2 (Fig. 3) [13,16,18]. However, this model can only be used for processes where scattering is governed by random factors, i.e. for processes with products of normally or nearly-normally distributed characteristics. For this reason and because the typical sampling plans applied in the QC rely on normally distributed samples, tests for the normality of the value distribution, e.g. as according to [19], should be included into the qualification requirements as well.

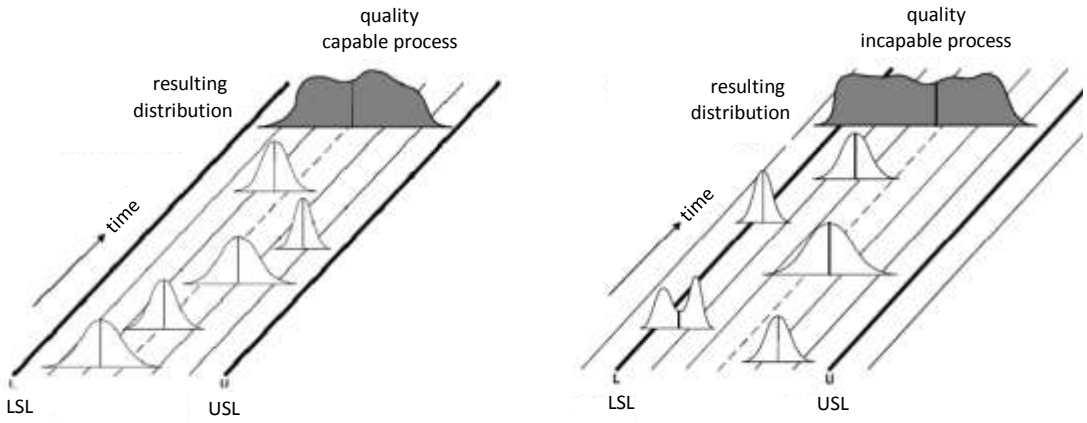


Fig. 2 Scattering of sample characteristics in capable and incapable fabrication processes (according to [14]).

The described process capability indices were developed to describe the process stability. For a product qualification on an already qualified fabrication line, different criteria would be applied. Different general statistical sampling plans were developed and are in use. For

example in [13,20] the Acceptance Quality Limit (AQL) as per [21] can be applied for the normal production of a product component. During a product qualification, it shall be demonstrated that the new or modified product can be produced. Consequently more intensive sampling rates and tightened acceptance criteria can be applied for demonstration purposes on a limited volume of the product. In the example shown in Fig. 3, the sampling volume according to the tightened inspection at Level III as per [21] can be used. Also the application of statistical tolerance (TL) or confidence (CL) limits according to [22,23] can be applied in advance [13,24,25]. For the majority of the statistical tools, the normal distribution of the values is a precondition of their applicability, therefore the normality tests shall be required in this case as well.

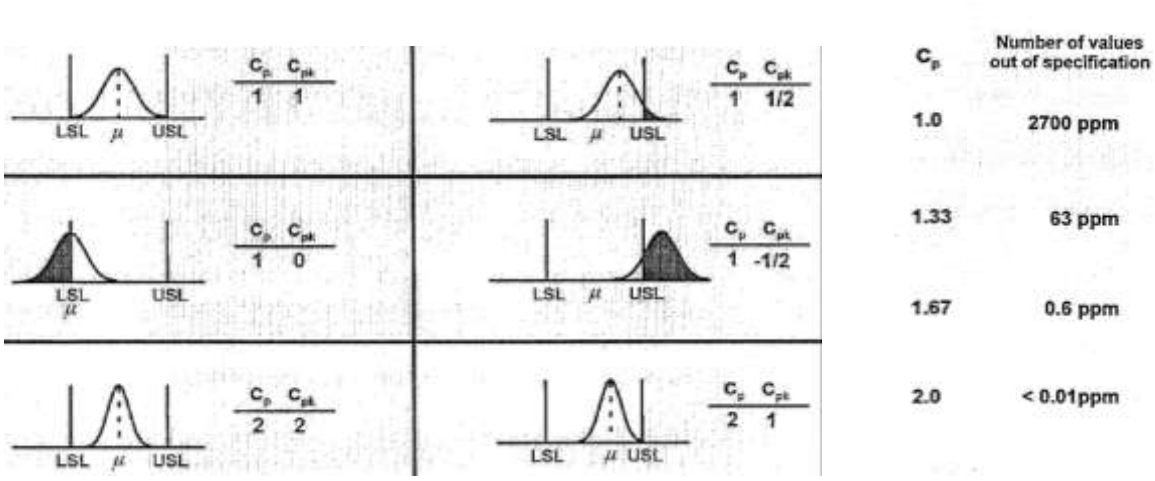


Fig. 3 Graphical representation of the c_P and c_{PK} indices [16].

5.2. Fuel Inspection Methods

Adequate and properly qualified inspection methods form the solid foundation of the QC [13]. Most of the methods applied in the nuclear industry have been standardized by the American Society for Testing Materials (ASTM). The qualification is also described e.g. in [26]. In fuel manufacturing a large variety of different inspection methods are in use. The selection of the qualification tool depends on the nature of inspection method and on the quality system itself. Several statistical tools for the verification and qualification of inspection methods exist [27]. Generally, for a validation of a measurement system, the detection limit and the measurement uncertainty have to be known (Fig. 4, see also [28,29]. This information can be obtained from the reproducibility and repeatability (R&R) analyzes [9]. In practice, also Analysis of Variances (ANOVA) tools are applied [18]. Interlab comparative analyses or round-robin test series can also be used for validation and qualification of inspection methods or equipment. As a result, the compatibility of the inspection systems with the requirements of the specifications and/or drawings shall be demonstrated.

Similar to the qualifications in the manufacturing, a qualification program for the inspection method or equipment to be qualified has to be prepared, including the description of the method or equipment, the measurement conditions and parameters, and applied criteria for qualification. Usually, a series of test measurements is performed on certified standards and/or real products to demonstrate the required stability and precision. The certified standards are therefore a substantial part of the qualification and of the inspection method later in normal production. The results are summarized again in a QA document evidencing the results, stating detection limits and measurement uncertainty of the method and defining the inspection conditions.

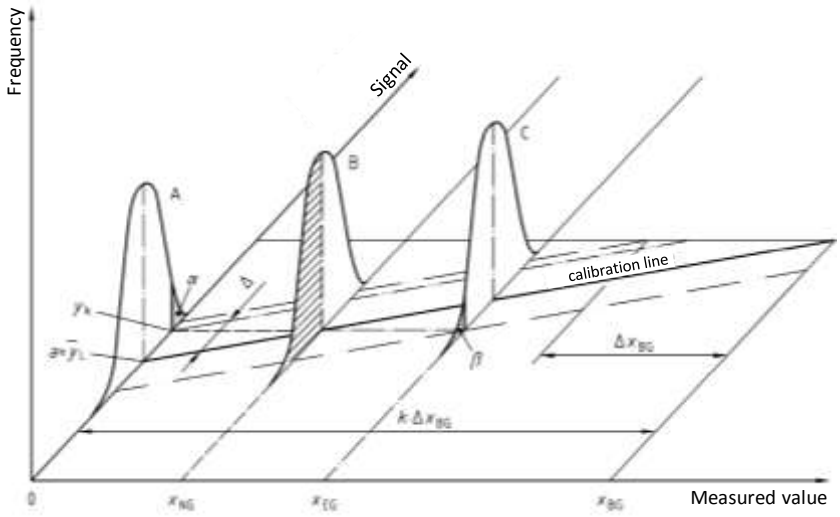


Fig. 4 Example of the detection limits (x_{EG}) quantification during qualification of an analytical measurement method according to [29].

Finally it has to be pointed out that - although rather helpful - none of the qualification methods or statistical tools can replace the deep understanding of the processes in the fabrication and inspection of nuclear fuel. Continuous know-how maintenance is crucial to build-up and assure a high quality level in the production. Detailed knowledge is needed not only for fabrication but also for the preparation of adequate qualification programs and thus for a thorough and representative demonstration of quality capability.

6. Experiences of Axpo with Qualifications in the Nuclear Fuel Fabrication

Axpo has participated in all types of qualifications and validations mentioned above during the last decades.

As example for Plant Qualification, the two fuel plants "MOX Demonstration Facilities" (MDF) and SMP Sellafield were qualified for MOX FA delivery to Beznau. As a result it was demonstrated that the sites were capable to produce FA for Axpo according to the specified requirements. In the case of these two plant qualifications all main manufacturing and inspection installations have been qualified for MOX FA delivery to Beznau, i.e. an installation (line) qualification was performed. Furthermore the process/operation, product, and supplier qualifications have been reviewed intensively by Axpo for these two sites.

During the FA fabrication the process and product qualifications at different levels are in the focus of Axpo's activities. Qualification of a procedure is of special importance if the resulting products cannot be tested thoroughly enough by mere non-destructive testing (as it is the case e.g. for welded parts).

A review of the fabrication documentation is performed by Axpo personnel before the start of a new FA fabrication campaign, both for reloads and for Lead Use Assemblies. For reloads already delivered earlier to Leibstadt or Beznau the main target is to review modifications regarding manufacturing facilities, fabrication processes, installations, inspection methods and products. Modifications have to be analyzed and are eventually subjected to review by Axpo. As a rule, modified fabrication processes can only be appliedafter their qualification, i.e. after their capability to produce reliably within specification requirements has been demonstrated. Often existing qualifications shall be applied for an Axpo campaign. In this case a review of the qualification reports is done for the first application. In other cases new qualifications are made and Axpo is following the qualification process from the beginning on. Axpo uses all qualification documents mentioned in Chapter 3 and checks the fulfillment of all required conditions and parameters during the surveillance inspections.

During the years, different findings regarding qualification programs and reports and their applicability in the fabrication have been recorded and shall be given here as examples.

During the review of the qualification reports the following observations were made:

- old qualification reports deviating from actual requirements or specifications
- qualification reports issued in spite of deviations occurred during the qualification procedure
- qualification reports not fully compatible with the requirements of the program or of the manufacturing documents,
- qualification reports with results copied from other qualifications, i.e. in principle not applicable to the current qualification
- operation procedures issued without qualification
- qualification reports with erroneous data
- qualification reports mentioned and used in spite of their appreciable age

In the production:
- manufacturing slightly outside of the qualified range
- use of non-qualified parameters
- conducting non-qualified operations/inspections
- different interpretation of the qualified instructions
- replacement of key components of installations without requalification
- qualified standards mentioned on documents do not exist

It was sometimes observed, that companies do not conduct:
- a periodically systematic control of the qualifications (no requirements)
- a comparison of the improvements conducted in the production with the original qualifications

More friction can occur when the manufacturing company also relies on different suppliers:
- It was found, that in some cases qualification philosophies between manufacturers and their suppliers were quite different and compromises had to be made; nonetheless the manufacturer were willing to take the full responsibility.
- Comparable differences can also be found between fabrication facilities of the same supplier but located in different countries.
- It was also found that qualified manufacturers are not regularly inspected as required.

7. Conclusions

Qualifications on different levels are crucial for reliable quality of nuclear FA. Adequate methods and strategies are available and should be strictly applied. Nevertheless, Axpo has experienced a number of inconsistencies with respect to qualifications of nuclear fuel fabrication processes and equipment. All these issues could be solved without jeopardizing the delivery. Despite of the significant efforts spent on reviewing and challenging qualification reports, Axpo considers this approach as key for ensuring the delivery of flawless defect free FA to our power plants.

References

[1] Axpo paper by Top Fuel 2010: Fuel Assembly Fabrication Monitoring and Inspection. 30 Years in Charge

[2] Regulatory guidelines HSK-60 and HSK-61

[3] Deviller R. et al., Fundamentals of Spatial Data Quality, Lavoisier 2005

[4] Kristeller P.O., Geschichte zur Rhetorik und zum Begriff des Menschen in der Renaissancen, Gratia-Verlag Göttingen, 1981

[5] Harper D., Online Etymology Dictionary, 2001-2012

[6] Wikipedia

[7] ISO – Quality Management Standards, February 14 2012 http://www.iso.org/iso/about/the_iso_story/iso_story_quality_management_standards.htm

[8] Dietrich E., Statistische Verfahren zur Maschinen- und Prozessqualifikation, Carl Hanser Verlag, 2005

[9] Strasser, A. et al. Fuel Fabrication Process Handbook. s.l. : ANTInternational, 2005. ANTI 05-0008R.

[10] IAEA. Guidebook on Quality Control of Water Reactor Fuel. Vienna : IAEA, 1983. ISBN 92-0-155083-9.
[11] Cowel, B.S. et al. Fuel Qualification Issues and Strategies for Reactor-Based Surplus Plutonium Disposition. s.l. : OAK RIDGE NATIONAL LABORATORY, 1997. ORNL/TM-13405.
[12] Rogge, K.-T. et al., Introduction of the New Processes and Quality Control Methods in Fuel Fabrication at Siemens/ANF, Lingen (D) : s.n., 18-22 October 1999. IAEA-TECDOC-1166.
[13] Doerr, W. et al., Qualification of MSZ Elektrostal for Manufacturing SIEMENS REPU Fuel Assemblies, Top Fuel 1997, BNES
[14] ISO. Statistical methods - Process performance and capability statistics for measured quality characteristics. 2006. ISO 21747:2006.
[15] Dietrich, E. Statistische Verfahren zur Maschinen- und Prozessqualifikation. s.l. : Carl Hanser Verlag, 2005.
[16] Weidinger, H.G., Statistical Methods to Assess and Control Processes and Products during Nuclear Fuel Fabrication. Bratislava (SK) : IAEA, June 23 1999. Regional Training Course on Fuel Design, Fabrication, Performance and Back-End.
[17] Rinne, H. et al. Prozessfähigkeitsmessung für die industrielle Praxis. München - Wien : Carl Hanser Verlag, 1999.
[18] Balabanov, S., Qualification of Technological Processes and Technological and Matrological Instruments used at OAO MSZ. Bratislava (SK) : IAEA, June 22-23 1999. IAEA Regional Training Course.
[19] ISO. Statistical Interpreatation of Data - Tests for Departure from the Normal Distribution. s.l. : ISO, 1997. ISO 5479:1997.
[20] Kopff, G., Integration of Process-Oriented Control with Systematic Inspection in Framatome-FBFC Fuel Manufacturing. Lingen (D) : s.n., 18-22 October 1999. IAEA-TECDOC-1166.
[21] ISO. Sampling procedures for inspection by attributes. 1999. ISO 2859-1:1999 .
[22] Statistical interpretation of data - Determination of a statistical tolerance interval. s.l. : ISO, 1975. ISO 3207:1975.
[23] Statistical interpretation of data - Techniques of estimation and tests relating to means and variances. s.l. : ISO, 1976. ISO 2854:1976.
[24] Baur, K. et al. Fuel Quality Management Handbook. Skultuna (S) : ANTInternational, 2010.
[25] Collani, E., Möglichkeiten und Grenzen er Statistik. Lingen (D) : s.n., 26-27 November 2001. Statistikseminar E.ON.
[26] ASTM. Standard Guide for Qualification of Measurement Methods by a Laboratory Within the Nuclear Industry1. s.l. : ASTM, 2003. ASTM C 1068 – 03.
[27] Dietrich, E. et al. Statistische Verfahren zur Qualifikation von Messmitteln, Maschinen und Prozessen. München - Wien : Carl Hanser Verlag, 1998.
[28] Guide to the expression of uncertainty in measurement (GUM). ISO ENV 13005.
[29] Decision limit, detection limit and determination limit under repeatability conditions – Terms, methods, evaluation. s.l. : Deutsches Institut für Normung e.V., 2008. DIN 32645.

EXPERIMENTAL STUDY OF FA TO CORE BAFFLES IMPACTS UNDER SIMULATED SEISMIC LOADINGS

V.V. Makarov, A.V. Afanasiev, I.V. Matvienko
"Rosatom", OKB HYDROPRESS, street Ordzhonikidze 21, 142103,Podolsk, Russia
A.B. Dolgov, S.E. Volkov
"Rosatom", JSC "TVEL", Kashirskoe highway 49, 115409, Moscow, Russia

ABSTRACT

The paper covers the methods and results of the experimental studies of the forces that arise in VVER and PWR FA-K mockups as they interact with the core baffle simulator when seismic loads are applied to the FA supports in accordance with the calculated accelerograms for the supports. Piezoelectric dynamometers were placed on two opposite sides to measure the force of FA-to-column collision. In the course of the tests the gap between the FA and the core baffle varied within 1 to 7 mm and the earthquake intensity varied as well.

When the calculated accelerograms are applied to the VVER FA supports the FAs collide many times over by the opposite sides with the core baffle. The impact lasts 20-30 μs. In the implemented test procedure with three FA-to-core baffle contact belts the maximum forces were observed on all the three reference spacer grids depending on such factors as the earthquake intensity, medium and the FA-to-core baffle simulator gap value. The maximum forces of interaction decrease considerably at medium change-over from air to still water and from still water to flow. The collision forces increase as the earthquake intensity rises. It is necessary to take into account the change in coolant thermophysical properties, mechanical properties of the FA structural materials, differences between the schemes of contacting the FA-to-fuel assemblies around and the reactor internals as well as the test column and the core baffle simulators to obtain an assessment of the reactor conditions in the experiments performed. No residual distortions of the spacer grid were detected following the performed seismic tests with FA collision with the core baffle simulator.

Keywords: experimental, studies, force, strength, impact, FA, VVER, PWR, core baffle, core, earthquake, magnitude.

1 Introduction

One of the requirements of the regulatory documentation for NPP reactor FAs is the assurance of strength and stability under seismic inputs up to safe shutdown earthquake (SSE). One of essential problems is the determination of the forces compressing the spacer grids in case of FA collision with the core baffle as the safe shutdown earthquake affects a reactor.

The purpose of the tests was to study the forces that arise in single VVER and PWR FA mockups as they collide with the core baffle simulator when seismic loads are applied to the FA supports.

The task of the tests was to determine the effect of the gap size, earthquake intensity (magnitude 6, 6,5 and 7) and medium (air, still water and flow) on the maximum force of FA-to-core baffle simulator collision.

The test results are used for the verification of FA strength calculations.

2. Program and methods of VVER FA tests

Two single FAs were studied in the seismic and vibration test facility (Fig.1) that allows applying the calculated seismograms for FA supports.

The main systems of the test facility are the hydraulic system, vibration load system, vibration test instrumentation and control system, process parameter monitoring system.

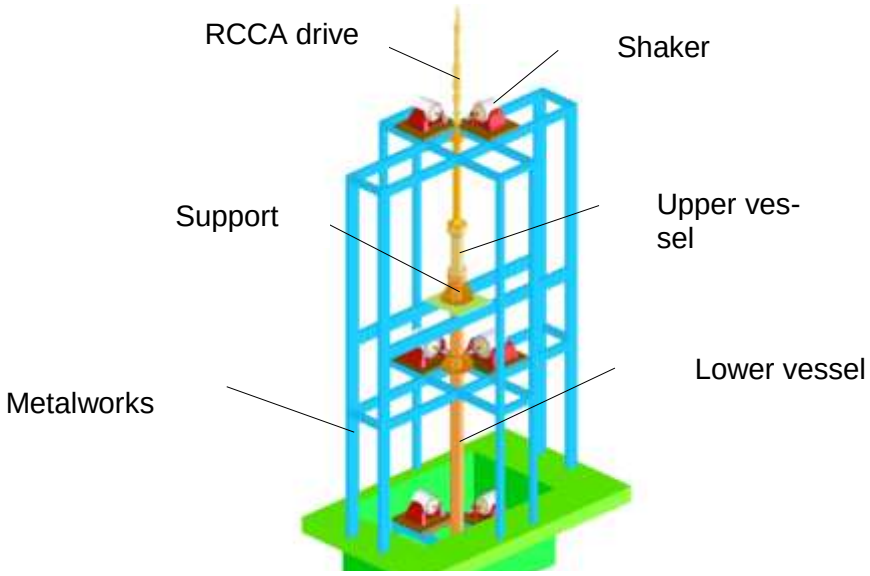


Figure 1 – General view of facility

The tested FA was installed in the test vessel that is the core baffle model.

VVER FA mockup contains 13 spacer grids with cells 30 mm tall, cell wall thickness 0,3 mm and the length of bulge-to-fuel rod contact line (6,00 ± 0,50) mm. The mockup corresponds to fresh fuel with fuel rod-to-spacer grid cell interference gap.

The facility column (Figure 2) is a hexahedral prism (for VVER FA) or a square prism (for FA-K). In the lower and upper parts the facility column is provided with chambers with four inlet nozzles (in the lower part) or outlet nozzles (in the upper part). The upper and the lower FA support ensure standard conditions of FA fastening in what refers to the geometrical dimensions of the seats. Nine transparent observation windows are located on two opposite sides of the column to make it possible to measure the vibration response of FAs and fuel rods with laser vibrometers. When the force of VVER FA impact with the core simulator was measured in the seismic tests, six windows opposite to the middle spacer grids were replaced with strain gauges. The control forces were chosen for earthquakes at particular sites of European NPPs with both VVER and PWR. The frequency range of the simulated loads varied from 1,5 to 50 Hz.

3. Test Results for VVER FA

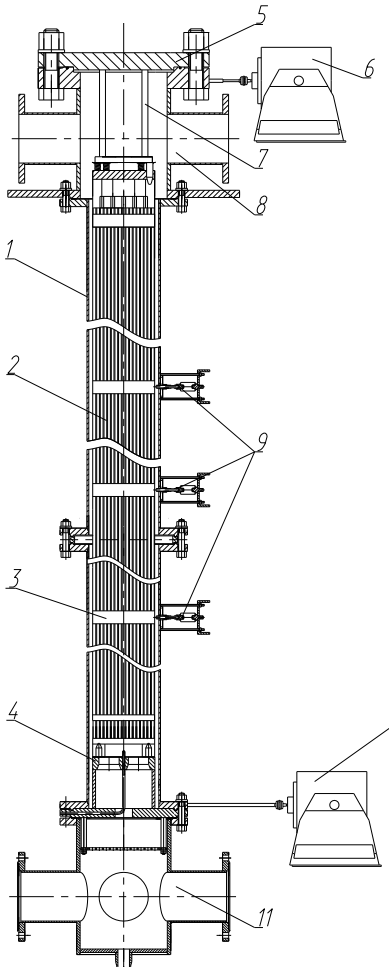
The difference in the seismograms applied in the tests and the calculated ones did not exceed 10%. The dependences of the forces of FA interaction and the baffle simulator testifies to the effect that a collision can take place in one, two or three of the spacer grids with a time shift not above 30 μs. The collision lasts about 20-30 μs. In the uniaxial excitation of vibration the mockup collided in turn with the opposite sides. The collisions take place the moments the direction the supports move in changes for the opposite and the FA keeps on moving in the previous direction. The maximum collision forces were more often observed at the level of the middle spacer grid SG8. The dependencies for the force of collision versus the acceleration amplitude (earthquake intensity) are provided in Figure 3. The maximum forces at earthquake magnitude 7 were observed for the gap of 6 mm. At

earthquake magnitude 6 and 6,5 the maximum forces in the air were also observed for the gap of 6 mm and in the flow at 4 mm, which is related with a decrease in the amplitude of mutual displacements of the FA and the column. At transition from air to still water and from still water to flow the forces of collision the observed decrease in the forces was 1,5-2 which was caused by an increase in the damping forces of the medium.

4. A study of the forces for square FA-K -to-core baffle interaction under seismic loads

The mockup of FA-K of a 17×17 fuel rod array for PWR reactor consists of a top nozzle, a rigid skeleton, fuel rods with lead fuel pellet simulators and a bottom nozzle. The width-across-the flats for the mockup of a FA-K of is 213,5 mm and its overall height in the free state is 4100 mm.

The rigid skeleton consists of 24 guide tubes and a central tube with spacer grids, mixing grids, and anti-vibration grids welded to them. The plate-type spacer grids are provided with spring elements of cell type optimized for the force of fuel rod motion. SG1 is placed above the anti-vibration grid. The program and the methods are similar to those described above for the VVER FA tests. The collision force was measured on the middle spacer grids Nos. 3,4,5. All the studies were made at longitudinal compression of the FA-K by 22 mm.



1 – vessel;
2 –FA-K mockup;
3 – spacer grid
4 – support tube;
5- cover
7 – PTU simulator;
8 – upper chamber;
9 – strain gauges;
6, 10 – shaker;
11 – lower chamber

Figure 2– Column with FA installed in it

The collisions can take place with one, two or three spacer grids with a time shift not above 30 μs. The collisions last 20-30 μs. In the uniaxial excitation of vibration implemented in the experiment the mockup collides with the opposite facets in turn. For the first 2 s since the beginning of reproducing the accelerograms the collisions take place before the outermost positions are reached and at this the impact comes on the opposite facet, i.e. the square FA-K and the column move in the antiphase. Further on the collisions are more or less spontaneous which confirms lack of distinct correlation between the movements of the square FA-K and the column.

The maximum forces were mostly observed (in 60 % of the experiments) on spacer grid No.3.

The dependence of the square FA-K -to-internals simulator maximum collision forces versus the gap value is provided in Figure 3.

The maximum forces were noted as a rule at gaps of 3 and 4 mm. In coolant flow the collisions only occur at earthquake of magnitude 7 , the forces of interaction are maximum at gaps of 2 and 3 mm. At a gap of 4 mm the force exceeds only 1,5-2 times th elevel

of the noise of the measurement channels which allows assuming that at a further increase in the gap there will be no collisions of the square FA-K with the internals.

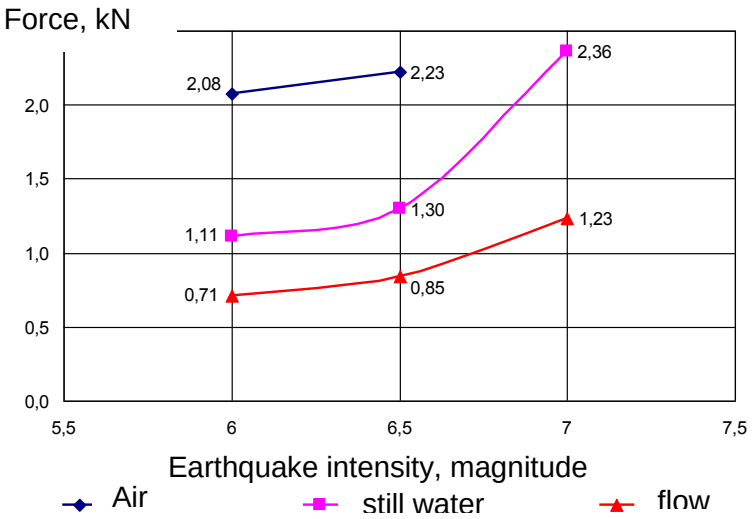


Figure 3– Maximum force of VVER FA collision versus acceleration amplitude

The collision forces with the earthquake intensity are provided in Figure 3. Under earthquakes of intensity below the threshold value the square FA-K does not interact with the internals as the amplitude of mutual movements of the square FA and the facility column does not exceed the value of the gap. In the experiments performed there was no interaction in the flow at earthquake magnitude 6 and 6,5. At earthquake intensities that exceed the threshold value the experimental results for still water state that the force of collision increases as the intensity increases but at this the rate of the force increment drops. The environment produces considerable effect on the interaction forces which are reduced at same intensities and gap at transfers from air to still water and from still water to flow. So, the maximum forces of collision in the air are higher than in still water 2,7 times over at magnitude 6 and 1,1 times over at magnitude 6,5 . In still water under earthquake magnitude 7 the maximum force is 3,4 times higher than in coolant flow. The mentioned effect is explained by the increase in vibration damping at transfers from air to still water and from still water to flow.

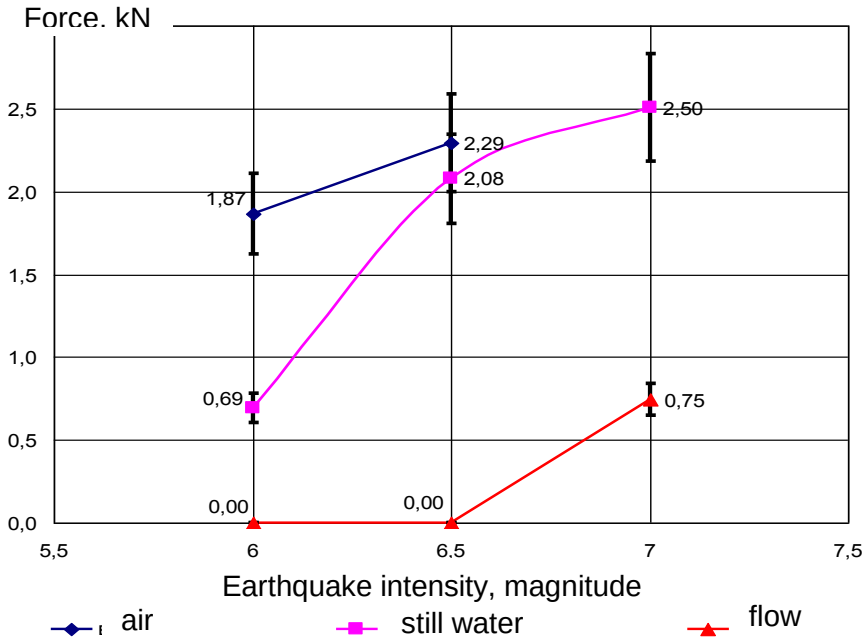


Figure 4 – Maximum force of FA-K collisions versus earthquake intensity

5. Analysis of the Results

The effect of the reduction in bending rigidity in response to the impact force that simulates the status of a FA at the end of the fuel cycle is still unknown as there were two dummy FAs tested. The bending rigidity of the tested square FA-K is twice as low as the bending rigidity of the VVER FA dummy, with 5 mm bow in the middle part of the two FAs. Reference /1/ states that there are no FA oscillations on resonance frequencies. In our opinion, smaller impact forces of FA-K are explained by a relatively higher pressure loss and FA damping due to larger areas of the facets and spacer grids in comparison with the hexahedral spacer grid at transverse FA motion inside the column. These conclusions do not contradict to the results provided in /2-4/.

The maximum increased time of RCCA drop due to seismic loads from SSE /1/ was 1,32 s for VVER and 0,23 s for FA-K.

The purpose of the tests was to verify the computer codes and the obtained results are not a direct confirmation of the seismic stability of the investigated FAs as the experiments do not model the either multiple FA collisions against each other nor as they crash moving towards one another. Besides, the limited volume of water in the experimental facility column can tell on the damping coefficients in the tests as compared to the value for a tested set of FAs. Re-distribution of the force interaction between the FAs over the entire core volume even if the seismic effect is directed along one of the spacer grid symmetry axes is an important effect not modeled in the test.

6. Conclusion

1. The methods was elaborated and the investigations made of the forces that arise in VVER and PWR FA mockups as they interact with the core baffle simulator when seismic loads are applied to the FA supports in accordance with the calculated accelerograms for the supports at earthquakes magnitude 6, 6,5 and 7 as per MSK scale.

2. Considerable effect of the medium, earthquake intensity and initial FA-to-core baffle gap on the FA-to-core baffle interaction force was observed. Positive effect of vibration damping and a decrease in collision forces in water flow was also observed.

3. The maximum forces obtained in the experiments are several times lower than the forces that arouse plastic deformations in the spacer grids for the both FAs.

4. No residual deformations of the spacer grid were detected following the performed seismic tests in both FAs.

7. References

1. V.V. Makarov, and a., “Experimental Studies Of Proper And Forced Oscillations Of Fuel Assemblies In The Coolant Flow “, WR FPM T1-048, Chengdu, 2011.
2. QUEVAL J.C., BROC D., RIGAUDEAU J.,VIALLET E.-“Seismic Test On Interacting Full-Scale Assemblies On Shaking Table” SmiRT 16-Paper #1691-Washington-August 2001.
3. E.VIALLET and T.KESTENS, “Prediction of Flow Indused Damping of a PWR Fuel Assembly in Case of Seismic and Loca Load Case”, SmiRT 17-Paper #CO 01-1,Prague, August 2003.
4. E.VIALLET and a., “Validation of PWR Core Seismic Models with Shaking Table Tests on Interacting Scale 1 Fuel Assemblies”, SmiRT 17-Paper #CO 01-2,Prague, August 2003.

European Nuclear Society
Rue Belliard 65
1040 Brussels, Belgium
Telephone: +32 2 505 30 50 - FAX: +32 2 502 39 02
topfuel2012@euronuclear.org
www.topfuel2012.org