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THE LESSONS LEARNED FROM CONVERSION OF 10 MW RESEARCH REACTOR TO LOW ENRICHMENT FUEL

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ABSTRACT

The conversion of the 10 Mw research reactor of the Institute of Nuclear Physics in Tashkent from 90% enriched fuel to the low (19.7% enrichment) one was done in two major steps. First step, which started in August 1998 and completed in February 1999, was including a conversion of reactor to 36% enrichment fuel. The second stage (started in February 2008 and accomplished in November 2009) allowed the full conversion of reactor to 19.7% enrichment fuel. In parallel with these activities the continuous works on testing of the new types of fuel elements were carried out.

We present the review of our experience obtained from conversion of research reactor to the low enriched fuel including theoretical estimates, the chosen geometry of core elements, the results on neutron flux measurements as well as other data on reactor utilization which were accumulated after full conversion of reactor.

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1. Introduction.

The water-water research reactor WWR-SM of the Institute of Nuclear Physics of Uzbekistan Academy of Sciences (Tashkent) was commissioned in October 1959. From 1959 till 1978, reactor has been operating at power of 2 MW. After reconstruction and upgrading works in period from 1974 till 1977 its power has been increased up to 10 MW. The reactor belongs to the standard type of research reactors supplied in 1960-70's by former Soviet Union to some nuclear research centers including several countries in the Eastern Europe.

At earlier stages reactor was using the 10%-enrichment fuel and then starting from 1979 it has been converted to 90%-enrichment fuel. That type of fuel has been used till 1997 and in 1998 reactor was converted to 36%-enriched fuel as a first stage of program of full conversion to 19.6% fuel by 2008-2009.

Currently reactor is used (the average operational time is about 4000 hours per year) for basic and applied research in nuclear physics, radiation physics of condensed matter, modification of materials, radiochemistry, activation analysis as well as for testing new types of reactor fuel and reactor materials. Applied research includes creation of technologies for new type of isotopes, application of neutron diffraction methods, studies in radiation hardness etc. The most of its time reactor operates for production of radioisotopes and irradiation services (for more information about utilization of reactor see e.g.[1]). In 2006 the highly enriched spent fuel from reactor was sent to Chelyabinsk (Russia) in the framework of RRRRF program [2].

2. Reactor Parameters and Conversion to Low Enriched Fuel

The reactor WWR-SM has cylindrical core of about 58 cm in diameter and 60 cm height, 10 control rods of which 3 are safety rods (B_4C), 6 - shim rods (B_4C) and one is automatic regulating rod. The neutron reflector consists of beryllium blocks. The reactor has 44 vertical and 9 horizontal channels, the average total flux of neutrons is about $2.0*10^{14}$ neutrons per cm^2 sec. Normally reactor operates from 5000 to 6000 hours per year depending upon requests and orders. There is one 480 hour operating cycle per month.

After commissioning in October 1959 reactor was using 10%-enrichment fuel of type EK-10 and had thermal power 2 MW. Since 1971 reactor started to operate with new fuel, 90%-enrichment of IRT-2M type. The upgrading and reconstruction works performed in period from 1972 till 1978 allowed to increase the thermal power of reactor up 10 MW and since 1979 till the middle of 1998 the machine was operating with use of 90%-enrichment fuel of IRT-3M type.

From the end of 1998 till March 2008 reactor was operating with 36%-enrichment fuel also of IRT-3M type. The fuel assembly (FA) with 36%-enrichment is made of (UO_2-Al)- alloy (meat) with Al cladding and has 880mm in length of which the fuel meat part is 600mm. There are 6 concentric layers (5 square round tubes inside and 1 external one). The fuel meat thickness is 0.5mm and cladding thickness is minimum 0.3mm. The average burn-up of fuel was not less than 60%. This has been achieved by several experiments which confirmed the firmness and good behavior of fuel elements despite the producer's 40% of burn-up warranty. In order to keep the neutron flux at previous level some changes the geometry of fuel load have been done and additional fuel assembles (up to 24 pieces with total content of U-235 6.4 kg) were added.

As a first step in full conversion of reactor to 19.7% fuel in March 2008 reactor started to operate with a mixture of sixteen 36% - and four 19.7%-enrichment fuel assembles (IRT-4M type). In total 11 cycles (with shutting down and re-starting reactor, unloading highly enriched and downloading low enriched fuel, measuring the isotope content of water, controlling all necessary

parameters of reactor operation) were carried out and no any noticeable deviations in reactor operation were recorded. In May 2008 another four additional 19.7%-enrichment IRT-4M fuel assembles were loaded instead of 36% - enrichment FA. And on 17 November 2009 the reactor was fully downloaded with twenty 19.6% FA only.

3. Conclusion

From our experience gained till now we can conclude that after full conversion of reactor to 19.7%-enriched fuel the fluxes of neutrons did not changed dramatically. For example the flux of thermal neutrons ($E_n \leq 0.625$ MeV) decreased by about (7-10)% and within experimental uncertainties that is in consistence with theoretical calculations which were performed earlier in cooperation with Argonne National Laboratory team [3]. At the same time we observe up to 7% increase in the flux of fast ($E_n \geq 0.821$ MeV) neutrons which means that irradiation time for production of some types of isotopes (for example P-33) remains practically at the same level if not less. Also while operating with 19.7% enriched fuel we did not observe any noticeable changes in operational parameters of reactor like the reactivity coefficient, the chemical composition and radioactivity of coolant, the levels of aerosol and radioactive gases etc[5,6]. The fuel burn-up reached has 60%.
However, one should notice that unexpected increase of fuel cost within the short period of time and increase in number of fuel assembles per load resulted in operational cost of reactor.

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A consideration for uranium high density plate type target of uranium metal particles dispersion in aluminum matrix for 99Mo production

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ABSTRACT

In order to convert HEU to LEU for fission Mo target, higher uranium density material is more beneficial. The uranium aluminide targets used world widely for commercial ^{99}Mo production are limited to 3.0 g-U/cc in uranium density in the target meat. A consideration of high uranium density using uranium metal particles dispersion plate target is taken into account. The dispersion target is quite same as the dispersion fuel of research reactors in the fabrication aspect. When the uranium particles are produced by atomization process, the spherical shape particles would enable the uranium particles volume fraction up to 50 volume %. Consequently the maximum achievable uranium density would be achievable up to about 9.0 g-U/cc in the target meat. The temperature increase of the dispersion target center was calculated to be about 20 °C, which is considered almost not to cause t the degradation of the target integrity by irradiation. In order to enhance the fuel performance some alloying elements such as Si, Cr, Fe, and Mo known as grain refinement could be added to uranium particles. A small amount of Si could be alloyed to aluminum of matrix phase for retarding the interaction between uranium particles and aluminum matrix. In separation process the aluminum material of matrix phase and cladding could be removed by alkaline dissolution as a low level of radioactive waste. The remained residue could be treated following the Cintichem process.

1. Introduction

The medical isotope of ^{99}Mo has been produced mainly by extracting fission products of uranium. From productivity aspect highly enriched uranium has been used as the raw material for fission Mo. Now a days HEU minimization efforts in fission Mo production are underway in connection with global threat reduction policy. In order to convert HEU to LEU for fission Mo target, higher uranium density material could be applied. From the stable behaviour during irradiation UAlx and UO_2 have been mainly used as fission Mo target material. The HEU UO_2 annular target could be replaced easily with the uranium foil target developed by ANL. This study focused on the dispersion plate target, which is very similar with the dispersion plate fuel of research reactor. The uranium aluminide targets used world widely for commercial ^{99}Mo production are limited to 3.0 g-U/cc in uranium density of the target meat. A consideration of high uranium density using uranium metal particles dispersion plate target is taken into account. When fission Mo target is compared to research reactor fuel, the irradiation burnup is much lower such as 6 % of fission fragments. The irradiation period is very short as less than 7 days. Pure uranium material has higher thermal conductivity than uranium compounds or alloys, which are uranium aluminide, uranium silicide and U-Mo. It is considered that the degradation by the irradiation would be almost negligible.

In this study using the computer code of the PLATE developed by ANL the irradiation behavior was estimated. Some considerations were taken to improve the irradiation performance further. It has been known that some alloying elements of Si, Cr, Fe, and Mo are beneficial for reducing the swelling by grain refinement. Recently in the RERTR program the interaction problem could be solved by adding a

small amount of Si to aluminum matrix phase. The fabrication process and the separation process for the proposed atomized uranium particles dispersion target were reviewed.

2. Suggestion of A Very High Uranium Density Plate Target

In connection with converting HEU to LEU higher uranium density material has been requested as a candidate of fission Mo target to secure the fissionable material of ^{235}U. In developing the high uranium density fuel development of research reactor fuel, the various uranium densities were reviewed for various candidate materials as in Table 1. From the density point of view pure uranium is best for LEU fission target material. The uranium density of pure uranium metal is 19.05 g/cm^3, which is 4.2 times than uranium density of UAlx. It was reported that the uranium density of the fission Mo target in South Africa is about 3.0 g-U/cm^3. The UAl$_2$ phase in the mixture UAlx is presumed to be higher than the mixture UAlx in the table.

Table 1. Density date for fuel compounds.

Material	Density (g·cm^{-3})	U density (g·cm^{-3})	Material	Density (g·cm^{-3})	U density (g·cm^{-3})
UAl$_4$	5.7	3.7	U$_3$Si$_2$	12.2	11.3
UAl$_3$	6.8	5.1	U$_3$Si	15.3	14.7
UAl$_2$	8.14	6.64	U-10wt%Mo	17.2	15.5
*UAl$_x$	6.4	4.6	U-7.5wt%Mo	17.6	16.3
U$_3$O$_8$	8.3	7.0	U-5wt%Mo	18.0	17.1
UO$_2$	10.96	9.67	U	19.05	19.05

* A mixture of 69wt.% UAl$_3$ and 31 wt%.% UAl$_4$

In KAERI, atomization technology for producing uranium silicide as well as U-Mo powders had been developed in 1990s. In the mean time some pure uranium metal powder has been produced many times to provide to an advanced nuclear fuel development. So it is considered that the pure uranium metal powder can be supplied without any difficulties.

The fabrication process of UAl plate target was reported as in Fig. 1. If fission Mo target meat is supposed atomized uranium powder, atomized U metal powder is mixed with aluminum powder and pressed to a compact using a forming die. The compact is loaded in the space among a frame and cover plates of upper and bottom sides as in the dispersion plate fuel fabrication process for research reactor. The other steps are almost same as the fabrication process of UAl plate target. It is considered that the process of atomized particles dispersion plate target could be established with some modification.
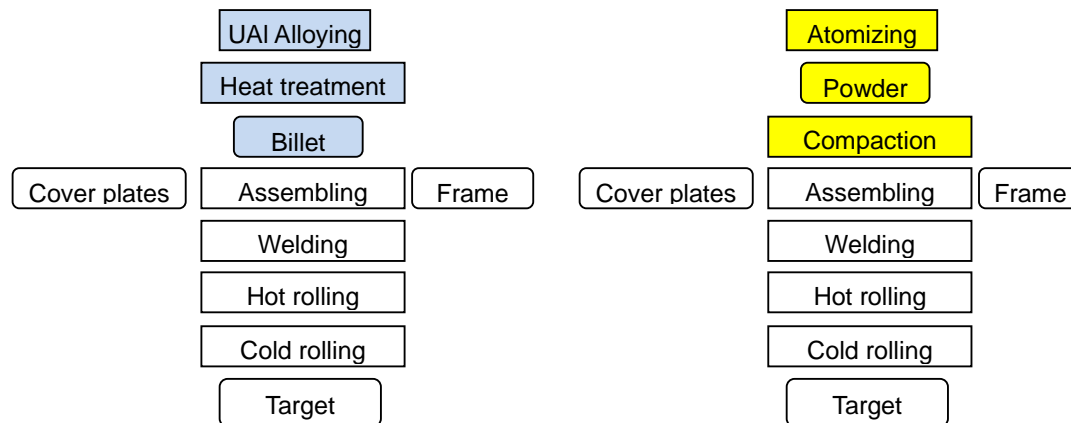

Fig. 1.   Comparison for the fabrication processes of UAlx plate target and atomized particles dispersion plate target

Atomized uranium particle tends to have a spherical shape. In general the spherical particles powder has better plasticity in rolling work than irregular particles powder due to smooth surface. The average fabrication porosity of research reactor fuel dispersed with atomized particles appeared to be very small such as less than 3 Vol.% from the better plasticity. The average volume of U3Si2 dispersion fuel by rolling work was about 45 Vol.%. The maximum volume fraction of dispersion fuel was reported about 55 Vol.%. If the volume fraction is supposed 50 Vol.% in case of atomized spherical particles powder, the available highest uranium density would be 9.2 g-U/cm^3. When HEU fission Mo target of 1.5 g-U/cm^3 is converted to LEU, uranium density of LEU is required more than 7.5 g-U/cm^3. It is considered that the atomized U metal particles dispersion plate target of more than 9.0 g-U/cm^3 would be much enough.

3. Estimation on target temperatures during irradiation

The fission Mo target should maintain the integrity during the irradiation. The 1st major degradation of the target is swelling. Especially uranium metal is very weak in temperature. Accordingly the target center temperature was calculated using the PLATE computer code developed by ANL with supposing that heat flux and cooling water speed are 250 W/cm^2.and 6 m/sec, respectively. The average thermal conductivity of atomized particles dispersion was calculated 85 W/m-K. The temperature increases for uranium particles dispersion target with thickness 1.0 mm and cladding with thickness 0.3 mm were calculated 4.5 °C and 1.5 °C. The temperature increase at the interface between cladding and coolant water was about 25 °C. When the coolant temperature at outlet is supposed 40 °C, the peak temperature at the target center is estimated 71 °C. It is considered that these values are negligible from the aspect of affecting the thermal inducing swelling. It is considered that these values have enough safe margins from the aspect of affecting the thermal inducing swelling. The center temperature of about 71 °C is presumed to be too low to induce an interaction at the interface between uranium particles and aluminium matrix.

4. Application of Other Fuel R&D Experiences to improve Dispersion Target Performance

In uranium metal fuel the irradiation behaviour could be improved by alloying some elements of Fe, si, Al, Cr, and Mo. The irradiation test could be extended to greater than 10,000 MWd/t at temperature up to 400 °C. Silicon concentrations of 250 to 350 ppm alone, or in conjunction with other alloying elements, especially Mo, were most effective for exposures up to 5000 MWd/t. At exposures over 10,000 MWd/t, alloys containing 800 ppm Al in conjunction with other alloying were more swelling-resistant than those containing 250 to ppm silicon.

The fission Mo target is irradiated less than 8 at% burnup, which is approximately equivalent to 15,000 MWd/t. As shown in the above, the fission target temperature is much lower than the metal fuel temperature due to very thin thickness. The irradiation period of the fission target in reactor is much shorter than that of research reactor fuel. It is considered that the integrity of the atomized uranium particles dispersion target could maintain up to the aiming irradiation.

The dispersed uranium particles containing those kinds of alloying elements can be easily made by adding the alloying elements into crucible when the melting operation is prepared. The atomization process has an advantage of very rapidly solidification in forming particles. So the grains inside the particles are very fine. The grain sizes for atomized U particles were measured to be a few microns while the grain sizes of uranium common cast ingot were measured several hundred microns. There would be possible not to add the grain refining alloying elements.

In developing U-Mo dispersion fuel for high performance research reactor an interaction problem between dispersed U-Mo particles and Al matrix occurred. Silicon addition to aluminium matrix was found to give a good effect on retarding the interaction rate. If it is supposed that the interaction at the interface between uranium particles and aluminium matrix occurs, the dispersion target temperature would increase. Then the thermal conductivity would decrease gradually. As the irradiation continues, the temperature increasing rate would be accelerated. However, the calculated temperature is rightly

estimated, such a quite low temperature would not approach up to the accelerating stage criteria.

5. Advantages on other areas

In the aspect of the fabrication process, the existing fabricators can adapt this atomized particles dispersion process if atomized U powder is supplied effectively. The additional steps such as mixing of u powder and Al powder and compacting are very common and proven technology in research reactor fuel fabrication. Presumably the cost could not increase much for adapting the additional steps.
In dissolving steps for separating 99Mo from the solution, the fission target can put into the dissolving vessel. Because the target is not opened, any leakage radioactive gas would not take place. If aluminium matrix and cladding is dissolved by NaOH and the solution containing aluminium is decanted, aluminium material can be separated as a low level- waste. Then the remaining particles can be dissolved by HNO_3. The following steps are same as the CITICHEM process. Most of fission products are contained in the uranium particles. The high radioactive waste would be produced as a small quantity. Presumably the radioactive would be more easily managed.

6. Conclusion

Taking an advantage of atomization technology developed in KAERI, a very high uranium fission target with atomized uranium particles dispersion in aluminium matrix is taken into a consideration. If the volume fraction is supposed 50 Vol.%, the available highest uranium density would be more than 9.02 g-U/cm^3. The temperature increases for uranium particles dispersion target with thickness 1.0 mm and cladding with thickness 0.3 mm were calculated 4.5 °C and 1.5 °C. The temperature increase at the interface between cladding and coolant water was about 25 °C. It is considered that these values have enough safe margins from the aspect of affecting the thermal inducing swelling. The atomization process has an advantage of very rapidly solidification in forming particles. So the grains inside the particles are very fine. The dispersed uranium particles containing alloying elements of Fe, Si, Al, Cr, and Mo, which are used for grain refinement, can be easily made by adding the alloying elements into crucible when the melting operation is prepared. These additions could improve the irradiation behaviour of target. In the aspect of the fabrication process, the existing fabricators can adapt this atomized particles dispersion process if atomized U powder is supplied effectively. Presumably the production cost could not increase much for adapting the additional steps. In dissolving steps for separating 99Mo from the solution, the fission target can put into the dissolving vessel. Because the target is not opened, any leakage radioactive gas would not take place. If aluminium matrix and cladding is dissolved by NaOH and the solution containing aluminium is decanted, aluminium material can be separated as a low level- waste.

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QUALIFICATION OF URANIUM-MOLYBDENUM ALLOYS FOR RESEARCH REACTOR COMMUNITY

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ABSTRACT

Uranium-molybdenum (U-Mo) alloys are being produced to refuel international research reactors – replacing current highly-enriched uranium fuel assemblies. Over the past two years, Y-12 Analytical Chemistry has been the primary qualification laboratory for current U-Mo materials development in the U.S. During this time, multiple analytical techniques have been explored to obtain complete and accurate characterization of U-Mo materials. For the chemical characterization of U-Mo materials, three primary techniques have been utilized: (i) thermal ionization mass spectrometry (TIMS) for uranium content and isotopic analyses, (ii) a combination of inductively-coupled plasma (ICP) techniques for determination of molybdenum content and trace elemental concentrations and (iii) combustion analyses for trace elemental analyses. Determination of uranium content, uranium isotopic composition and elemental impurities by combustion analyses (H, C, O, N) required only minimal changes to existing analytical methodology for uranium metal analyses. However, spectral interferences (both isobaric and optical) due to high molybdenum content presented significant challenges to the use of ICP instrumentation. While providing a brief description of methods for determination of uranium content and H, C, O and N content, this manuscript concentrates on the challenges faced in applying ICP techniques to qualification of U-Mo fuels. Multiple ICP techniques were explored to determine the effectiveness (e.g., accuracy, precision, speed of analysis, etc.) for determining both molybdenum content and trace elemental impurity concentrations: high-resolution inductively-coupled plasma mass spectrometry (HR-ICPMS), inductively-coupled plasma quadrupole mass spectrometry (ICP-QMS) and inductively-coupled plasma optical emission spectroscopy (ICP-OES). The merits and limitations of these techniques for qualification of U-Mo alloys are presented, to include the limits of quantitation and uncertainties of measurements regarding the most efficient methods for qualifying the U-Mo alloys.

1. Introduction

While the development of U-Mo fuel for Reduced Enrichment for Research and Test Reactors program has been underway for many years, the development of modern analytical techniques for qualifying this fuel has lagged significantly. The modeling and ultimate performance of this high density fuel will depend heavily on the chemical composition of the fuel and all elemental constituents in the final product. While the determination of the uranium and molybdenum content and the uranium enrichment level is of utmost importance, the elemental impurities in the resulting fuel product must be included to provide accurate models for the performance of the fuel. Of special consideration, the trace-level concentrations of nuclear fuel poisons (e.g. boron, cadmium, dysprosium, gadolinium, samarium, etc.) must be determined as accurately as possible to model the precise flux produced by the fuel assembly.

Standard methods exist for measurement of uranium isotopic content (by thermal ionization mass spectrometry[1]) and total uranium content (by either isotope dilution mass spectrometry[2] or Davies-Gray titration[3]). In addition, the determination of trace-level hydrogen, carbon, nitrogen and oxygen in uranium-molybdenum alloys do not differ significantly from standard methods of measurement for these elements in metals and alloys (for example, see reference 4). However, determining the concentration of elemental constituents and (trace-level) metallic impurities in alloy systems can provide a significant challenge to modern (trace-) elemental analysis techniques. Inductively-coupled plasma (ICP) techniques are widely used to determine the trace-level concentrations of elements in a variety of metals and alloys. Because of the highly energetic atomization (and ionization) efficiency of ICP-based analytical methods, multiple spectral interferences must be considered to determine the concentration of metallic impurities. In ICP optical emission spectroscopy (ICP-OES), emission from excited atoms and ions of elemental constituents can interfere with the measurement of emission from trace-level elements of interest. Likewise, isotopes from various elemental constituents can interfere with the measurement of trace-elements when using ICP mass spectrometry (ICP-MS).

In addition to simple atomic isobaric interferences in ICP-MS, the formation of polyatomic ions in the plasma can interfere with the measurement of elements of interest. Of specific concern to analyzing U-Mo materials for nuclear application, the multiple, abundant isotopes of molybdenum present a significant challenge to accurately determining the concentration of a number of elements by ICP-MS. This manuscript describes the multiple ICP-based techniques used to determine the concentration of both major (molybdenum) and trace-level constituents in U-Mo alloys being considered for the RERTR program.

2. Experimental

Figure 1 provides a general overview of the current techniques being employed for complete elemental analysis of U-Mo alloys. The uranium content and isotopic composition of the uranium is performed by isotope dilution mass spectrometry (using thermal ionization mass spectrometry).[5] Similarly, the concentration of trace-level hydrogen, carbon, nitrogen and oxygen are determined by standard methods based on the analysis of metals and alloys.[4] In these tests, ~0.5 g of solid material is combusted under various atmospheric conditions to produce gases that are measured by either infrared spectroscopy or gas chromatographic separation. These analyses (and results) for the U-Mo alloy did not differ appreciably from standard combustion analyses and are not provided in this manuscript. The concentrations of the remainder of elements in the periodic table are determined by either ICP optical emission spectroscopy or ICP mass spectrometry. The following sections give a general description of the methods used to dissolve the solid sample and present the dissolved solid to the various ICP instrumentation.

Dissolution of solid samples (received as either powders or foils) was achieved by weighing a solid aliquot (between 0.49g and 0.51g to nearest 0.0001g) of the sample and dissolving the solid in 20 mL of 8M nitric acid, 0.1M hydrofluoric acid, 0.1M hydrochloric acid and placed in a hot block digestion system. The sample was then heated at 70-75 $^{\circ}$C until completely dissolved. After cooling to room temperature, the solution was inspected for any non-dissolved solids and then diluted to 50 mL with ultra-pure water. [Precipitation of molybdate species was observed when HF was not used.] The resulting solution was used for all ICP-based analytical techniques – i.e. aliquots were removed and further processed according to technique being employed for measurement: high-resolution ICP mass spectrometry (HR-ICPMS), ICP Optical Emission Spectroscopy (ICP-OES), and/or ICP-Quadrupole Mass Spectrometry (ICP-QMS).

HR-ICPMS analysis represents the most fundamental analysis method employed. The dissolved solid solution was simply diluted to approximately 100 ug(solid)/mL and injected into the instrument. The instrument used for these studies is a ThermoFisher (Bremen, Germany)

Element2 high-resolution inductively-coupled plasma mass spectrometer (HR-ICPMS) with high abundance-sensitivity optics. The sample introduction consists of a Cetac AS-500 series autosampler followed by HF-resistant components (Elemental Sciences, Inc. Omaha, NE). The sample is nebulized through a PFA-Microflow self-aspirating nebulizer into a PFA PureChamber/PureCap spray chamber. The resulting aerosol then flows through an o-ring free sapphire injector, which is mounted in a quartz torch.

Figure 1: Techniques used to provide complete characterization of U-Mo fuels. Blue background denotes direct ICP-QMS analysis of diluted solution, yellow background denotes ICP-QMS analysis of eluent from column stripping and green denotes ICP-OES analysis of eluent from column stripping. Orange corresponds to ICP-OES %-molybdenum analysis, red corresponds to thermal ionization mass spectrometry and gray corresponds to combustion analysis.

ICP-OES and ICP-QMS analyses required multiple steps to prepare solutions that minimized spectral interferences (See Scheme 1). The molybdenum content was determined on multiple aliquots of the dissolved sample by dilution and injection of the resulting solution into a Prodigy High-Dispersion spectrophotometer with large-format programmable array detector. (Teledyne-Leeman). To determine the concentration of the trace metallic elements, a combination of ICP-QMS and ICP-OES techniques were used. The first step was removal of a small aliquot of the dissolved solid solution, dilution and injection of the resulting solution into an X-Series ICP-quadrupole mass spectrometer (ICP-QMS). The elemental concentrations determined in this step are shown as blue (MS-A), and the analysis was performed within 1-2 hours of dissolution due to significant loss of gold by precipitation from the solution. After removal of this small aliquot, the remainder of the solution is passed through a UTEVA column (Eichrom) to remove the uranium from the solution because the optical emission from uranium can interfere with measurement of multiple elements by ICP-OES. After collection of the eluent from the column separation, the solution is presented to both ICP-OES and ICP-QMS instrumentation. The concentrations of elements denoted by a yellow background are determined from ICP-QMS analysis of the resulting eluent (MS-B). The instruments used for both MS-A and MS-B analyses are ThermoFisher Scientific X-Series inductively-coupled plasma quadrupole mass spectrometers (ICP-QMS). The concentrations of elements denoted by green background are determined from ICP-OES analysis of the resulting eluent solution using a Prodigy High-Dispersion spectrophotometer with large-format programmable array detector. (Teledyne-Leeman).

Various molybdenum-containing sample types were used to develop the current methods and general approach to obtaining the complete compositional analysis of U-Mo alloys: (i) pure molybdenum stock and feed for U-Mo alloys, (ii) mixtures of aqueous solutions obtained from NIST-traceable standards (High-Purity Standards and CPI Standards), (iii) mixtures of aqueous solutions from uranium and molybdenum solids and (iv) uranium/molybdenum alloy samples. However, currently no U-Mo standard or control sample exists to provide direct (matrix-matched) comparison of analytical results to 'known' values.

3. Results and Discussion

Initial inductively-coupled plasma studies for trace-level impurities revolved around the use of high-resolution inductively-coupled plasma mass spectrometry (HR-ICPMS) because this technique holds the possibility of analyzing for all elements with a single technique. However, polyatomic ion interferences present using this technique proved insurmountable for the U-Mo alloys. Higher-mass peaks corresponding to $[MoO_xH_y]^+$ ions (where x=2-3 and y=1-3) interfered with the measurement of many elements in the lanthanide series. **Figure 2** shows segments of the mass spectrum corresponding to ions from the lanthanide elements in the presence of molybdenum. Of special note is the inability to resolve the isotopes of samarium from the interference ions generated from the molybdenum-oxygen-hydrogen ions generated in the plasma. Even when ions corresponding to these lanthanide isotopes could be resolved by the instrumental conditions, this mode of operation did not allow for sufficiently low detection limits and limits of quantitation to meet the current material specifications.

After determining all elemental impurities could not be analyzed by HR-ICPMS methods effectively, a combination of ICP-optical emission and quadrupole mass spectrometry techniques were applied to provide complete coverage of the periodic table. While not providing the low-level sensitivity of HR-ICPMS, the combination of ICP-OES and ICP-QMS

Figure 2: Overlays of the m/z region from 140 to 150 obtained at high resolution for a standard containing no molybdenum with analytes at 1 ng/mL (Black), a 10 ug/mL solution of molybdenum (Red) and a 10 ug/mL solution of molybdenum spiked with the analytes at 0.5 ng/mL (Blue). Unless clearly resolved, only the most abundant MoO₃H(0-3)+ species was labeled.

methods did allow for both %-level quantitation of molybdenum content and trace-level determination of elemental impurities (down to high-ppb concentration).

Initial studies involved the measurement of Mo-content by ICP-OES. Solutions prepared as above were diluted to 500 microgram-per-gram and injected into the instrument. Atomic emission lines (202.030 nm and 204.598 nm) were used to determine the molybdenum content in the diluted solution and corrected for background due to the broad atomic emission of uranium in the solution. **Figure 3** shows the measurement of molybdenum content in 36 U-Mo

alloys during 2010 – the materials were well within the tentative specifications for molybdenum (9-11%) at 95% confidence level.

Using the ICP-OES/ICP-QMS combined analyses provided the best approach to achieve full trace elemental coverage for the U-Mo fuel analysis. This approach allowed for multiple measurements of the dissolved solid solutions for selected elements by two independent techniques. **Table 1** provides the limits of quantitation (LOQ) and estimated uncertainties (U_c) for 8 selected elements (chosen due to importance in fuel fabrication). Four elements from this selective list are consistently measured above LOQ values: (i) boron between 0.2 and 1 ug/g, (ii) carbon between 100 and 400 ug/g, (iii) iron between 100 and 200 ug/g and (iv) silicon between 100 and 200 ug/g. The other (rare-earth) elements and cadmium have not been measured above their respective LOQ values in typical U-Mo samples to date.
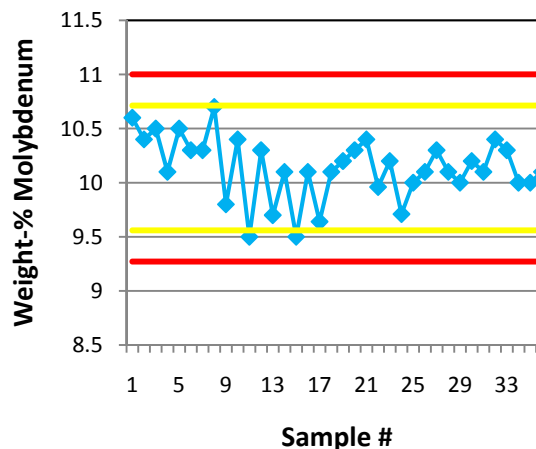
Figure 3: Molybdenum content for 36 U-Mo samples during FY2010. The yellow lines correspond to 2-sigma limits and the red line corresponds to 3-sigma limits.

4. Conclusions

While the current methodology provides an avenue to complete elemental analysis of the U-Mo fuels, significant improvement in uncertainties and measurement efficiency can be made. Additional investigations are planned in 2011 and 2012 to determine the homogeneity of uranium isotopic composition, molybdenum content and trace-element content using laser ablation mass HR-ICPMS. Currently, a reference material does not exist for this specialized fuel. Y-12 is in the process of retaining a larger sample that can be used both as a measurement control monitor for analytical results and provide future validation to methods being developed for analysis.

Table 1: Method of Analysis, Limits of Quantitation and Uncertainty for Selected Elements in U-Mo Fuels

Element	Method	LOQ (ug/g)	Uc (ug/g)
Boron	ICP-QMS	0.2	0.1
Cadmium	ICP-OES	0.5	0.1
Carbon	Leco	20	5.9
Europium	ICP-QMS	0.15	0.067
Gadolinium	ICP-QMS	0.15	0.072
Iron	ICP-OES	21	4.1
Samarium	ICP-QMS	0.05	0.018
Silicon	ICP-OES	18	4.7

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QUALIFICATION IRRADIATION OF NEW HIGH DENSITY UMO LEU FUEL PLATES AND OPERATING CONDITIONS DURING IRRADIATION AT THE BR2 HIGH FLUX MATERIALS TESTING REACTOR

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ABSTRACT

A new qualification irradiation of high-density (8gU/cc) LEU fuel plates containing different dispersed UMo fuel compositions and different cladding has been performed at the BR2 high flux material testing reactor. The irradiation of U7Mo plates was performed in the framework of the LEONIDAS programme at the BR2 reactor during 3 operating cycles inside a dedicated E-FUTURE basket. Irradiation conditions in U7Mo plates have been optimised in order to be equivalent to the conditions at which HEU fuel plates are operating in high flux research reactors. Special attention during preparing irradiation programme and choosing operating conditions at the BR2 was given to the position inside the BR2 core which provides equivalent conditions for all U7Mo tested plates. Optimization and determination of irradiation conditions in U7Mo plates for all irradiation cycles were processed using a BR2 computer model with the help of MCNP Monte Carlo transport code. Credibility of the model has been verified on available dosimetry and thermal balance measurements in different experiments during BR2 irradiation cycles. Three-dimensional calculation model of BR2 permits to simulate simultaneously spatial distributions of heat fluxes and the fuel burn-up distribution in all plates of the full-scale prototype of LEU fuel during several successive irradiation cycles of BR2. The model is using the prediction-correction approach for calculating an evolution of 3-D burn-up distributions in fuel plates and includes simulations of neutron transport and the fuel burn-up using MCNP Monte Carlo transport and SCALE burn-up codes. It is anticipated that the maximum thermal heat flux on the cooling surface of the irradiated U7MO plates during irradiation is comparable with operating condition for existing HEU fuel plates. An evaluation of the maximum burn-up of the [235]U reached in plates during irradiation was performed using the present simulation model.

1.   Introduction

The development of high-density LEU fuels has made important progress in the last years. In particular, the qualification of the dispersed UMo fuel system with densities up to ~ 7.5 - 8.5 gUtot/cm³ is making substantial progress although irradiation tests at very high heat-fluxes still are to be performed.

BR2 is actively involved in the qualification process of high-density (~8 gUtot/cc) LEU fuel based on dispersed UMo.
A European group (the so-called LEONIDAS group) for the qualification of high density LEU fuel based on dispersed UMo has been formed early 2009 and is launching an experimental program (irradiations & PIE) of common interest in close collaboration with the US GTRI/Convert program.

The first irradiation concerns 4 full-size flat high density U7Mo plates containing Al+Si matrix with ^{235}U enrichment of 19.7%. The experimental plates have been manufactured by AREVA-CERCA. After commissioning tests, including destructive tests and cutting of some plates, 7 plates were still available for irradiation. The final choice of the plates to be irradiated was made during the final commissioning of the manufactured plates at the Romans plant of CERCA (see Table 1).

Plate label	U7Mo, g	U_{tot}, g	^{235}U, g	Matrix Al+Si, g	Si, %	T, ^{o}C	Cladding
U7MC6111	151.90	140.36	27.59	22.46	6	425 - 2h	AlFeNi
U7MC6301	151.85	140.31	27.58	22.46	6	475 - 4h	AG3
U7MC4111	152.14	140.88	27.55	22.51	4	425 - 2h	AlFeNi
U7MC4202	152.27	140.00	27.57	22.53	4	475 - 2h	AG3

Tab 1: Specifications of U7Mo plates chosen for irradiation

The irradiation conditions to be fulfilled for these plates at the BR2 are the following:
- the irradiation conditions of all 4 plates should be as similar as possible as the outcome of this irradiation should be used for the selection of one set of fuel plate manufacture parameters (%Si, heat treatment),
- at BOC of the first irradiation cycle a maximum heat flux of 470 W/cm² should be obtained on at least one plate, while the minimum value of 450 W/cm² should be obtained for all plates,
- the irradiation scenario for the following two or three cycles will be fixed when the results of the first irradiation cycle are known (wet-sipping, visual inspection),
- the plates should be irradiated up to a mean burn-up of at least 50% (with a maximum value of at least 80%).

2. Characteristics of U7Mo plates and of E-Future basket

The E-FUTURE fuel plates are placed in the dedicated E-FUTURE irradiation basket (see Fig.1). A distance between fuel plates in the basket is equal to 6.4 mm. The central holder plate in the basket is designed to receive dosimeter wires for measurements of thermal and fast neutron fluxes.
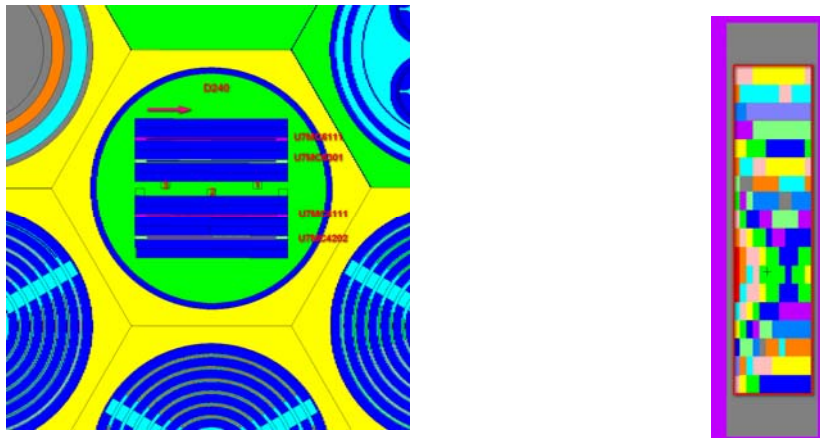
Fig.1 E-FUTURE basket with four U7Mo fuel plates in BR2 channel (left figure) and an example of adaptive material mesh for simulating fuel burn-up in UMo plates(on the right figure)

Geometrical specifications for U7Mo fuel plates	
Mean width of fuel zone, mm	45
Mean length of fuel zone, mm	762
Mean thickness of fuel zone, mm	0.51
Fuel plate thickness, mm	1.27
Water gap between plates and the basket, mm	6.4
E-Future basket material	AlMgSi1

Tab 2. Geometrical specifications of U7Mo fuel plates and E-Future basket

3. Irradiation of U7Mo fuel plates at BR2
3.1 Provisional power density in the hot spot

The choice of the E-FUTURE irradiation channel in BR2 is very much influenced by the progress of the EVITA irradiation programme in the central 200 mm flux-trap channel and depends on type of the plug loaded in this channel. A provisional simulation of the not-confirmed BR2 reactor core loads for cycle 03/2010 was performed in order to find an optimal position of the E-FUTURE-E basket (see Fig.2). The confirmed load of a BR2 cycle is a load realized in the core that will effectively be operated (see Fig.2). The best irradiation conditions for all 4 E-FUTURE plates were obtained for an appropriate orientation of the E-Future basket in channel D-240 at a BR2 operating power of 59 MW. At this reactor power the maximum heat flux in all 4 plates is equal to 460-470Wcm^2. The mean heat flux over the width of U7Mo plates at hot plane level is about 380 W/cm^2 in all 4 plates.
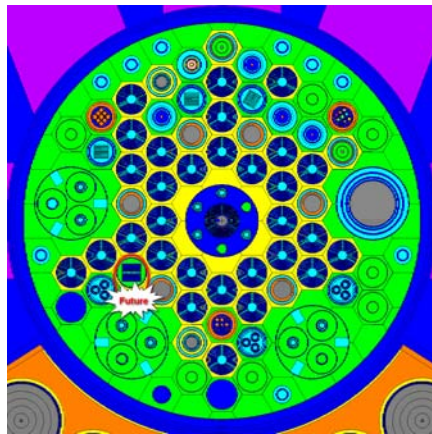
Fig.2 E-FUTURE basket with four U7Mo fuel plates in BR2 core

3.2 Provisional cladding temperature for plates irradiated in the E-Future basket

The initial design of the basket was made for a peak heat flux of 500 W/cm². With the foreseen flow rate of 30.9 l/s (corresponding to a water speed between the experimental plates of ~14.2 m/s) the maximum cladding temperature was evaluated to be 167 °C. This evaluation was made with a 3D mesh model, programmed by the assigned project engineer on basis of the FEM methodology.
Test measurements in the out-of-pile test-loop H4 (which simulates a standard BR2 reactor channel) have shown that the actual flow through the E-FUTURE basket (for the nominal BR2 hydraulic conditions) was only 25.4 l/s (corresponding to a water speed between the experimental plates of ~11.4 m/s). The maximum cladding temperature was again investigated for a maximum heat flux of 470 W/cm²: the 3D model gave a value of 182.5 °C.

This still leaves some margin to the ONB limit of 189.5 °C. At the same time, another calculation with a 1D model based on the basic heat transfer theory yielded 165°C.

A third evaluation with the PLTEMP code (using the Dittus-Boelter correlation) also gave 165 °C. When using the Sieder-Tate correlation a value of only 142 °C. All these temperature evaluations don't take into account the influence of an oxide-layer formation on the outer surfaces of the plates.

Most evaluations have been made for 470 W/cm² as this is the maximum allowed heat flux for the standard BR2 driver fuel elements. Let us note that at BOC of the first irradiation cycle of the E-FUTURE basket (cycle 04/2010A.4**)** the maximum heat flux on the hottest experimental plate was evaluated ~ 470 W/cm².

3.3 Actual power history of BR2 during irradiation

The first LEONIDAS irradiation programme (the 4 selected LEU U7Mo plates) using the E-FUTURE basket has been started at the BR2 cycle 03/10 (June 2010) and continued during 3 cycles. The history of BR2 power variation during irradiation is shown in Fig.3.
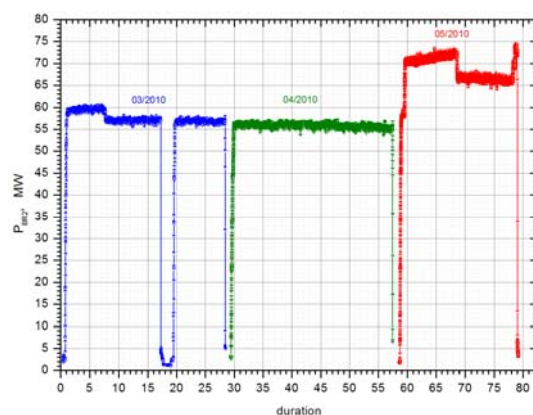
Fig 3. Operating power of BR2 during irradiation of U7Mo plates

Computer model, used to determine irradiation conditions in the E-FUTURE basket in cycle 04/2010, includes neutronic simulation of neutron transport in the whole BR2 core, simulation of fuel burn-up in fuel elements and in all experimental devices, simulation of Be poisoning and movement of controls rods. Calculations of the fuel burn-up in U7Mo plates during irradiation were performed on a uniform adaptive material mesh inside the fuel zones of U7Mo fuel plates. The dimension of each mesh cell is equal to 3.75×42.3mm, i.e. each plate contains 216 fuel zones with own distribution of fuel compositions, which are dependent on the calculated distribution of ^{235}U burn-up. An example of the material mesh, used in the fuel zone of one U7Mo plate, is presented in Fig.1. Each of four U7Mo plates is processed with independent 2D distributions of ^{235}U burn-up. Calculations were performed using a prediction-correction approach with the help of MCNP-4c [1], and SCALE-4.4a [2] codes and applying additional post-processing algorithm to perform fuel burn-up calculation on material meshes.

After the visual inspection of plates, and before the cycle 05/2010, an orientation of one U7Mo plate in the basket was changed (plate #6111 was rotated by 180° relative z-axis), while the rest three plates kept their positions in the basket during all three irradiation cycles. This modification in the plate position was introduced into the computer simulation model for the last irradiation cycle. Estimated variations of power density in the hot spot for each plate during irradiation are presented in Table 3. Mean burn-up of ^{235}U at the end of irradiation in plates is approximately equal in all U7Mo plates (see Table4) and is about 47-48%, while in the hot spot the calculated maximum of the ^{235}U burn-up is about 67-72%.

An example of calculated two-dimensional distributions of the power density on the cooling surface of fuel plates and distributions of ^{235}U burn-up are presented in Figs. 4-5 for selected irradiation moments.

Irradiation performance graph (Fig.6) presents an evolution of the heat flux in the hot spot versus the fuel burn-up in the hot spot during irradiation. The plates 6301, 4111, and 4202 were irradiated with gradually decreasing heat fluxes in the hot sport during increasing the fuel burn-up. Because of the plate 6111 rotation in cycle 5/10, the irradiation performance graph for this plate has a second maximum at the fuel burn-up equal to 50% (see left Fig.6).

Plate #	Cycle 3/10			Cycle 4/10			Cycle 5/10		
	BOC	7d	EOC	BOC	7d	EOC	BOC	10d	EOC
6111	465	436	336	333	320	272	450	320	272
6301	472	425	337	336	325	267	318	288	259
4111	457	429	329	336	320	263	309	286	254
4202	453	424	328	324	316	263	305	278	249

Tab 3. Hot spot (W/cm^2) in U7Mo plates during irradiation

Plate	Cycle 3/10	Cycle 4/10	Cycle 5/10
U7MC6111	~18.4%	~34.8%	~46.9%
U7MC6301	~18.4%	~35.1%	~47.2%
U7MC4111	~18.9%	~35.7%	~47.8%
U7MC4202	~18.9%	~35.7%	~47.8%

Tab 4. Mean burn-up of ^{235}U, $^5\beta_{mean}$, reached in U7Mo plates during irradiation.

Plate 6301   plate 6301   plate 6111   plate 6301
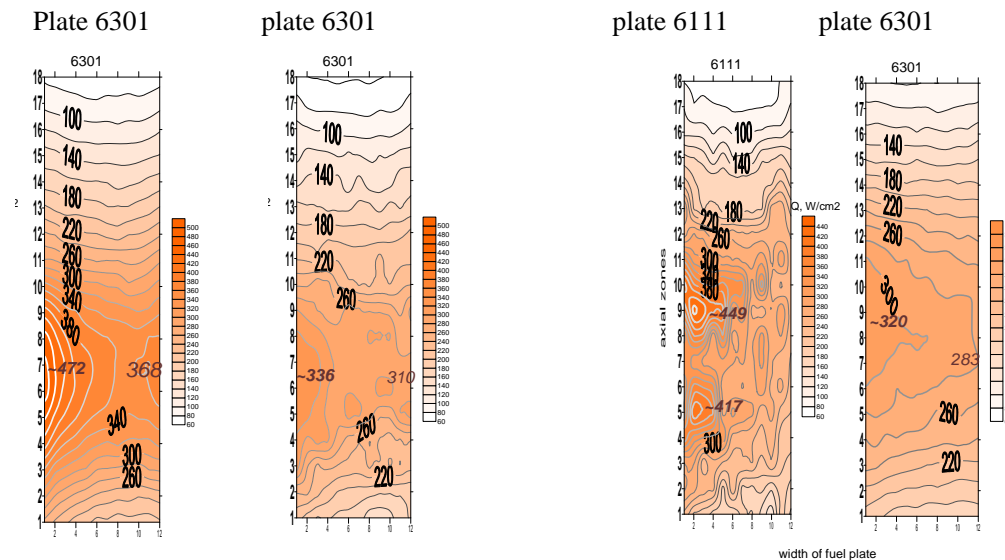
Fig 4. Variation of 2D distribution of power density distribution on cooling surface of the plate 6301 at BOC of cycles 3/4/5. The plate 6111 was rotated by 180 degree before cycle 5/10

Plate 6301	plate 6301	plate 6111	plate 6301
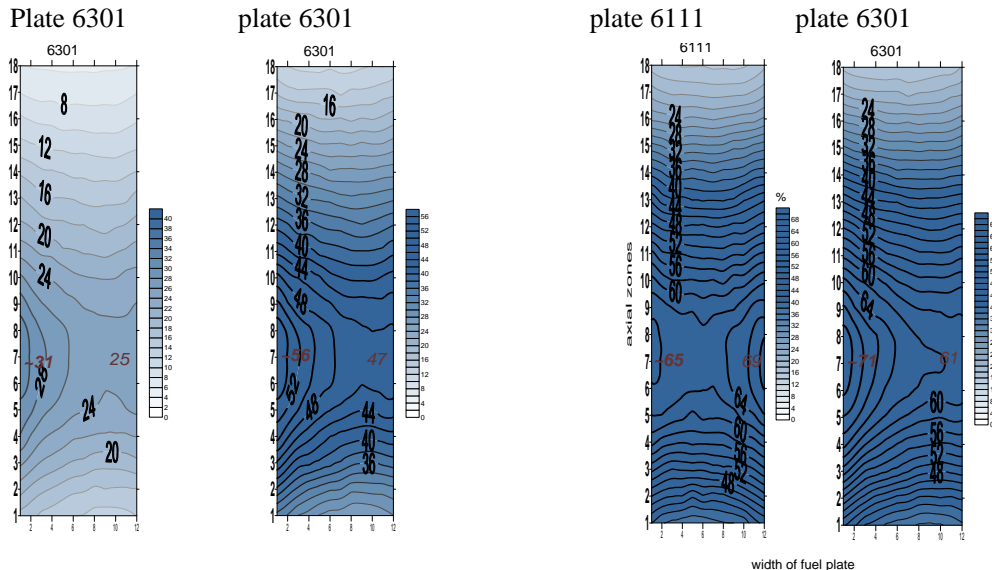
EOC 3/10   EOC 4/10   EOC 5/10   EOC 5/10

Fig 5. Calculated 2D distribution of ^{235}U burn-up in the plate 6301 at EOC after each irradiation cycle. The plate 6111 was rotated by 180 degree in cycle 5/10
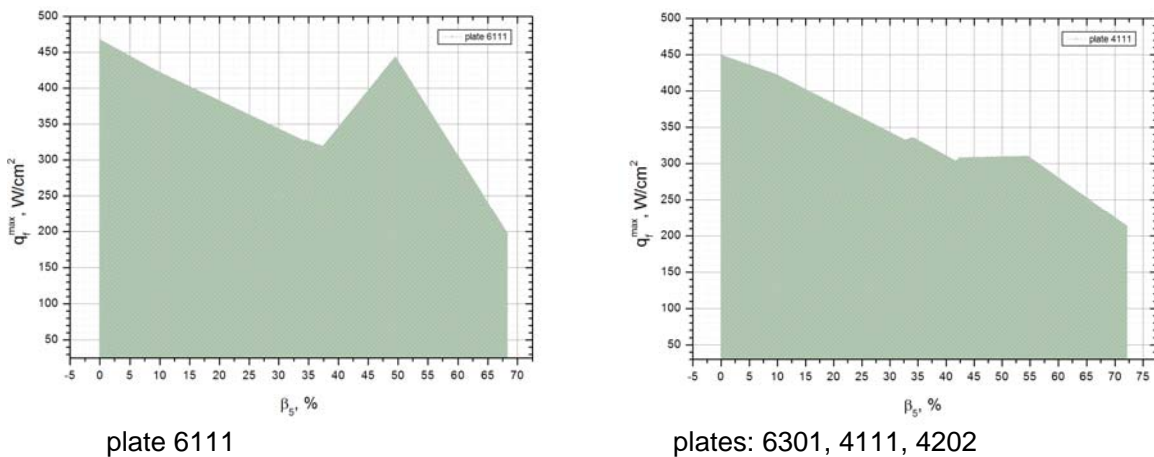
      plate 6111   plates: 6301, 4111, 4202

Fig 6. Irradiation performance graph for the maximum power density versus the fuel burn-up in the hot spot of U7Mo plate (left figure - the plate 6111, right figure – plates 6301, 4202, and 4111)

4.   Summary

Irradiation of 4 U7Mo high density fuel plates with Al+Si matrix were performed at the BR2 High Flux Material Testing Reactor during 3 irradiation cycles in the framework of LEONIDAS programme at the conditions comparable with the conditions for BR2 HEU fuel plates. No release of fission products was detected during irradiation. Visual inspection after irradiation revealed no apparent damages on the surface of fuel plates. The maximum hot spot of heat flux in plates was estimated using computer simulation model: $q_{max} \approx 450\text{-}470$ W/cm^2. The mean burn-up of ^{235}U in plates as estimated is about 47-48%, while the maximum burn-up is about 67-72%.

5.   References
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The Y-12 National Security Complex Foreign Research Reactor Uranium Supply Production

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Abstract

The Foreign Research Reactor (FRR) Uranium Supply Program at the Y-12 National Security Complex supports the nonproliferation objectives of the National Nuclear Security Administration (NNSA) HEU Disposition, the Reduced Enrichment Research and Test Reactors (RERTR), and the United States (U.S.) FRR Spent Nuclear Fuel (SNF) Acceptance Programs. The FRR Supply Program supports the important U.S. government nuclear nonproliferation commitment to serve as a reliable and cost-effective uranium supplier for those foreign research reactors that are converting or have converted to Low-Enriched Uranium (LEU) fuel under the RERTR Program. The NNSA Y-12 Site Office maintains the prime contracts with foreign government agencies for the supply of LEU for their research reactors. The LEU is produced by down blending Highly Enriched Uranium (HEU) that has been declared surplus to the U.S. national defense needs. The down blending and sale of the LEU supports the Surplus HEU Disposition Program Record of Decision to make the HEU non-weapons usable and to recover the economic value of the uranium to the extent feasible. In addition to uranium metal feedstock for fuel fabrication, Y-12 can produce LEU in different forms to support new fuel development or target fabrication for medical isotope production. With production improvements and efficient delivery preparations, Y-12 continues to successfully support the global research reactor community.

Y-12 Foreign Research Reactor Supply Program Overview

Y-12 supplies foreign research reactors with low enriched uranium (LEU) at 19.75 wt. % ^{235}U under the Foreign Research Reactor (FRR) Uranium Supply Program, primarily in the form of uranium metal. For small-scale research and development, Y-12 has the capability to provide various forms and enrichments of LEU based on research

reactor requirements. The LEU is produced at Y-12 by down blending surplus U.S.-origin HEU. In 1995, approximately 174 metric tons (MT) of highly enriched uranium (HEU) were declared surplus to the national security needs by the President of the United States. A commitment was made by the U.S. to permanently remove this material from the U.S. defense stockpile and to use it for peaceful uses to the extent possible. In 2005, an additional 200 MT were declared excess to national defense purposes. Between the two surplus declarations, approximately 10 MT HEU have been designated for disposition to research and test reactor fuel and targets for medical isotope production through at least 2028.

The down blending and sale of the LEU for FRR fuel or targets for medical isotope production supports the Record of Decision for the Surplus HEU Disposition Program to make the weapons non-weapons usable and to recover the economic value of the uranium to the extent feasible. As of the end of January 2011, over 134 MT of surplus HEU have been down blended and approximately 3.7 MT have been down blended at Y-12 for research reactor fuel and target feedstock.

The FRR uranium supply program supports the important U.S. government nuclear nonproliferation commitment to serve as a reliable and cost-effective supplier of feed material for those foreign research reactors that are converting or have converted to LEU fuel under the guidance of the NNSA Reduced Enrichment for Research and Test Reactors (RERTR). The Y-12 NNSA Site Office is authorized to administer the FRR uranium supply contracts with foreign governments in accordance with Section 54a of the Atomic Energy Act of 1954, as amended, and Section 3112 (d) and (e) of the United States Enrichment Corporation (USEC) Privatization Act of 1996. DOE NNSA is authorized to distribute special nuclear material to countries that have entered into an Agreement for Cooperation with the U.S. Government concerning peaceful uses of nuclear energy and that DOE may sell enriched uranium to "any State or local agency or nonprofit, charitable, or educational institution for use other than the generation of electricity for commercial use." In addition, DOE may sell LEU to commercial entities as long as the material is not necessary for national security needs; that the sale will not have an adverse impact on the domestic uranium industry and the price is not less than the fair market value of the material.

In support of the NNSA Office of Global Threat Reduction and the FRR Spent Nuclear Fuel (SNF) Acceptance Program goal for the safe, secure removal of U.S.–origin HEU from foreign research reactors, NNSA often negotiates the removal of HEU by offering an equivalent LEU credit based on the net value of the material to be returned. The FRR can apply the LEU credit to an order under an LEU supply contract with NNSA Y-12.

LEU Production Process

2

The LEU demand for foreign research and isotope production reactors is approximately 1,500 to 2,000 kilograms per year and is expected to increase as reactors convert from HEU to LEU. The demand is expected to increase significantly beyond 2014-15 when the high flux HEU research reactors are targeted for conversion (potentially a 2 to 5 times increase in LEU demand). How does Y-12 maintain an LEU inventory to meet this demand?

First, Y-12 forecasts the yearly demand by current and potential FRR customer and projects the deliveries by month. Based on these quantities and timeframes, the LEU production schedule is established for the fiscal year. Figure 1 is an example of the total inventory, production castings and the forecasted delivery quantities. The bars illustrate the various FRR customers and quantities.
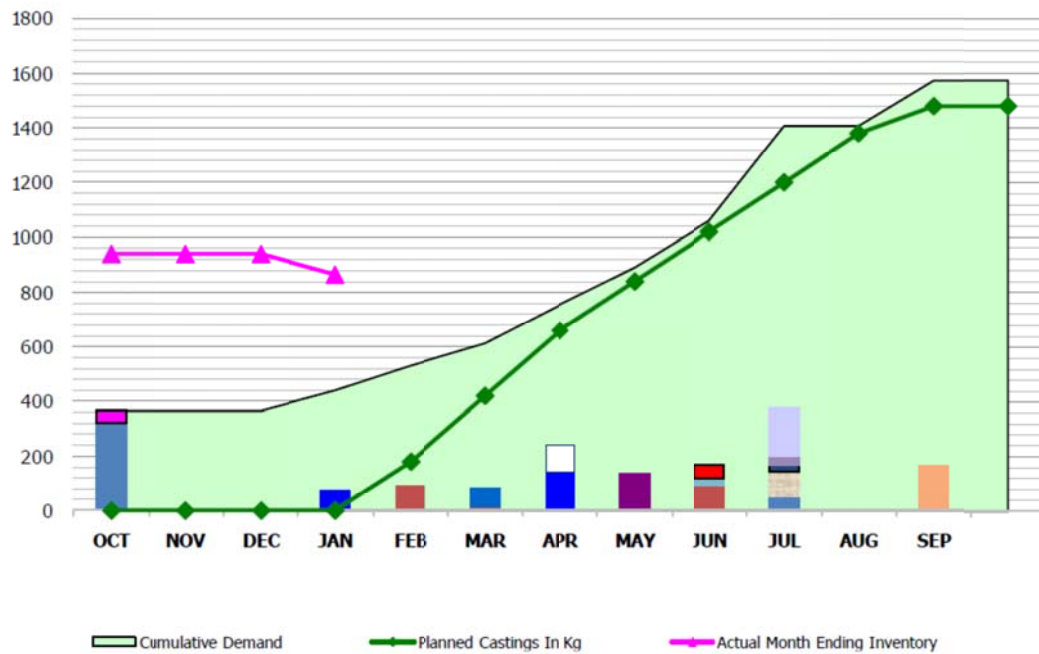
Figure 1. LEU Production and Inventory in Support of Research Reactor Supply

The normal form of uranium produced at Y-12 is as broken metal. Y-12 employs a molten metal casting process to down blend the surplus HEU with either depleted or natural uranium to nominally 19.75 weight percent ^{235}U. The HEU items are selected based on the chemical characterization and availability. The feed materials are

melted in a vacuum induction furnace and cast into a right annular cylinder (or hollow log), which has a critically safe geometry. The batch sizes range between 18 and 20 kgs U.

Samples are drilled from the hollow log and the samples are analyzed to ensure enrichment, uranium isotopic composition and impurities meet the material specifications for each batch. One sample per casting batch can be provided to the customer with the bulk metal. Depending on the shipping container, the sample is wrapped in aluminum foil or placed in an inerted sample bottle. The sample is then placed inside one of the inner cans with the bulk metal.

The hollow logs are broken in a hydraulic press, and then sheared to make broken metal pieces ranging in size from 80 to 300 grams. The broken metal is loaded into carbon steel or stainless steel cans with press-fit lids under an argon atmosphere. The cans are 4¼ inches (10.8 cm) by either 4¾ (12.1 cm), 8¾ (22.2 cm) or 10 (25.4 cm) inches tall and are lined with either aluminum or carbon steel mesh to minimize damage to the cans during transport.

When a customer's order is placed, the cans are loaded into the selected shipping container certified for international transport of fissile material. A Mylar tamper indicating device is applied to the cans and/or shipping container. The containers are then staged for shipment.

In 2010, a certificate of compliance was issued for the U.S. DOE-designed ES-3100 Type-B shipping container for air transport of unirradiated bulk uranium metal, certain uranium compounds, and specific fuel elements. Having the empty containers available at Y-12 allows the LEU to be packaged in a more efficient manner. Loading of the shipping containers can be campaigned in the production work schedule to maximize use of resources. The loaded ES-3100s can then be stored in the new enriched uranium storage facility at Y-12 well in advance of the scheduled delivery. The containers can be staged for shipment whenever the transport agent arranges for pick up from Y-12 and subsequent delivery to the fuel and/or target fabricator.

LEU Quality

The origin of the material in the production of LEU for the research reactor community is the major contributor to the quality of the LEU product. At Y-12, the LEU is produced by down blending weapons-grade HEU material with a carefully selected diluent. Reprocessed material is usually less suitable due to the minor uranium isotope concentrations and the processing required to remove the impurities.

4

Efforts have been made in the past by research reactor suppliers to agree upon a worldwide unified technical specification for LEU. An American Standards and Test Materials (ASTM) standard specification (ASTM specification C-1462-00) was developed in order to facilitate supplies of LEU for fabrication of research reactor fuel elements. However, the effort to develop one specification that met the different organizational needs created a specification that is not acceptable to many research reactor customers. For example, C-1462-00 has higher [234]U, [236]U, and transuranic element limits to allow for the use of reprocessed uranium, which is not acceptable to some of the fuel and target fabricators and reactor users.

Consequently, Y-12 developed a standard specification for LEU based on its understanding of the material quality and the requirements of the various FRR customers. By producing LEU that meets a standard specification, Y-12 is able to maintain an inventory of LEU metal in support of current and future NNSA uranium supply contracts.

All the limits (except for dysprosium) in the Y-12 LEU Standard Metal Specifications for Research and Test Reactors are equal to or less than the ASTM-C-1462-00 specification limits. Table 1 shows a comparison of the specification limits for several of the parameters between the Y-12 LEU metal produced from down blended HEU and the ASTM-C-1462-00 standard specification limits.

For the [99]Mo targets for medical isotope production, the tungsten (W) levels need to be very low in the feed material as the target manufacturing process often introduces W which affects the performance of the target. An analysis of all LEU batches (over 400), excluding the LEU/Molybdenum (Mo) batches, produced at Y-12 since 2002 shows W levels with a mean 3.526 ug/gU even though the Y-12 standard and ASTM specification for W is 100 ug/gU.

Table 1. Comparison of Y-12 Specification to ASTM-C-1462-00

Element	Symbol	Units	Y-12 LEU Metal Y/GNSS-05-02 r2	ASTM C1462-00
Uranium	U	wt %	99.88%	99.85%
U-232	U-232	µg/gU	0.002	0.002
U-234	U-234	wt %	0.26%	1.00%
U-235	U-235	wt %	19.75%	19.75%
U-236	U-236	µg/gU	4,600	40,000
Trans-U (Alpha)	TRU	Bq/gU	100	250

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Activation Products	ActProd	Bq/gU	100	
Fission Products	Gamma	Bq/gU	600	600
Carbon	C	µg/gU	350	800
Cobalt	Co	µg/gU	5	10
Dysprosium	Dy	µg/gU	5	Sum < 3
Europium	Eu	µg/gU	2	Sum < 3
Gadolinium	Gd	µg/gU	1	Sum < 3
Lead	Pb	µg/gU	5	10
Lithium	Li	µg/gU	2	10
Manganese	Mn	µg/gU	24	50
Phosphorus	P	µg/gU	50	100
Samarium	Sm	µg/gU	2	Sum < 3
Silicon	Si	µg/gU	100	250
Total Impurities	TotImp	µg/gU	1,200	1,500
Equivalent Boron Content			3	4

Standardization has enabled Y-12 to respond quicker to FRR customer requests by maintaining an LEU inventory that meets a standard specification and it has simplified production requirements and quality control checks which have improved Y-12's efficiency to prepare LEU orders for delivery.

Y-12 continues to evaluate ways to standardize the uranium metal form that is provided to its FRR customers. The current form is broken metal with irregular shaped pieces. One objective is to cast material into small, regular shapes at a more uniform mass that would meet a customer's equipment and process requirements. Several mold designs were tested on surrogate material and one has been accepted that produces "gumdrop" shaped pieces of LEU at ~ 180 kg U. Samples of the test LEU castings are currently being analyzed to determine homogeneity of the down blending and casting process. A standard form will also greatly optimize production efficiency by reducing material handling and packaging requirements.

Y-12 is actively involved in the development of new LEU fuels in support of the RERTR Program in order for reactors to convert from HEU to LEU fuel with developing and validating a production oriented, monolithic uranium molybdenum (U-Mo) foil fabrication process. Between 2006 and 2010, Y-12 developed and validated its foil fabrication process by producing multiple U-Mo foils and coupons to meet design specifications. Also, Y-12 is currently evaluating the re-implementation of its metal powder atomization and oxide processes for production of research and development scale quantities for the development of new LEU fuels and targets.

Summary

The U.S. DOE NNSA and Y-12 is honored to support the global research reactor community. The Y-12 production process has resulted in a reliable and cost-effective FRR uranium supply program. The Y-12 supply of high-quality LEU is essential to the present and future successful operation of the world's research reactors.

NEW EXPERIMETAL INSTRUMENTATION FOR STUDY OF REACTOR DYNAMICS AT ZERO POWER REACTOR VR-1

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ABSTRACT

The paper presents new experimental device developed for a study of the reactor dynamics at zero power reactor VR-1 and the first experiences of its utilisation.

VR-1 reactor has been operating by Czech Technical University (CTU). It is a pool-type light water reactor based on enriched uranium with maximum thermal power 1kW and for short time period up to 5kW. The reactor has been utilised very efficiently particularly for education and training of university students and NPP's specialists more than 20 years.

The improvement of the reactor equipment and experimental instrumentation is necessary background for keeping the reactor attractive for students. The example of such way is the installation of the device for dynamics study in the VR-1 reactor which was developed in a close cooperation between CTU and ŠKODA JS Company. The device is based on a pneumatic drive which allows carrying out fast movement of absorbing or fissionable specimens. In this way the fast reactivity changes are caused in the reactor core. The specimen is located in a dry aluminium channel (with outer diameter 14 mm). Channel dimension allows its placing into various core positions. A user can select one of the three predefined modes for specimen movements: step, pulse or cycle.

The step mode represents the specimen transfer from the start position to the upper/lower position and stay there within predefined time. The pulse mode allows the specimen transfer either from the start position to the upper/lower position and its prompt return to the start position. The cycle mode represents a periodical movement of sample around the start position. One cycle is defined as the specimen movement from the start position to the upper/lower position and then to the lower/upper position and back to the start position. Therefore it is possible to study the reactor response to wide range of reactivity perturbations and observe its dynamics behaviour, e.g. pulse or transient characteristic. Furthermore the device can be used also as an experimental tool, e.g. for determination of β_{eff}.

The new experimental instrumentation for study of reactor dynamics has been included into the education at the reactor at the end 2009. The first experiences of its utilisation show that it brings new potential in the educational process and improve quality of training at the reactor VR-1.

1. Introduction

The training reactor VR-1 was commissioned in 1990 at the Czech Technical University in Prague (CTU). The reactor is a pool-type light-water reactor based on low enriched uranium with maximum thermal power 1kW and for short time period up to 5kW [1]. The moderator of neutrons is light demineralised water, which is also used as a reflector, a biological shielding, and a coolant. Heat is removed from the core by natural convection. The pool disposition of the reactor facilitates access to the core, setting and removing of various experimental samples and detectors, easy and safe handling of fuel assemblies. The reactor core contains 15 to 21 fuel assemblies, depending on the geometric arrangement and kind of experiments to be performed in the reactor. The core is accommodated in a cylindrical stainless steel vessel - pool, which is filled with water. The reactor is used mainly for education of technical university students, training of specialists in the nuclear industry, and finally in promotional activities within the field of nuclear power.

CTU as a reactor licensee makes permanent effort to improve the reactor equipment, experimental instrumentation and to develop the educational and training methodologies. It is way how to provide education and training at very effective level and keep the reactor attractive for the students [2]. The example of such improvement is new device for study of reactor dynamics which has been installed at the VR-1 reactor. The device was designed and developed in a close cooperation between CTU and ŠKODA JS Company.

2. Device for Study of Reactor Dynamics in VR-1 Reactor

The device for study of reactor dynamics is based on a pneumatic drive which allows carrying out fast movement of absorbing or fissionable specimens. In this way the fast reactivity changes are caused in the reactor core. Therefore it is possible to study basic reactor dynamics characteristics such as reactor response to the pulse, step and periodical reactivity changes.

The device consists of:
- control unit,
- pneumatic drive,
- guide rod,
- case for specimen,
- dry channel.

The drive contains a microprocessor-controlled pneumatic valve which control inlet of pressure air going under and over pneumatic drive plunger (piston). The system ensures up-and-down movement of specimens which is well defined and controlled by the control unit. A specimen is closed in the aluminium case which is easily demountable through the bayonet joint. From Fig 1 follows maximal dimension of specimen which can be inserted in the aluminium case. Cadmium sheet is used at the VR-1 reactor but the case can be filled also by other materials (e.g. boron carbide or uranium). The case is attached to the guide rod and it is connected to the pneumatic drive (see Fig 1).
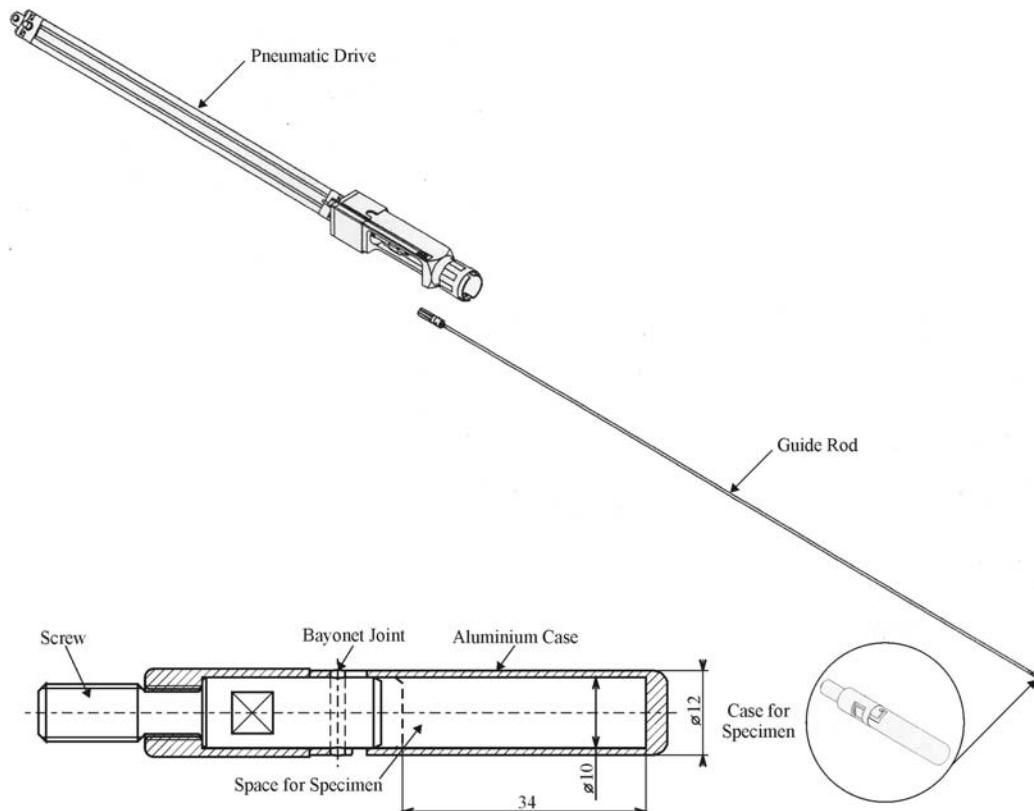

Fig 1. The device for dynamics study - main components [1]

The guide rod with case containing specimens is located in a dry aluminium channel (with outer diameter 14 mm). Channel dimension allows its placing into an arbitrary position of reactor core (e.g. fuel assembly or fuel dummy). The specimen may move through the entire core height: i.e. from the bottom support plate to the core top (in total 750 mm).

The specimen movement is fully controlled by a control unit. The unit contains a programmable logic controller for definition of movement modes, setting the movement parameters and controls the pneumatic drive. A user can select one of the three predefined modes for specimen movement:

- step,
- pulse,
- cycle.

The step mode can define and perform the specimen transfer from the start position to the upper/lower position and to remain there. The pulse mode enables to define transfer of the specimen either from the start position to the upper/lower position and its prompt return to the start position. The cycle mode provides a periodical movement of sample around the start position. The cycle is defined as the specimen movement from the start position to the upper/lower position and then to the lower/upper position and back to the start position. The following parameters of specimen movement can be set up within the modes:

- maximal movement velocity, the range is from 10 to 1000 $mm.s^{-1}$
- the start movement point, the range is from 10 to 740 mm
- movement range from start point to lower position
- movement range from start point to upper position
- delay time in lower position, the range is from 0 to 999 s
- delay time in upper position, the range is from 0 to 999 s
- number of cycles for periodical movement, the range is 0 to 99 cycles.

Fig 2 shows schematic disposition of the device for dynamics study in the VR-1 reactor and its connection to the neutron flux monitoring system. One plastic tube

Fig 2. Schematic disposition and connection of the device for dynamics study in the VR-1 reactor

distributes pressure air from a pressure air supply system through filters (removing potential impurities from pressure air) to the pneumatic drive. Two lines ensure communication between control unit and pneumatic drive. The first line connects control unit with pneumatic drive control. Information about specimen position is sent through the second line. Control unit can send specimen position data to the neutron flux monitoring system as well. The data are sent as a TTL signal of frequency from 0 to 40 kHz which is proportional to the pneumatic piston position.

The neutron flux monitoring system consists of a low current monitor TEMA LCM310 [3] and compensated ionisation chamber. TEMA LCM310 is a compact apparatus which is used for
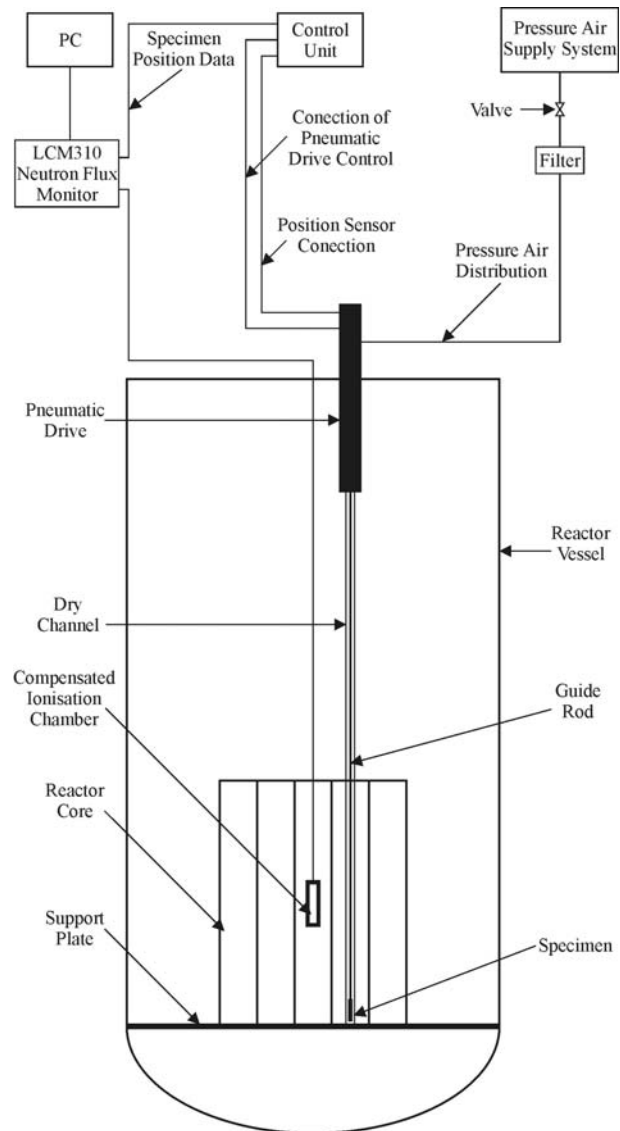
a processing of a current signal from neutron detectors working in current mode. The minimal dwell time of the apparatus is 1 ms which sufficiently allows register the reactor response to the fast reactivity changes. TEMA LCM310 is connected by coax cable to control unit of device for dynamics study and received TTL signal about the specimen position. It allows synchronized measurement of neutron flux and specimen position.

3.   Testing and first experiences

Device for study of reactor dynamics was designed at CTU in the year 2007 and it was produces by SKODA JS Company in the year 2008. Device was installed on December 2008 and first preliminary tests, mainly manipulation tests, were performed soon after delivery. Later on, during spring semester, complex testing of the device was carried out. Educational and training methodologies were developed in the fall semester in 2009 and device was fully incorporated to the education process at VR-1 reactor in the beginning of the year 2010.

The tests confirmed that the suitable choice of specimen allows device operation in all modes with no influence on reactor safety system. The optimal specimen is a ring with diameter of 5 mm and length of 10 mm, made from a thin (1 mm) cadmium sheet. Such specimen causes a low but well measurable reactivity changes which allows very fast movement of the specimen (e.g very fast reactivity changes) without activating the safety signals at the reactor control and safety system. The device was connected with system for neutron flux monitoring and tested.

More than one year experiences with the device utilisation show that it brings big benefit to experimental education and training. It allows the students compare the theory of zero power reactor dynamics and its basic fundamentals with experiments. Fig 3 and Fig 4 show reactor response to the step and periodic reactivity changes carried out by device for dynamics study and measured by TEMA LCM310 monitoring system.

Furthermore the device is used as an experimental tool for determination of effective delayed neutron fraction (β_{eff}) also. The experimental method has been prepared and verified in close cooperation with Slovak Technical University in Bratislava. The method is based on an in-pile kinetic technique applied to zero-power reactor response to periodical low-worth reactivity changes. In such way the experimental value of β_{eff} for VR-1 reactor was determined for the first time and corresponds to calculated value from MCNP analyses [4].
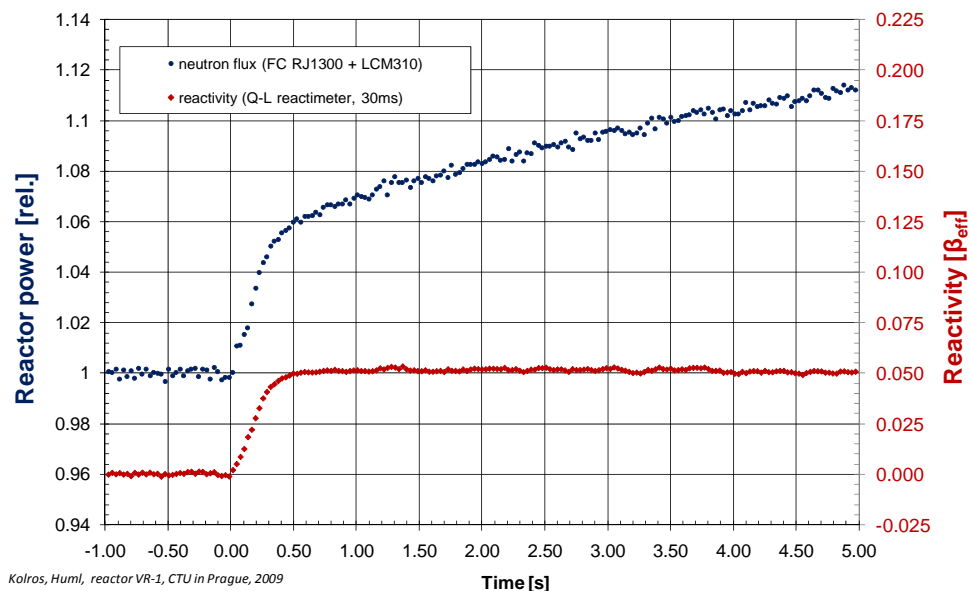

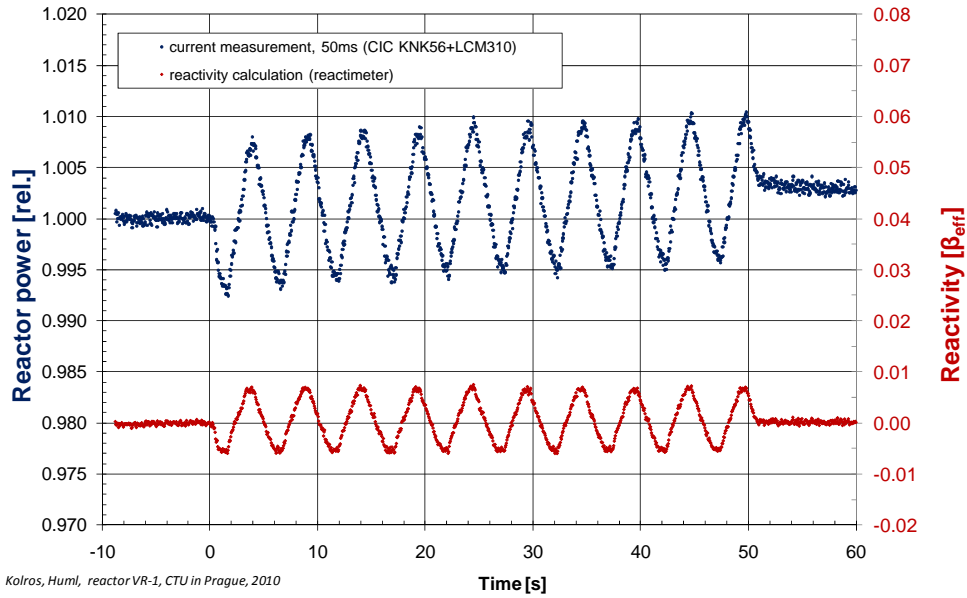Fig. 3. The response of the VR-1 reactor to the step reactivity perturbation [1]

Kolros, Huml, reactor VR-1, CTU in Prague, 2010

Fig 4. The response of the VR-1 reactor to the periodical reactivity oscillation [1]

4. Conclusions

Czech Technical University in Prague and SKODA JS Company have designed and developed new experimental instrumentation for study of reactor dynamics. Complete instrumentation was included into the education at the reactor in the beginning of the year 2010. The first year experiences confirmed that the device is suitable for effective education and training and help to keep the reactor attractive for the students. The instrumentation allows precisely define and carry out step, pulse or periodical reactivity changes and monitor the reactor response to them. It helps to students to better understand the principles of zero power reactor dynamics and to verify related theory.

The instrumentation has become an important experimental tool for education and training at VR-1 reactor. Furthermore it extended the experimental possibilities of the reactor.

5. References

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"UPGRADING OF THE HALDEN REACTOR EXPERIMENTAL CAPABILITIES FOR INNOVATIVE FUEL AND MATERIAL TESTING"

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Abstract

The resurgence of world-wide interest for utilization of nuclear materials to produce energy in the last decades, and common problems in the area of nuclear fuel and material developments, are motivating the establishment of joint research efforts. The OECD Halden Reactor Project (HRP) is a good example of such a cooperative effort. The Halden Boiling Water Reactor (HBWR) at the heart of the HRP is able to simulate in-core conditions of modern commercial power reactors. The experimental capabilities of the Halden Reactor enable the HRP to be a crucial test facility for nuclear fuels and materials development for many years. Long-term international cooperation at the HRP is based on a flexible organizational structure and ensures the Project's continuing success.

This paper describes the experience of the Halden reactor utilization and a description of the important experimental capabilities, which are constantly upgraded to meet the requirements of industry for innovative fuel and material development. The achievements of the HRP and future prospective plans rely on the versatility of the research carried out in the reactor with reliable testing techniques and in-pile instrumentation.

1. Introduction

The Halden Reactor Project (HRP) is well known as an undertaking of many nuclear organisations from 18 countries (in 2010) which joined efforts with the main objective to support safe, reliable and effective utilization of nuclear fuel in operating commercial nuclear power plants. This international Project is hosted by the Norwegian Institute for Energy Technology (IFE) which is leading the HRP joint nuclear research program established from member countries shown in Figure 1. The program is renewed by the member organizations every third year to meet nuclear industry requirements. In addition to the joint program experiments, a number of organisations from the participating countries execute their own development work on a bilateral or multilateral basis with IFE-HRP structures. The researchers are utilizing the HRP infrastructure with the maximum possible effectiveness whereas IFE is committed to continue efficiently responding on technical issues and to

Figure 1 HRP member countries as for 2010

maintain the facilities in the conditions keeping high competitive capacity. [1]

The Halden Boiling Water Reactor (HBWR) has been operated for many years and at the same time has been progressively upgraded with innovative techniques and systems which enable it to be one of the most versatile test reactors having unique experimental capabilities [2].

This paper shares the experience of HBWR utilisation, describes its experimental capabilities and provides information about systems upgrading for innovative fuel and material investigations. The reactor is operated under close supervision of the national and international radiation protection authorities, who make recommendations in regard of enhancing the reactor system operational safety as well as extending reactor lifetime. The current and future programs of the HRP are thus based on the long-term operation of HBWR.

2. Upgrading of the Halden Boiling Water Reactor

The Halden reactor is located in a coastal town in the south-east of Norway near to the Sweden border. The reactor hall (see Figure 2) is built in a rock cavern with an area of about 7000 m^2 and 30 – 60 m thick that acts as a natural containment. The control room, service buildings with offices, workshops and laboratories are placed outside the excavation as shown in Figure 3.

From history, IFE (original name: Institute for Atomic Energy) was initially established with the plan to build a prototype nuclear reactor for civil industry. The HBWR was initially designed to utilize natural, metallic uranium loaded in a steel pressure vessel with heavy water as the moderator and also the coolant, circulating under natural convection. The energy released in the reactor was transferred by steam transformers to a second, closed circuit containing light water. The pipes transporting the steam in the second circuit passed out of the containment, transferring the heat to a third, light water circuit which in turn delivered steam to the

Figure 2 Halden reactor hall

neighbouring paper factory. The circulatory system of the HBWR has the same design today, and steam is still delivered to the neighbouring paper factory, however the fuel core has undergone many design changes.

The first fuel core of the Halden reactor comprised seven tons of natural uranium generating a maximum power output of 5 MW with steam at about 150°C. The research performed during the initial reactor operating cycles focused mainly on reactor physics, water chemistry and the effects of radiation. Since 1967, the reactor has operated at a pressure of 33 bar, with a corresponding saturation temperature of 240°C, and has utilized enriched fuel to generate a maximum thermal power of 25 MW. This made the HBWR a unique research reactor, with its spacious and accessible core where it is possible to irradiate simultaneously 30-35 fuel and material test rigs. The scheme of the reactor core is shown in Figure 4 and operational data are given in Table 1.

Figure 3 HBWR plant site
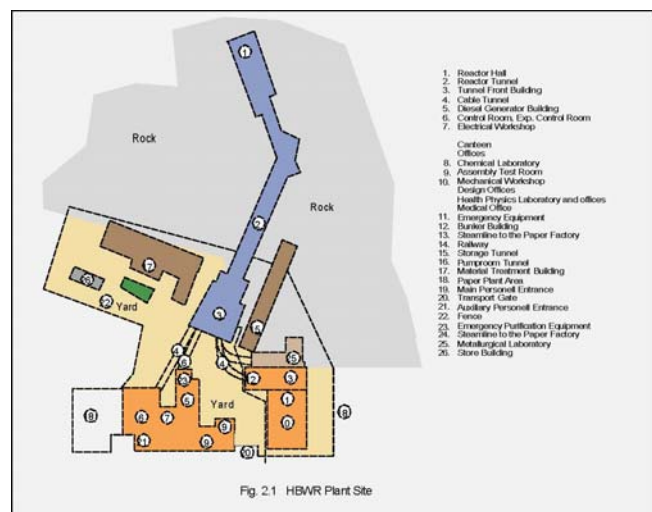
Table 1 HBWR data

Figure 4 HBWR core with indication of some of the special test rigs

Power	Up to 20 MW
Reactor type	Pressurized vessel reactor
Moderator / coolant	D_2O
Pressure	33.3 bar
Saturation temperature	240 ^{o}C
Primary circuit	Natural +forced
Primary flow	160 ton/h
Total number of cells	300
Number of central cells	110
Number of control stations	30
Number of cells for experiments	30
Max height of fuel	1.71 m
Height of active core	0.8 m
Driver fuel* (Table 2)	Fuel rod assemblies
Lattice pitch	Hexagonal – 130 mm
Av. thermal neutron flux	$3.0\ 10^{13}$ n/cm^2 s

Data on the driver fuel assemblies for the reactor are given in Table 2.

Utilization of enriched driver fuel enabled the active core radius to be reduced enabling not only fuel testing in the central part of the core under higher power but also a reduction in radiation dose to the reactor vessel and hence a prolongation of the reactor lifetime. In addition, the vessel welds are protected by neutron shield bolts.

According to the requirements, the inspection and test programmes include ultrasonic examination of vessel welds, lid, bolts, bottom nozzle and primary system piping. The irradiation induced changes in the vessel material are being monitored by material testing every 6th year, flux evaluations and fracture mechanics analysis. Charpy testing and fracture mechanics analysis of surveillance specimens are performed by VTT's laboratory in Finland, using material specimens with appropriate fluence. Flux assessments during operational time enable

Table 2 HBWR driver fuel data

Shroud diameter	71 mm
Wall thickness	1.0 mm
Number of rods	8
Length	0.8 m
Fuel	UO_2
Form	Sintered pellets
Enrichment	6%
Density	10.5 g/cm^3
Pellet diameter	10.5 mm
Pellet height	8.5 – 10.8 mm
Cladding	Zr-2, Zr-4
Wall thickness	0.8 mm
Inner diameter	10.67 mm

quantification of the fluence received by the different parts of the vessel, taking into account the changing core loading over the years. Figure 5 shows the current and predicted adjusted reference temperature of the vessel surveillance samples tests vs. fast neutron fluence evaluated for the HBWR vessel. The material tests and the analysis performed have shown that the reactor can be operated safely well beyond 2030.

The average availability of the HBWR is around 50% from calendar time, thus the actual time of the reactor vessel irradiation is less than 30 years of its lifetime. During all the years of operation the HBWR, systems have been constantly upgraded during periodic shutdowns and short-stops for refurbishments. While about 70% of all the outage time has been used for inspection of current tests, loading new test rigs and unloading of completed tests, the remaining 30% (or 8 years) of this accumulated outage time has been dedicated to plant refurbishments and upgrading. The reactor control systems, primary and secondary cooling circuits, electrical
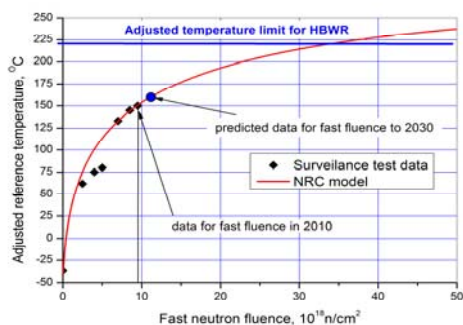
Figure 5 Surveillance data on vessel

supply systems, instrumentation, safety system and other parts important for normal operation have been upgraded or replaced as indicated in Table 3.

The components and pipes in the primary, secondary and tertiary systems are also all inspected within 9-year periods. In 2003, during planned in-service inspection of the primary sub-cooler pipe, non-penetrating micro-cracks in the welds were detected by improved x-rays technique. The welds were repaired by the Weld Overlay Repair technique (WOR).

Table 3 Upgrading of the Halden reactor for years of operation and plans for the future

Upgrading / periods	1960-70	1970-80	1980-90	1990-2000	2000-2009	2010-2020 (plans)
Control rods	X				X	
Safety systems		X			X	**X**
Control room	X					**X**
Electrical systems	X			X	X	**X**
Cooling system	X	X		X		
Heat exchangers		X			X	
Instrumentation		X	X			**X**
Water Purification			X		X	**X**
Pumps			X			
Primary circuits				X	X	
Secondary circuits					X	**X**
Handling technique						**X**

The following main upgrades were done recently:
- Replacement of the two steam transformers
- Replacement of valves and piping in primary steam and sub-cooler circuits
- Replacement of manifold in primary sub-cooler circuit
- Replacement of one sub-cooler
- Replacement of pipes and valves in secondary circuit
- Replacement of the system for regulating of the control rods
- Installation of a new alarm system
- Installation of a new electrical transformer in the reactor hall
- Upgrading of fire safety in compliance with IAEA requirements.

As a result, the reactor vessel, lower main pipe and the steam manifold are the only remaining parts of the original construction as shown in Figure 6.

In addition there is a plan for upgrading of the reactor systems in the period of 2010-2020:
- Installation of new safety and shut down valves
- Rebuilding of the heavy water purification system in primary circuit and installation of purification system in secondary circuit
- Replacement of Uninterrupted Power Supply unit Replacement of three electrical transformers
- Building of new handling compartment
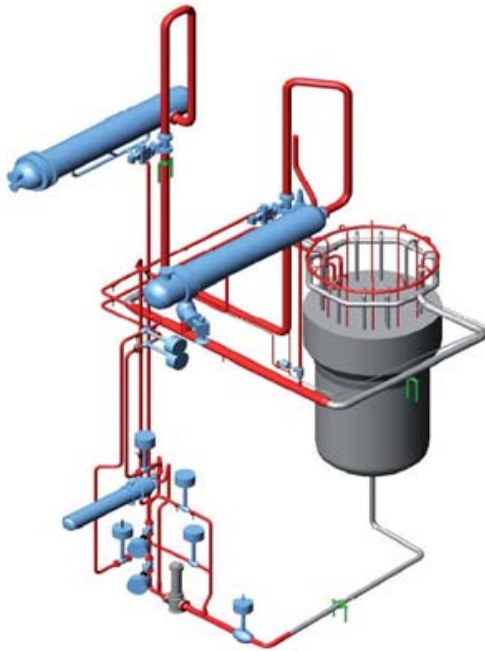- Installation of large screen control room

The main objectives of the upgrading are:
- Maintain a high level of safety:
  - Monitor vessel embrittlement
  - Update safety and alarm systems
  - Perform IAEA reviews
- Extend the reactor licence beyond 2018
- Increase reactor availability up to 60 %

The other main long-term and strategic goal of the HBWR upgrades is to enhance thermal and fast fluxes, which will create more testing opportunities for irradiation of low enriched / high burn-up fuel, power ramps and materials tests. This is possible to achieve by reducing the neutron absorbing materials in the reactor core and the following measures are being considered:
- Exchange/upgrading of the D_2O moderator
- Minimising the stainless steel volume in the reactor by Zr-replacement where possible
- Reducing utilization of B-10 (absorbing isotope) in PWR loops by replacement with B-11 (non-absorbing isotope)
- Substituting light water with heavy water in test loops where PWR thermal-hydraulic conditions are required with no special requirements for the water chemistry

Figure 6 Upgrading of the HBWR primary circuit:
Grey: keep original;
Red: replaced pipes;
Blue: replaced components

- Optimising the number and placement of instruments that are neutron absorbing
- Utilizing alternative materials for production of tests rig or instrument parts that are less neutron absorbing than the materials currently used

It should be noted that there is no issue of transition from HEU to LEU for the HBWR because the driver fuel elements already in use are of only 6 wt% enrichment. Nevertheless, the economic aspect of using 4.95 wt% enriched fuel is under consideration also for the strategic advantage it would bring due to the wider availability of this enrichment compared to higher enrichments. Alternatively, higher U-density fuel is also being considered as a future candidate for the driver fuel assemblies.

3. HBWR Experimental Systems and Upgrading of Experimental Capabilities

From the 1980's, the experiments in the Halden reactor needed to be more complex to study detailed properties and behaviour of nuclear fuel and reactor materials under irradiation [3]. This was achieved by means of sophisticated experimental systems and in-pile instrumentation developed and produced in-house. In addition to the basic experiments performed under HBWR moderator conditions, a number of high-pressure water loops were installed to enable investigations of re-fabricated commercial fuel and core component materials under representative PWR, BWR and CANDU thermo-hydraulic conditions. More demanding corrosion tests also increased the requirements to cooling water quality, which led to the installation of new water purification systems and also interim inspection techniques. Nowadays

the external systems employed at HRP to perform special in-pile experiments in the Halden reactor are shown in Table 4.

Table 4 HBWR external systems used for special experiments

Tests / Systems	Water loops	Gas flow system	Ultra high pressure (to 500 bar)	Hydraulic drive system	He-3 power control	Inspection technique	Gamma scan system
FGR and PCMI	X				X		
Ramp	X			X	X	X	X
LOCA	X				X		X
Dry out	X						X
Lift-off	X	X	X				
Fuel rod gas flow		X					X
Clad corrosion	X					X	
Clad creep	X		X	X			
Fuel creep		X	X				
CRUD deposition	X			X		X	
Second degradation						X	X
Material relaxation		X	X				
Crack propagation	X	X	X				
IASCC test	X	X	X			X	
Material creep		X	X				

The gas flow and ultra-high pressure systems were developed for a series of tests such as fuel rod overpressure (lift off) and fuel rod flow test to study fission product with on-line gamma spectrometry. The ultra-high gas pressure system (up to 500 bar) is also used for stress loading via bellows units to study cladding and fuel creep as well as reactor material degradation in specially designed test units. The increasing complexity of the HRP research program has also made it necessary to extend the experimental control room at the reactor.

In the last decade, the HRP members initiated a test programme for the investigation of high burn-up fuel behaviour under LOCA conditions. A specially designed test rig and water loop, with

Figure 7 LOCA test rig and gamma scan of the rod tested

simulation of blow-down and a radioactive dump-tank, were developed for the series of experiments. Figure 7 shows a schematic view of the rig and an example of the fuel gamma scan obtained just after one of the LOCA tests. New gamma scanning equipment has also been
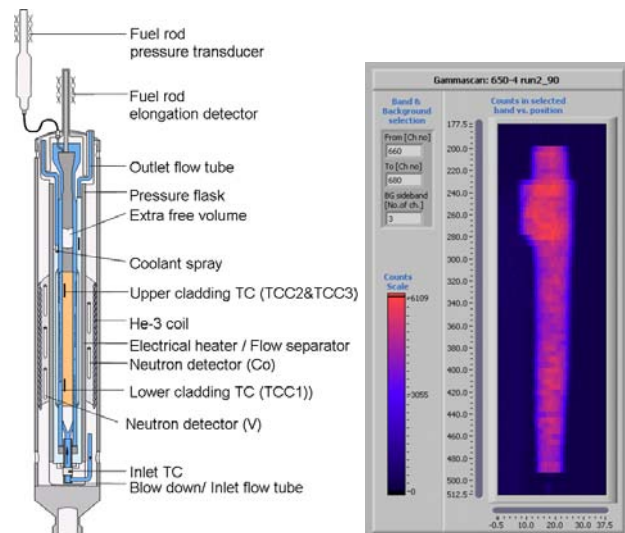
installed for these tests to investigate fuel relocation inside the ballooned region of the fuel cladding.

Recently, some of the external systems have been upgraded to improve or to automate the control of experiments. For example, the He-3 gas power control system has been updated with the installation of a PLC (Programme Logical Controller) and with a He-3 gas purification system. The system is employed for the power ramping programme. A new test rig has also been designed for ramping four fuel rods in one loading by moving them one at a time into the axial centre of the reactor core by means of a hydraulic drive system as show in Figure 8
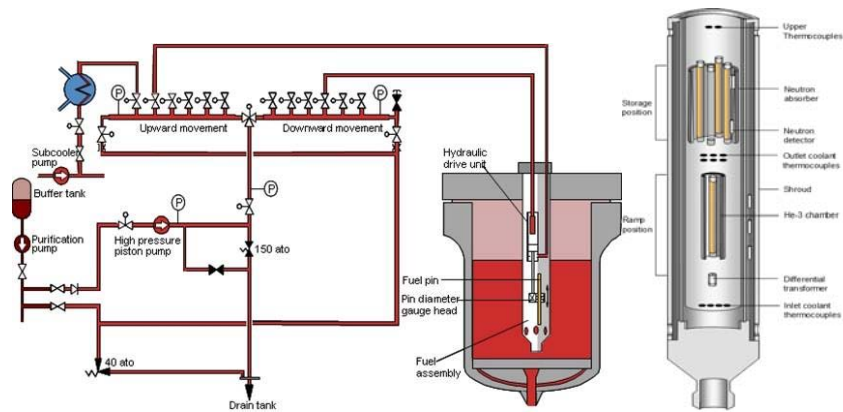
Figure 8 Hydraulic drive system used for ramp tests in HBWR

The HRP irradiation programme also relies on the extensive use of in-pile instrumentation for online monitoring of key parameters like gas pressure (fission gas release), fuel and cladding temperatures, fuel and cladding elongation (swelling, densification, irradiation growth and stress relaxation) and cladding diameter tracing (creep and CRUD deposition). The basis of many of the in-pile instruments developed at Halden is the LVDT (Linear Voltage Differential Transformer – see Figure *9*), which can provide in-pile measurements during a 5-10 year period.
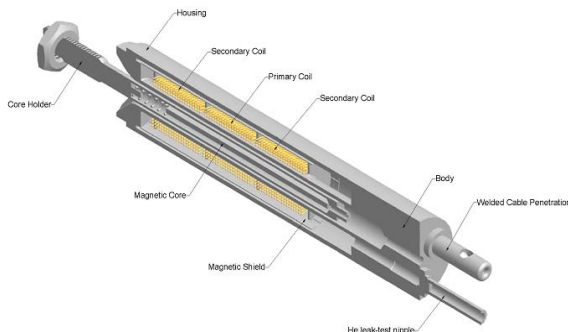

Figure 9. LVDT principal design

The quality of HBWR experiments are also maintained by the development of new instruments and special units for material tests. The latest instrument development currently underway is for on-line corrosion measurement that will assist in the study of material corrosion mechanisms. In this regard, new ECP sensors with different electrodes, on-line potential drop corrosion monitors and Electrochemical Impedance Spectroscopy (EIS) units are under development and testing in the Halden reactor.

Each of the test rigs loaded in the Halden reactor have different specific objectives, but many also have the overall goal to investigate innovative fuel and cladding materials, which are being developed to enable improved safe and reliable operation to as higher as practicable fuel burn-up. Fission gas release and cladding corrosion are the main issues which can directly limit allowed fuel discharge burn-up and reduce the effectiveness of NPPs. The sophistication of the experimental systems and test rigs with purpose-built instrumentation at the HBWR allows an integral approach for investigations of different types of innovative fuel and cladding materials. In this way the HRP are supporting the industry in resolving these issues.

The following are examples of innovative types of fuel under investigation in the HBWR:

- UO_2 large grain fuels which are potentially able to suppress excessive FGR at high burnup
- Additive fuels containing Gd isotopes for suppressing initial reactivity of the reactor core
- High conductivity UO_2 fuel containing BeO enabling reduced fuel temperature with the consequence of reduced FGR for the same operating power

There are plans to investigate other perspective fuels in future. The recently developed Zr-based cladding alloys with improved corrosion resistance, like ZIRLO, M5, E110M and M-MDA, are currently under testing in the Halden reactor. The possibility of testing innovative SiC claddings in the HBWR in the future is also under consideration.

There is also considerable and increasing interest to study aging of construction materials under irradiation, in order to enable extension of current NPP design lifetime. The HBWR flux, external experimental systems and instrumental capabilities are essentially suitable for such kinds of tests.

It should also be mentioned that the HRP is participating in a measured way in Generation IV reactor research. A feasibility study for the possible installation of a supercritical water loop in the Halden reactor for fuel and material irradiation at higher pressure and temperature (250 bar and 600°C) has been carried out and the development of instruments able to withstand such conditions is underway [4].

4. Conclusion

During many years of operation, the Halden reactor played an important role in support of world-wide nuclear energy development, particularly in the area of reliable, safe and effective utilization of nuclear fuel and materials [3].

A history of the HRP establishment gives a picture of an international Project which successfully relies on the versatility of the research carried out in its reactor with reliable testing techniques and in-pile instrumentation. The results from the Halden reactor continue to be the basis for the safe introduction of innovative fuels and materials to NPPs.

The reactor and its associated experimental systems have been steadily updated. The constant availability of the HBWR is based on maintenance, upgrading and supervision of the reactor systems which play a key role in safe operation. The recent modifications as well as plans for upgrading will continue the HBWR long term operation in the future.

5. Acknowledgements

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STATUS OF THE MELODIE EXPERIMENT, AN ADVANCED DEVICE FOR ONLINE BIAXIAL STUDY OF THE IRRADIATION CREEP OF LWR CLADDING

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As a prototype of future instrumented material experiments in the Jules Horowitz Reactor (JHR), the CEA in collaboration with VTT is in the process of starting the MeLoDIE experiment in the OSIRIS reactor in about a year. MeLoDIE is an initiative to go beyond the "cook and look" concept, often used for past material experiments in MTRs. Its innovative sample holder aims to an online measurement of the biaxial creep of a LWR cladding sample under twice the typical fast neutron flux in LWRs. It is designed to perform an online-controlled biaxial loading of the sample and an online biaxial measurement of its deformation with sensors co-designed with IFE Halden. Technical challenge is to perform reliably accurate measurements under the high nuclear heat load of in-core locations while keeping within their tight space and safety constraints.

Index Terms — OSIRIS, JHR, Zircaloy, biaxial stress, irradiation creep, online instrumentation

Introduction

During its stay inside a Light Water Reactor, zircaloy fuel cladding undergoes non-monotonic and multiaxial stresses due to the combination of the rise of fission gas pressure and of Pellet-Cladding Mechanical Interaction. Prediction of their effect is even made more complex by the anisotropic properties of zircaloy products of industrial use.
This drives the need for reliable experimental data on their behaviour under irradiation, when subjected to multiaxial stresses. This need was addressed at the conceptual design level by the work package WP1.1 of the European MTR+I3 program [1].
The MeLoDIE proof-of-principle experiment is a follow-on of this program [2]. It started in 2009 as a collaboration between CEA Nuclear Energy division and VTT Technical Research Center of Finland, in the framework of the Jules Horowitz Reactor project.
To meet these needs, The MeLoDIE sample holder is able to apply a biaxial stress on a fuel clad sample with full online control and also to measure online its biaxial creep. Reliability of instrumentation in a harsh environment and capability to apply high stresses with tight constraints on the space available for the loading frame are the main issues in its design.
This paper reports on the present status of MeLoDIE.

I. MELODIE overall setup

MeLoDIE, an acronym which stands for Mechanical Loading Device for Irradiation Experiment, has already been described elsewhere [2]. It is an instrumented in-core creep experiment for a fuel cladding tube specimen under controlled bi-axial loading. The irradiation will be carried out in one of the core positions of the OSIRIS research reactor, in static NaK, using a CHOUCA capsule. Target temperature is 350°C. The lower part of the MeLoDIE sample holder is shown in Figure 1.

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The core of MeLoDIE experiment is a mechanical device with pneumatic controls, i.e. the biaxial loading is induced by independent control of both the internal pressure and the bellows pressures to induce various hoop and axial stresses on the pressurized tube specimen. Additionally, the axial creep strain is measured continuously during the experiment by using a fixed LVDT probe and the diametral strain is measured periodically along the specimen surface, by a moving DG diameter gauge. Displacement of this DG probe is driven by a pneumatic mover and controlled by a second LVDT probe. Instrumentation of the MeLoDIE experiment requires a total of 12 thermocouples, 4 pressure tubes and 6 LVDT signal cables.

Figure 1. The lower part of the MeLoDIE sample holder showing from left: LVDT of mover, DG probe mover, specimen LVDT, bidirectional axial loading device, specimen and DG probe.

II. Helium loop and pressure control

MeLoDIE closed helium loop is designed to generate a continuous helium gas flow for four servo-controlled pressure loops (see figure 2). These loops are used to control:
-    the internal pressure of the test specimen,
-    the pressures of the two bellows of the loading device,
-    and the pressure of the bellows of the mover.
 The servo-valves and pressure sensors of each loop are electrically connected to the programmable servo-controller together with the DG and LVDT sensors. The closed helium loop is based on a gas booster which works as a pressure intensifier. The maximum helium feed pressure to the circulation loop is 180 bars and the return pressure from the loops is 4-6 bars. Operation of the closed helium loop relies on pressurized air and pneumatic control valves without any electrical components. The closed helium loop is connected to an helium reservoir in order to compensate for the unavoidable leakages.

The pressure control loops are fed by a continuous helium gas flow through the electrically controlled servo-valves. The movement of the bellows is controlled either by LVDT probe or by bellows pressure which give the feedback signal for the servo-controller. This controller compares the actual feedback signal to the set value and close/open servo valves to induce the movement of the bellows by increase/decrease of their pressure [3].
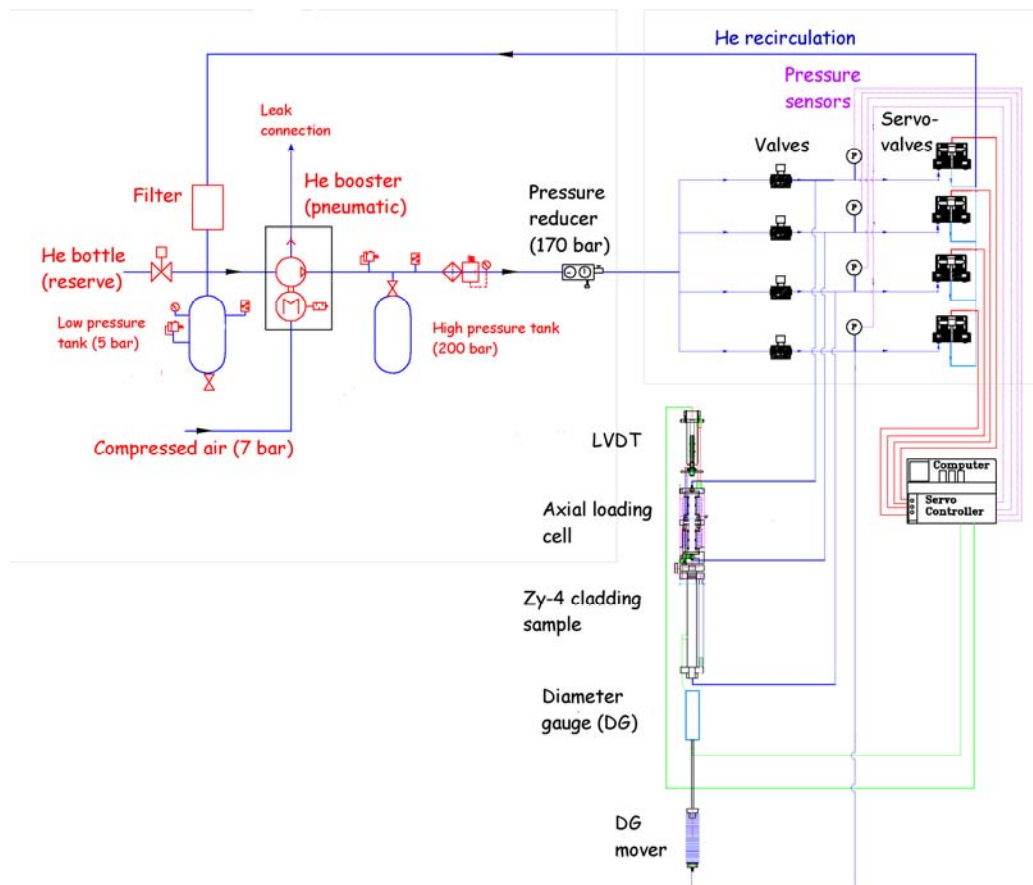
Figure 2 : Schematics of the MeLoDIE experiment : top is out-of-pile gas management system (online stress control), bottom is the in-pile section with the Zy-4 sample and the biaxial loading and strain measurement system.

III.   Measurement systems

Measuring online the sample elongation and diameter change is a major challenge for the MeLoDIE experiment. In Material Testing Reactors, fuel or material deformations are primarily detected using electromagnetic sensors. Furthermore, many MTRs rely on LVDTs fabricated by the Institutt For Energiteknikk (IFE) in Halden, Norway [4].

However, for use in MeLoDIE experiment, changes were required on IFE sensors to meet both the measurement needs and the geometrical constraints. For this reason CEA, in collaboration with the IFE, initiated an effort to enhance the performance of LVDTs and DGs and proposed several improvements to standard designs.

One is the development of '5-wire' or "self compensating" sensors and electronics. Due to the use of the secondary signal ratio to quantify dimension changes rather than the combined secondary signal, these sensors are virtually no longer susceptible to Curie temperature effects. IFE tests showed that the influence of temperature on the signal around the Curie point is reduced considerably compared to standard 4-wire configuration. Tests performed at CEA in 2010 showed that these 5-wire sensors have also better sensibility and accuracy than their 4-wire counterparts. As an illustration, the measured repeatability of the 5-wire LVDTs and DGs for 1mm range was respectively 3.0 µm and 1.0 µm (2σ).

Secondly, geometrical adaptations of standard IFE diameter gauges were necessary for implementation on the MeLoDIE sample holder, as the usable diameter in this irradiation experiment is only 24 mm. Thus CEA and IFE proposed a new DG design in which the sensor body is positioned axially under the sample. Three arms ended by ceramic feelers surround the sample and scan its diameter. The contact between the sample and the feelers is driven by two counterweights located at the DG bottom.

Profile measurements are performed by moving the DG up and down with a pneumatic drive system. The precise position of the DG is detected online using an additional LVDT

having an extended measurement range of 80 mm. A calibration step is incorporated in the bottom end plug of the sample, and is used as a reference diameter. For each diameter trace, the DG moves over this calibration step before and after the scan. The DG sensitivity can thus be checked and corrected at each profile measurement. Tests performed by VTT on MeLoDIE mock-up in Espoo showed the good repeatability of this profile measurement as indicated in Figure 3.
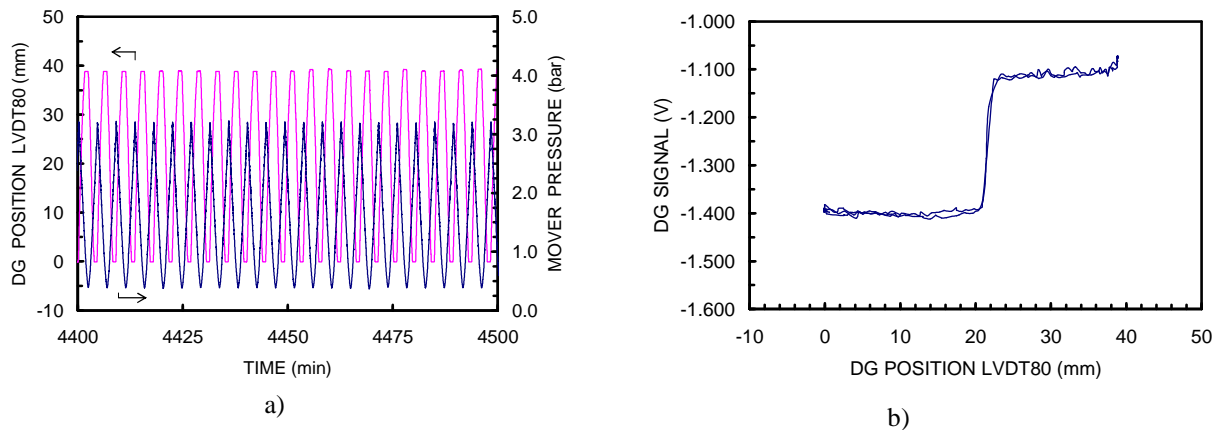

a)

b)

Figure 3. The MeLoDIE mock-up tests indicating a) the mover performance, i.e., mover pressure and corresponding DG position measured by LVDT80 and b) measured DG signal along the test specimen showing the effect of calibration step.

IV.   Thermal simulations

Thermal simulations for the MeLoDIE loading device were carried out for the OSIRIS reactor core position 44, with a fast neutron flux of 2.10^{18} n/m^2/s (above 1 MeV) and a specific nuclear heating of 9.5 W/g. The temperature distribution obtained by an axisymmetric model with an external heating power of 15 kW/m is shown in Figure 4. The results indicate that the maximum temperature is about 420°C at the upper and lower specimen fixing flanges. They also show temperature variations of about ±15 K along the vertical direction of the test specimen. These temperature gradients are caused by the higher mass distribution close to the specimen ends and by the gradient in the nuclear heating flux.
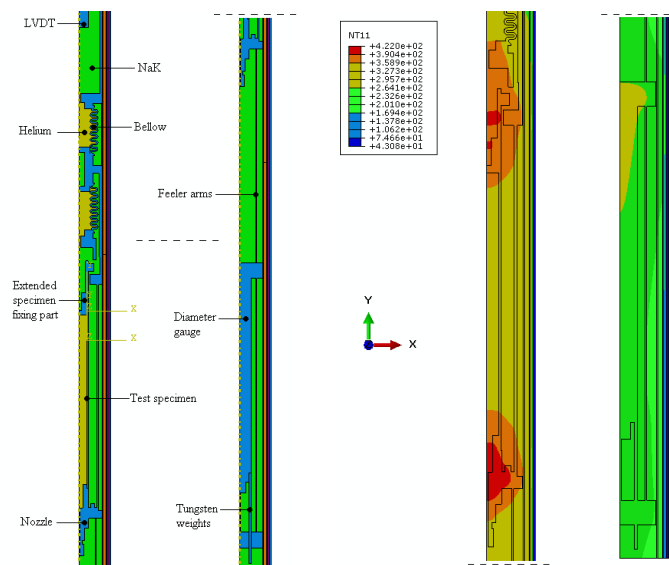

Figure 4. Axisymmetric model and corresponding temperature distributions of the loading device and DG probe when the DG probe is in the resting position with external heating power of 15 kW/m.

In order to simulate heating effects of the moving DG probe on the specimen temperature, a transient heat transfer analysis was performed. The movement of the DG probe was simulated by changing instantly at t=0 the material properties and thus the thermal loading in the zircaloy sample neighborhood. This means that, at the initial state of the simulation, the feelers are assigned to have the material properties and thermal loading of the NaK alloy, thus the initial state of the transient case is equivalent to the steady-state case calculations. This time-dependent transient simulation should predict the temperature perturbation during diameter scans in the MeLoDIE experiment more realistically than steady-state simulations which give the temperature distribution after an infinite period of time.

The results of the transient analysis, performed using the ABAQUS code, are presented in Figure 5. After 3 seconds the temperature of the contact point between the feeler arms and test specimen increase of about 9 K, i.e. about 3 K per second. This shows the need for a quick scan not to alter the local properties of the sample.
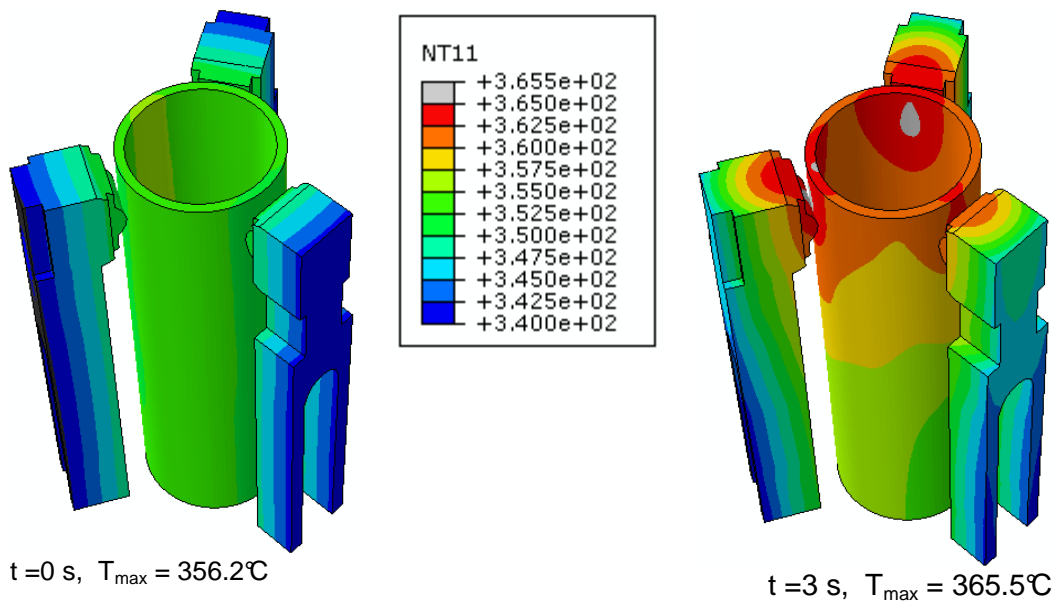

$t = 0$ s, $T_{max} = 356.2$℃

$t = 3$ s, $T_{max} = 365.5$℃

Figure 5. 3D transient analysis of the feeler arms of the DG probe showing temperature distribution after 0 and 3 seconds time interval.

Conclusion

The pneumatic systems and lower part of the sample holder of the MeLoDIE experiment are now operational. More work is to be done to fine tune and to assess the accuracy and the reliability of the diameter electromagnetic gauge. Data collection should start in about a year in the OSIRIS reactor. First irradiation will be devoted to benchmark MeLoDIE capabilities with a well-known material, the AREVA Zy-4 SR alloy.

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THE ULTRA-COLD NEUTRON LABORATORY
AT THE FRM II

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ABSTRACT

A new strong source for the production of ultra-cold neutrons (UCN) is currently built up at the Forschungsneutronenquelle Heinz Maier-Leibnitz (FRMII).It will be installed at the horizontal beam tube SR-6, with a solid hydrogen pre-moderator and a solid deuterium UCN-converter located approx. 60 cm away from the fuel element. UCN are produced inside the solid deuterium via the superthermal principle of conversion of the pre-cooled neutrons coming from the solid hydrogen. The UCN will be extracted out of the converter and guided through the biological shield to several experiments located inside the experiment halls of the FRMII. These experiments are investigating fundamental properties of the free neutron, such as its lifetime, a possible electric dipole moment or the quantum mechanical interaction of neutrons with the earth's gravitational field field. The expected UCN densities in the experiments will be 2-3 orders of magnitude higher than the densities reached at the currently strongest UCN-source located at the ILL. This talk will give an overview of the setup of the UCN source at the FRM-II, technologies used, and of the connected experiments. The current status of the project and future developments will be presented.

1. Introduction

Spontaneous breaking of fundamental symmetries is an attractive topic in modern particle physics. Understanding qualitatively and quantitatively the parameters involved in these kinds of processes could help to explain the unbalanced presence of matter (baryons) with respect to antimatter (anti-baryons) in the universe. Due to their intrinsic properties, ultra cold neutrons (UCN) are excellent candidates for experiments measuring with high level of accuracy parameters like the electric dipole moment

Figure 1 UCN Densities of existing Sources

(EDM), the neutron lifetime (τ_n), the axial-vector coupling constant (g_A), or in search of quantum effects of gravity. (1)

The research of these fundamental questions is the motivation to implement the UCN Laboratory at the FRM II in Garching. Furthermore UCN densities actually reached (see Figure 1) shall be improved up to 10^4 UCN/cm³. The project is supported in addition by the
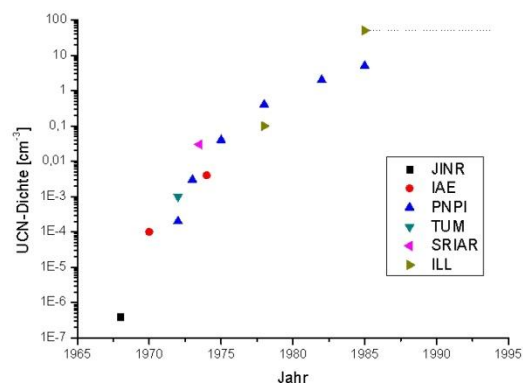
DFG Excellence Cluster "Origin and Structure of the Universe". Hereafter the major components of this laboratory are described.

2. The Ultra Cold Neutron Source

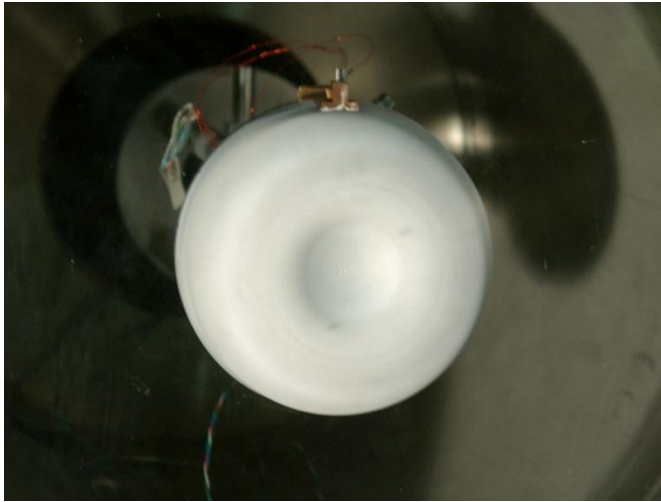The Munich high-flux reactor FRM II offers the possibility to install a unique source for ultra cold neutrons (UCN) with a small volume of solid deuterium (50 g) at a temperature of 5 K as converter, exposed to a neutron flux produced by solid hydrogen in the same converter assembly. One key-component in this new scientific equipment is the neutron converter: a customized vessel filled with solid hydrogen inside and solid deuterium (see figure 2). The converter will be installed in a through going beam tube (SR6). In order to manage safely this very cold material in different

Figure 2 UCN Converter Mockup with frozen Deuterium

Diameter 110 mm Temperatur 6-10 K

operation phases and accordingly the thermal loads, a dedicated cryogenic system was designed. The major characteristics are a supercritical helium loop at 3.5 bar_{abs} and am mass flow rate of 120 g/s.

Furthermore gas handling systems for the deuterium and hydrogen are required. The storage of 12.5 mol of deuterium and hydrogen each requires vessels of ~300 l. The system is operated by valves driven by pneumatic actuators powered with helium.

The design work was supported by many simulations based on Monte Carlo and finite element methods. The load on the converter is based on a maximum thermal neutron flux of $\Phi_{therm}= 1.25 \cdot 10^{14}$ 1/[cm²· s]. The heat load on the converter parts is between 0.12 and 0.3 W/g depending on the material and their position. (2) .

A major focus was on the safety of the system. On the one hand side the given research reactor protective requirements shall not be violated and on the other hand side the handling of radioactive tritium and potential explosive hydrogen has to be managed.

All barriers containing radioactive matter are constructed following higher quality standards. The helium loop entering the beam tube is separated by an extra heat exchanger from the compressor loop of the helium cooling system.
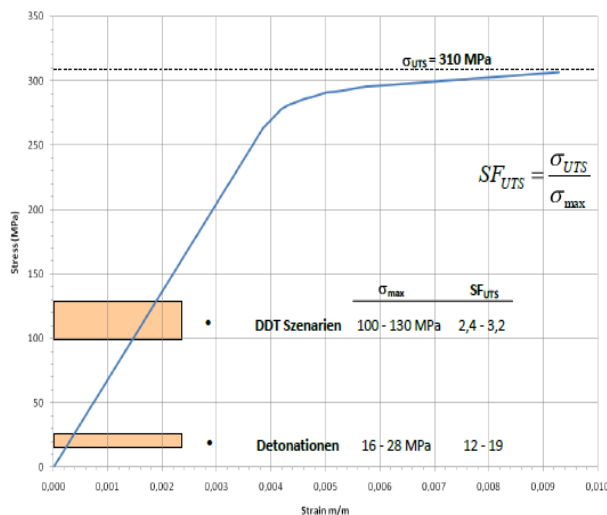

The potential ignitable gases hydrogen and deuterium are in double walled containments with a pressure below ambient and the outer part is filled with helium under higher pressure than atmosphere. This allows detection of leaks before an air inflow takes place. Finally the vessels are designed to withstand a potential blast.

The converter lance (appr. 3.5 m length) is installed in EN AW 6061T6 aluminum tube which protects beam tube SR6 in case of increased pressure. Figure 3 shows the

Figure 3 Stress & Strain Diagram for EN AW6061 T6 Double Tube in case of mixture ignition

stress to be expected. (3) The yellow bars are showing the pressure influence on the material of the two possible scenarios, which are in the elastic range of the aluminum.

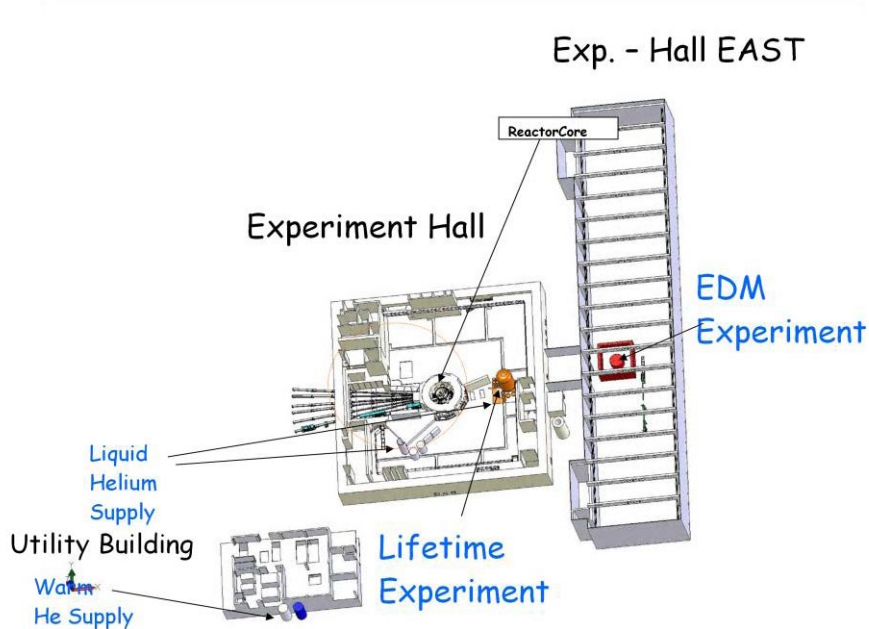Figure 4 shows the situation of the UCN Laboratory in the FRM II.

Figure 4 Situation of UCN Laboratory at FRM II Experiment Hall 40 x 40 m

3. The Lifetime Experiment

The lifetime of the free neutron is about (885.7±0.8) s(4). Improving the uncertainty of this value shall help to understand the processes of baryogenesis during the first seconds of the early universe.
The Lifetime experiment consists of a superconducting magnetic bottle (see Figure 5) which allows the storage of ultra cold neutrons due to their magnetic dipole moment. The neutron decay is measured indirectly by the measurement of the protons produced during the decay time. Out of these observations the lifetime is derived. The experiment shall be located in the FRM II experiment hall.

The cryo cooling system of the UCN source will also support the experiment with respect to the liquid helium cooling of the superconducting magnets.
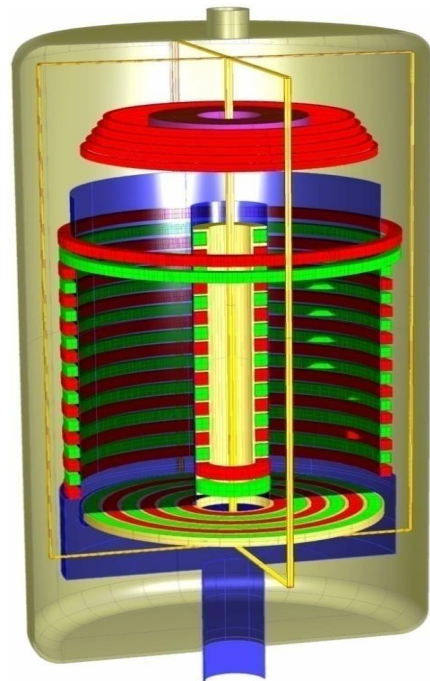
Figure 5 Superconducting magnetic storage bottle of Lifetime Experiment Diameter 0.6m height 1.8 m

4. The Neutron EDM Experiment

A measurement of the neutron EDM is typically based on Ramsey's method of separated oscillatory fields(5), an interferometric nuclear magnetic resonance method applied to trapped ultra-cold neutrons. This experiment requires polarized UCN. To minimize statistical and systematical uncertainties highest UCN densities in this experiment are required.
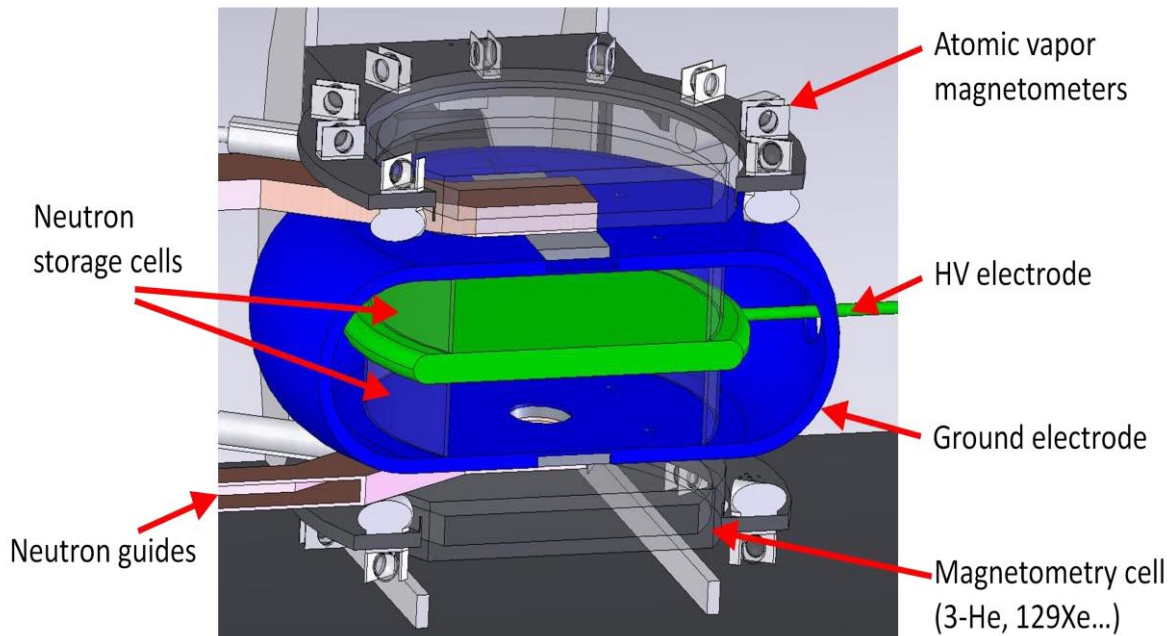
Figure 6 Measurement cell for neutron electrical dipole moment measurement about 0.5 m diameter

This is done in a chamber (see Figure 6) which has to be placed in an environment with controlled magnetic field gradients. This equipment is planned to be built up in the eastern experiment hall of FRM II.

To fill the experiment with UCN in about 35 m distance from the UCN source sophisticated UCN guide tubes are required. Special replica techniques are developed in cooperation with TUM to prepare such guides having transmission- factors of ~99%/m.

5. Acknowledgement

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NOISE THERMOMETER AT THE FRM II HOT NEUTRON SOURCE

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ABSTRACT

The 20 MW research reactor FRM II operated by the Technische Universität München is equipped with a hot neutron source (HNS). The source is aimed to shift the well thermalized neutron spectrum in the heavy water moderator tank to higher energies as requested by the experimental users.

The main component of the HNS is a solid graphite cylinder being heated by gamma radiation from the reactor core up to a temperature of about 2000 °C. The hot graphite cylinder is surrounded by a high-temperature insulation of carbon fiber, to achieve and maintain the high temperature. Due to the extremely harsh environment, the high temperature and the nuclear radiation, the temperature inside the graphite cylinder is measured by a purpose-built noise thermometer. It measures the white noise of an electrical resistor and determines the absolute temperature of the graphite cylinder.

During nuclear commissioning of the hot neutron source, the temperature of the graphite cylinder was measured by the noise thermometer at several power steps of the reactor. The following relevant parametrs of the HNS had been determined: the maximum temperature, the heating rate and the cooling rate after shut down of the reactor. The relative long time needed to reach the maximum temperature was used to measure the heat-up effect of the HNS.

Since the nuclear start-up of the reactor the noise thermometer of the HNS is operated without significant problems.

1. Introduction

FRM II is a heavy water moderated 20 MW multi-purpose research reactor. Among many other experimental installations the FRM II is equipped with a hot neutron source (HNS), a spectrum shifter aiming the increase of the flux density of superthermal neutrons as they are needed to investigate particular problems in condensed matter physics [1]. The central component of the HNS is a hot moderator, which is located within the D_2O-reflector tank of the FRM II close to the thermal neutron flux density maximum. A beam tube is directed towards the vessel of the HNS and supplies up to two experiments with hot neutrons of energies from 0.1 to 1 eV.

The essential part of the HNS is the hot moderator, a cylinder-shaped installation made out of nuclear-grade graphite. The cylinder is 200 mm in diameter and 300 mm in height. It is heated by gamma radiation of the reactor core to temperatures of about 2000 °C. To achieve this high temperature, the cylinder is thermally insulated by several layers of carbon fiber and a rigid hemisphere made of bounded carbon fiber at the bottom. Besides insulation these

components are also used to fix the graphite cylinder in the center of the HNS to avoid the contact with the "cold" walls.
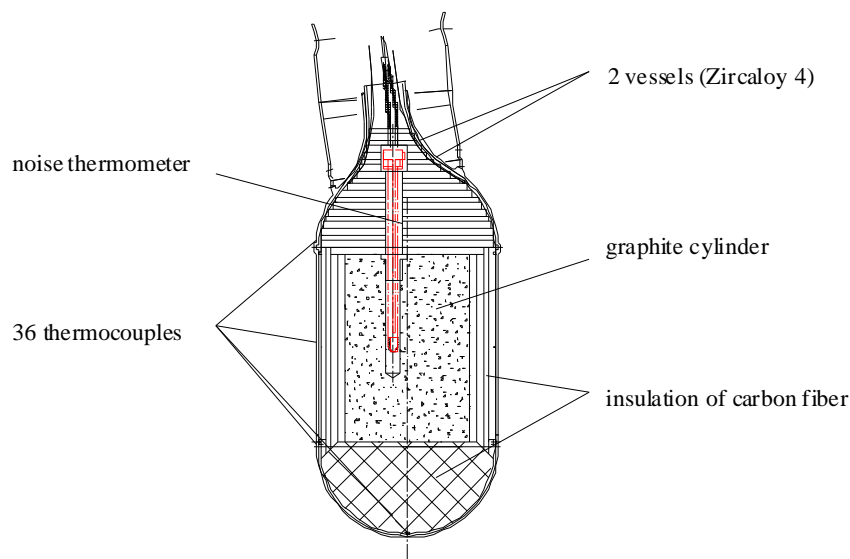
Fig. 1 Cross section of the hot neutron source

The hot graphite cylinder and the insulation are encapsulated in two vessels made of Zircaloy 4. The inner vessel is filled with neon gas at a pressure of 200 mbar and the small space between both vessels is filled with helium gas at 3 bar. The pressure value in both vessels is permanently monitored. Hence there are two barriers, which safely separate the hot graphite from air and water. For supervision in particular with respect to safety aspects, the HNS is equipped with 36 chromel/alumel thermocouples, which measure the surface temperature of the inner vessel. Further design details of the hot neutron source can be found in [2].

As major parameter from the user's perspective the temperature of the hot graphite cylinder is monitored permanently. Due to the extremely harsh environmental conditions as high temperature, chemical reactivity with graphite and nuclear radiation, the temperature of the graphite cylinder is very difficult to measure. Thermocouples, resistance thermometers, and pyrometers are not suitable for this purpose because they drift in an unpredictable way. Therefore a purpose-built noise thermometer that is hardly affected by radiation and/or temperature induced changes of the sensor was installed to measure the temperature of the hot graphite cylinder (Fig. 1).

2. Noise Thermometer

A noise thermometer is a contact thermometer. It measures the noise voltage of a passive electrical resistor and determines its absolute temperature. The basis of this measuring technique is a phenomenon which has for the first time been described by H. Nyquist and J. B. Johnson in 1928. They published [3] [4] that the mean square voltage fluctuating around zero at the clamps of a passive, unloaded resistor is directly proportional to the absolute temperature.

$$\overline{U^2} = 4 \times k_b \times T \times R \times \Delta f$$

$\overline{U^2}$ mean square of noise voltage
k_b Boltzmann constant
T absolute temperature
R noise resistor
Δf frequency interval

The absolute temperature of a noise resistor is determined from the above equation by measuring the mean square noise voltage $\overline{U^2}$ of a known noise resistor R in a frequency interval band Δf [5].

The noise sensor of the hot neutron source was developed by the Technische Universität München and tested successfully in a mock-up test before installation. Like the HNS itself it is manufactured from graphite and fixed in the graphite cylinder of the hot neutron source. The resistance of the noise sensor is 4 Ω at 20 °C and changes to 1.7 Ω at 2000 °C. This low resistance has been chosen because of the high temperatures that will force a decrease of the electric resistance of the insulation material. Consequently the noise voltage of the sensor is extremely low and has a value in the nanovolt-range at 2000 °C.

For the proper measurement of those low voltages a special equipment being based on the detection of the noise voltage by means of a correlation technique is used. The striking advantage of this technique is that all parasitic contributions to the noise voltage like the ones from cables, switches and preamplifier are eliminated [5]. The measurement system is made up of a preamplifier, a main unit and a PC. It was delivered from the Central Institute for Electronics (ZEL) of the Forschungszentrum Jülich in Germany [6].

The noise sensor and the measuring system are connected inside the hot neutron source via a 5.5 m long mineral isolated cable and outside the source via an 18.2 m long Kapton isolated cable. The cable is dual shielded and the wires are in a twisted-pair arrangement to avoid an influence of electromagnetic interferences on the small noise voltage.

The frequency interval of the measurement lies between 50 – 155 kHz.

3. Measurements using the noise thermometer

3.1 The temperature of the hot graphite cylinder

Already during nuclear commissioning of the FRM II in 2004, the temperature of the hot graphite cylinder was measured by the noise thermometer. Figure 2 shows the resulting temperatures as they had been recorded at different nuclear power steps and after thermal equilibrium within the HNS had been achieved.

It has to be noted that the temperature is not rising linear with reactor power. This effect is due the thermal conductivity of the insulation carbon fiber that changes with temperature. At full reactor power of the FRM II (20 MW) the graphite cylinder reaches a maximum temperature of 2030 °C.
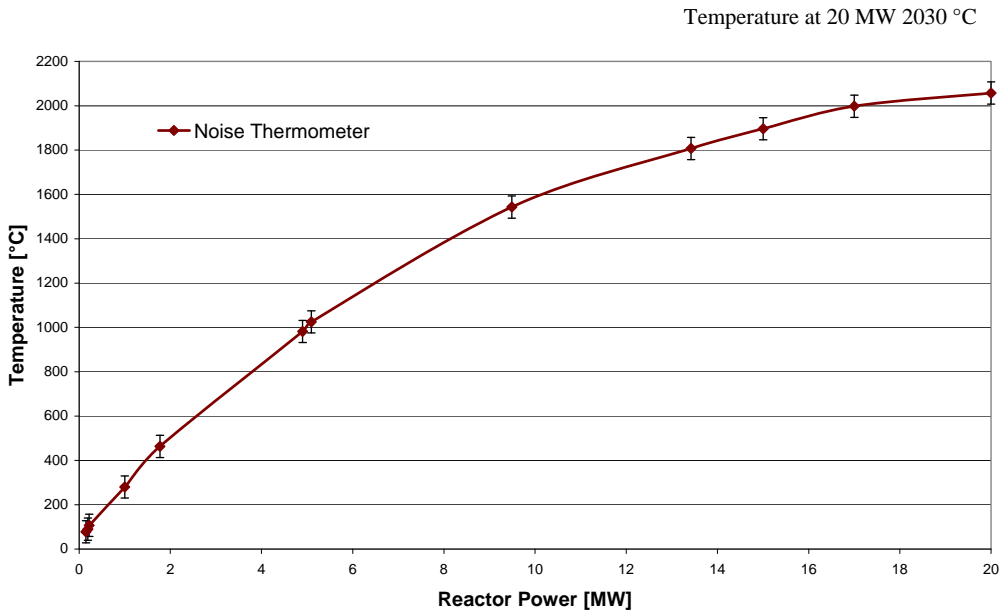
Fig. 2 Graphite temperature within the hot neutron source while nuclear commissioning

This maximum measured temperature of 2030 °C agrees sufficiently well with the precalculated temperature of 2200°C. The difference is attributed to the fact that the insulating carbon fibers were compressed strongly during manufacturing of the HNS in order to fix the graphite cylinder. Consequently the structure of carbon fibers was densified and its heat conductivity was increased as compared to the values used in the calculation.

3.2 Temperature transients of the hot graphite cylinder

Figure 3 will gives an idea how long the hot neutron source needs to reach the maximum temperature and how long it takes to cool down.

Fig. 3 Temperature transients of the hot moderator at 20 MW reactor power

When the FRM II reactor is run at full power (20 MW) the hot neutron source will reach the temperature of 2030 °C within 12 hours (red curve in Figure 3). The graphite cylinder has a weight of approximately 15 kg and consequently a considerable heat capacity. This fact explains the long time for the heating up.

After shut down of the reactor, the HNS cools down to room temperature over more than 20 h (blue curve).

3.3 Heat-Up effect of the HNS
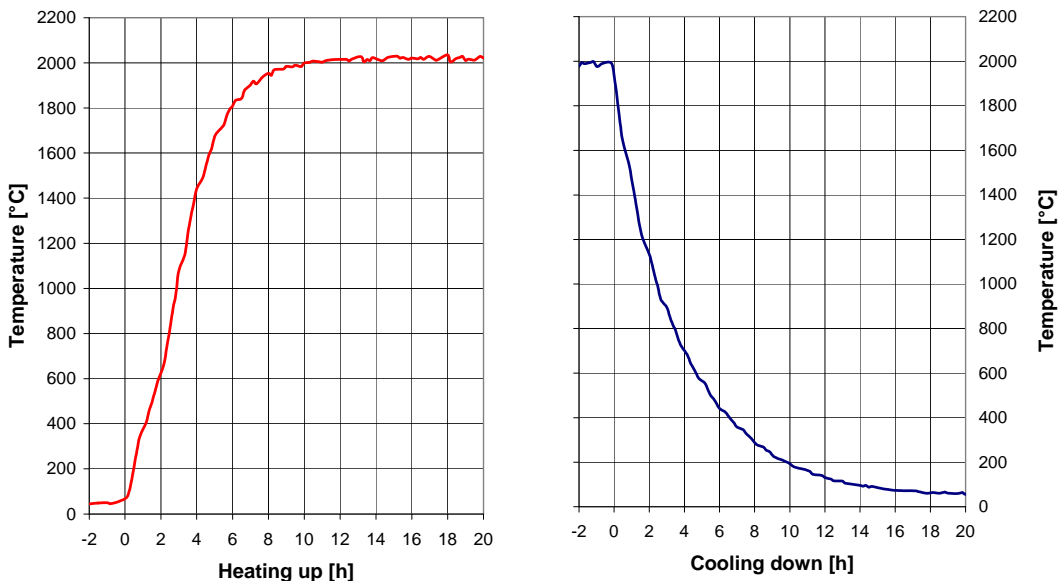
The long time needed to reach the maximum temperature was used to measure the heat-up effect of the HNS on the counting rate of superthermal neutrons exhibiting a wavelength of 0.552 Å in the connected beam tube (Fig. 4).
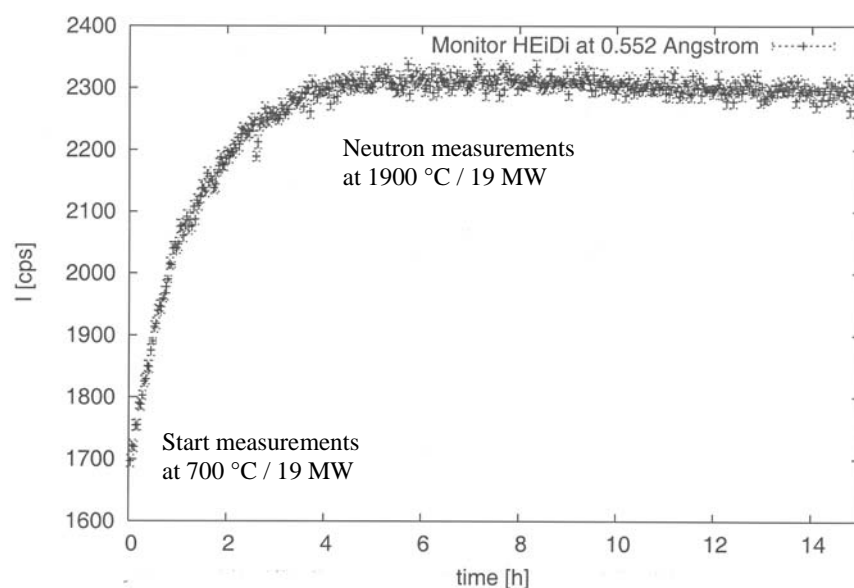
Fig. 4 Heat-Up effect of hot neutron source at 19 MW reactor power

At the start of the data taking the reactor power was increased to 19 MW and kept constant. The corresponding temperature of the hot moderator was 700 °C and the monitor counted 1700 neutrons per second.

Six hours later, the temperature had risen to 1900 °C and the corresponding neutron count rate to 2300 neutrons per second. Fig. 4 demonstrates that the HNS at FRM II works as envisaged and provides a considerably increased flux density of superthermal neutrons to the scientific users of FRM II.

4. Conclusion

The noise thermometer at the FRM II hot neutron source is now being operated for a period of more than six years. In this time the temperature of the graphite cylinder had been monitored continuously. It turned out that the equilibrium temperature of the cylinder has dropped around approx. 100 °C during this period. The most probable reason for this observation is the ongoing densification of the graphite fiber insulation.

In summary, the feasibility of an exact measurement of high temperatures in the very harsh environment of a nuclear research reactor even on a long term scale has been proven by the noise thermometer of the HNS. Besides other parameters and its value for the scientific user the temperature of the graphite cylinder measured by the noise thermometer gives the in-house staff of FRM II important information on the proper operation of the hot neutron source.

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FUEL MANAGEMENT
AT THE DALAT NUCLEAR RESEARCH REACTOR

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ABSTRACT

The Dalat Nuclear Research Reactor (DNRR) is a pool type research reactor which was reconstructed in 1982 from the old 250 kW TRIGA-MARK II reactor. The spent fuel storage was newly designed and installed in the place of the old thermalizing column for biological irradiation. The core was loaded by Russian WWR-M2 fuel assemblies (FAs) with 36% enrichment. The reconstructed reactor reached its initial criticality in November 1983 and attained it nominal power of 500 kW in February 1984. The first fuel reloading was executed in April 1994 after more than 10 years of operation with 89 highly enriched uranium (HEU) FAs. The third fuel reloading by shuffling of HEU FAs was executed in June 2004. After the shuffling the working configuration of reactor core kept unchanged of 104 HEU FAs. The fourth fuel reloading was executed in November 2006. The 2 new HEU FAs were loaded in the core periphery, at previous locations of wet irradiation channel and dry irradiation channel. After reloading the working configuration of reactor core consisted of 106 HEU FAs. Contracts for reactor core conversion between USA, Russia, Vietnam and the International Atomic Energy Agency for Nuclear fuel manufacture and supply for DNRR and Return of Russian-origin non-irradiated highly enriched uranium fuel to the Russian Federation have been realized in 2007. According to the results of design and safety analyses performed by the joint study between RERTR Program at Argonne National Laboratory and Vietnam Atomic Energy Institute the mixed core configurations of irradiated HEU and new low enriched uranium (LEU) FAs has been created on 12 September, 2007 and on 20 July, 2009. After reloading in 2009, the 14 HEU FAs with highest burnup were removed from the core and put in the interim storage in reactor pool. The works on full core conversion for the DNRR are being realized in cooperation with the organizations, DOE and IAEA. Contract for Nuclear fuel manufacture and supply of 66 LEU FAs for DNRR among JSC TVEL, Moscow, Russia and Vietnam Atomic Energy Institute and Battelle Enery Alliance, LLC, Idaho Falls, USA has been realized. The plan for realization of full core configuration of LEU fuel is planned. In the plan the first working core with 92 fresh LEU FAs will be created. This paper presents the fuel management at the DNRR.

1.   Introduction

The DNRR is a pool type research reactor which was reconstructed from the 250 kW TRIGA-MARK II reactor. During reconstruction, some structures of the former reactor such as the reactor aluminium tank, the graphite reflector, the thermal column, the horizontal beam tubes and the radiation concrete shielding were retained. The reactor core, the control and instrumentation system, the primary and secondary cooling systems, the spent fuel storage as well as other associated systems were newly designed and installed [1]. The core was loaded with VVR-M2 fuel assemblies with 36% enrichment. The reconstructed reactor achieved first criticality in November 1983 with 69 fuel assemblies and attained its planed nominal power of 500 kW in February 1984. The vertical section of the reactor is shown in Figure 1. On the supporting base located at the end of upper cylindrical shell in reactor pool

there are 24 piercing holes provided to hold 24 pits for interim storage of irradiated or spent fuel assemblies as well as of other objects removed from the core. Each pit can contain 3 fuel assemblies placed vertically. This interim storage is capable of containing 72 fuel assemblies. Next to the reactor pool in the same concrete shield structure, there is spent fuel storage. It was the old bulk-shielding experimental tank, kept from the former TRIGA reactor. For the present reactor, this tank is coated with stainless steel and filled with demineralised water. The capacity of the spent fuel storage is to contain 300 fuel assemblies. A 2.5-ton transportation cask is provided to contain a spent fuel assembly for transferring it from the interim storage to the spent fuel storage.
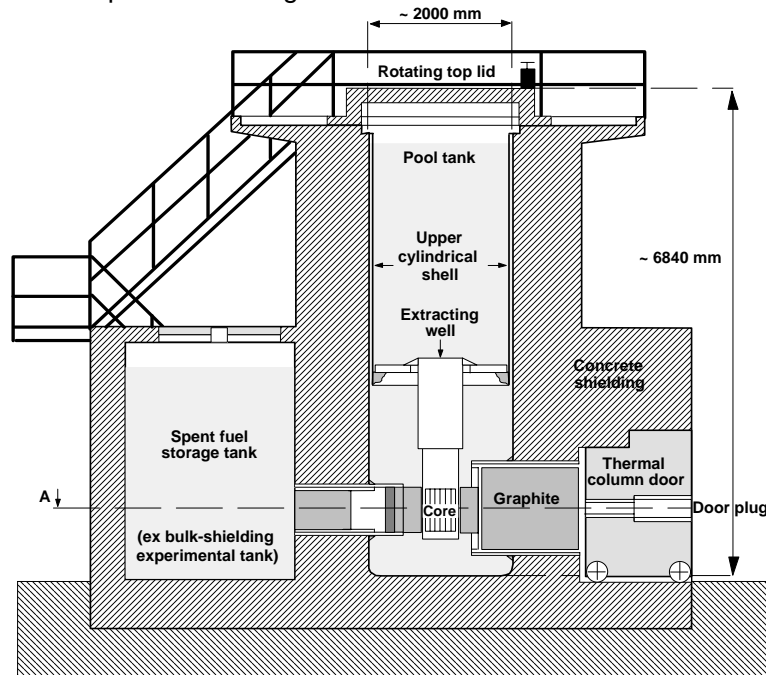

Fig 1. Vertical section view of the reactor

Contracts for reactor core conversion between USA, Russia, Vietnam and the International Atomic Energy Agency for Nuclear fuel manufacture and supply for DNRR and Return of Russian-origin non-irradiated highly enriched uranium fuel to the Russian Federation have been realized in 2007. The mixed core configurations of irradiated HEU and new LEU FAs has been created. The works on full core conversion for the DNRR are being realized in cooperation with the organizations, DOE and IAEA. The plan for realization of core configuration of only LEU FAs is planned in the middle of 2011.

2. HEU Core management of the Dalat Nuclear Research Reactor

The first fuel reloading was executed in April 1994 after more than 10 years of operation with 89 HEU FAs. The 11 new HEU FAs were added in the core periphery, at previous beryllium element locations. After reloading the working configuration of reactor core consisted of 100 HEU FAs. Second reloading for Dalat Nuclear Research Reactor was realized in March 2002. The 4 new HEU FAs were added in the core periphery, at previous beryllium element locations. After reloading the working configuration of reactor core consisted of 104 HEU FAs [2]. The third fuel reloading by shuffling of HEU FAs was executed in June 2004. The shuffling of 16 HEU FAs with highest burn up in the centre of the core and 16 HEU FAs with low burn up in the core periphery was done. The working configuration of reactor core kept unchanged of 104 HEU FAs. The core configuration of DNRR with 104 HEU FAs is shown in Figure 2. In waiting for execution of fuel conversion and saving of fresh HEU FAs, the fourth fuel reloading was executed in November 2006. Only 2 new HEU FAs were loaded in the core periphery, at previous locations of wet irradiation channel and dry irradiation channel. After reloading the working configuration of reactor core consisted of 106 HEU FAs.

Fig 2. Core configuration of DNRR with 104 HEU FAs

3.   Realization of mixed Core for the Dalat Nuclear Research Reactor

Contracts for reactor core conversion between USA, Russia, Vietnam and the International Atomic Energy Agency for Nuclear fuel manufacture and supply for DNRR and Return of Russian-origin non-irradiated highly enriched uranium fuel to the Russian Federation have been realized in 2007. The 35 fresh HEU FAs were sent back to Russian Federation. The 36 new LEU FAs from Russian Federation have been received. The characteristics of HEU and LEU FAs is shown in Table 1.

Parameter	VVR-M2 HEU	VVR-M2 LEU
Enrichment, %	36	19.75
Average Mass of 235U in FA, g	40.20	49.70
Fuel Meat Composition	U-Al Alloy	UO2+Al
Uranium Density of Fuel Meat, g/cm^3	1.40	2.50
Cladding Material	Al alloy	Al alloy
Fuel Element Thickness (Fuel Meat and Cladding), mm	2.50	2.50
Fuel Meat Thickness, mm	0.70	0.94
Cladding Thickness, mm	0.90	0.78

Tab 1: The characteristics of HEU and LEU FAs

According to the results of design and safety analyses performed by the joint study between RERTR Program at Argonne National Laboratory and Vietnam Atomic Energy Commission

the mixed core configurations of irradiated HEU and new LEU FAs has been created [3]. The first mixed core configuration has been created on 12 September, 2007. The 8 HEU FAs with highest burnup were removed from the core periphery positions. The 8 HEU FAs from second ring counted from neutron trap were moved to the core periphery positions. The 2 HEU FAs from locations of wet irradiation channel and dry irradiation channel were moved to 2 positions in second ring. The 6 new LEU FAs were added in 6 positions in second ring. The 2 wet irradiation channels were added in 2 positions of previous locations of wet irradiation channel and dry irradiation channel. After reloading th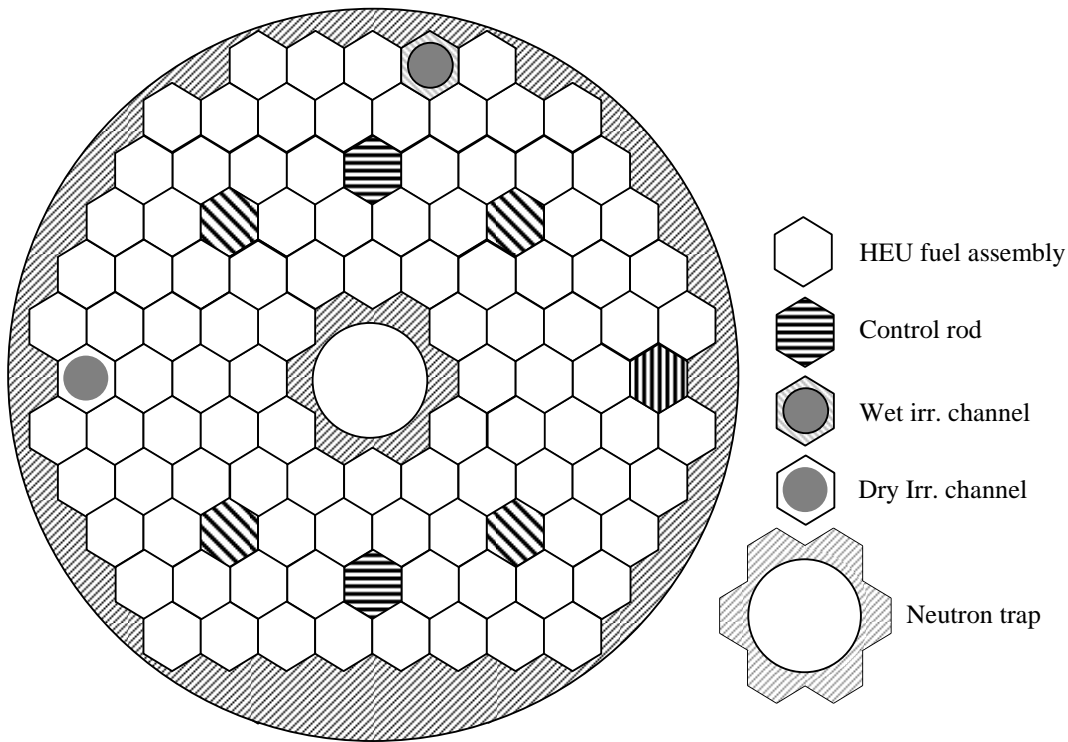e working configuration of reactor core consisted of 104 FAs (98 HEU FAs and 6 new LEU FAs). We had first 8 spent HEU FAs. The second mixed core configuration has been created on 20 July, 2009. The 6 HEU FAs with highest burnup were removed from the core periphery positions. The 6 HEU FAs from second ring counted from neutron trap were moved to the core periphery positions. The 6 new LEU FAs were added in 6 positions in second ring. After reloading the working configuration of reactor core consisted of 104 FAs. The working core configuration of DNRR with 12 LEU FAs and 92 HEU FAs is shown in Figure 3.
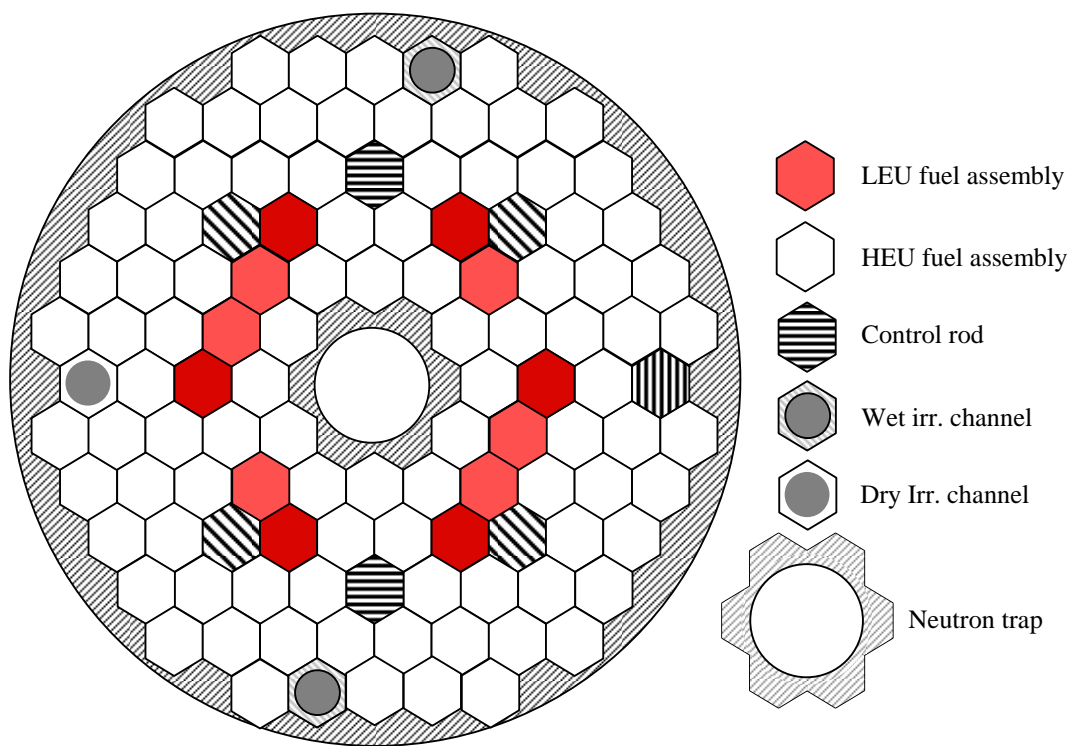
Fig 3. Working core configuration of DNRR with 12 LEU FAs and 92 HEU FAs

4. Full core conversion status of the Dalat Nuclear Research Reactor

The works on full core conversion for the DNRR are being realized in cooperation with the organizations, DOE and IAEA. The LEU core design commenced by establishing several possible cores and roughly analyzing to get some main safety and utilization characteristics such as shutdown margin, radial power peaking factors, and neutron performance at some irradiation positions. Without neutron trap, criticality will be achieved with 66 fresh LEU VVR-M2 FA, 13 beryllium rods and 34 aluminium choke rods. Neutronics steady-state thermal-hydraulic and safety analyses for Dalat Nuclear Research Reactor show that with a change in arrangement of Be rods, the main features of 92 LEU VVR-M2 FA cores are equivalent to those of HEU and current mixed fuel cores. Neutronic and thermal hydraulic calculation results can be seen in reference [4]. Figure 4 shows planned working core configuration of DNRR with 92 LEU FAs.

Contract for nuclear fuel manufacture and supply for Dalat Nuclear Reactor among JSC TVEL, Russia and Vietnam Atomic Energy Institute, Vietnam and Battelle Energy Alliance, LLC, USA have been realized in fourth quarter of 2010. The 66 new LEU FAs from Russian Federation have been received. The preparing works for shipment of HEU irradiated FAs to Russia are going. Restart-up and commissioning plan with full LEU core in realization of full core conversion is planned in the middle of 2011. Return of Russian irradiated highly enriched uranium fuel to the Russian Federation is proposed in 2013.
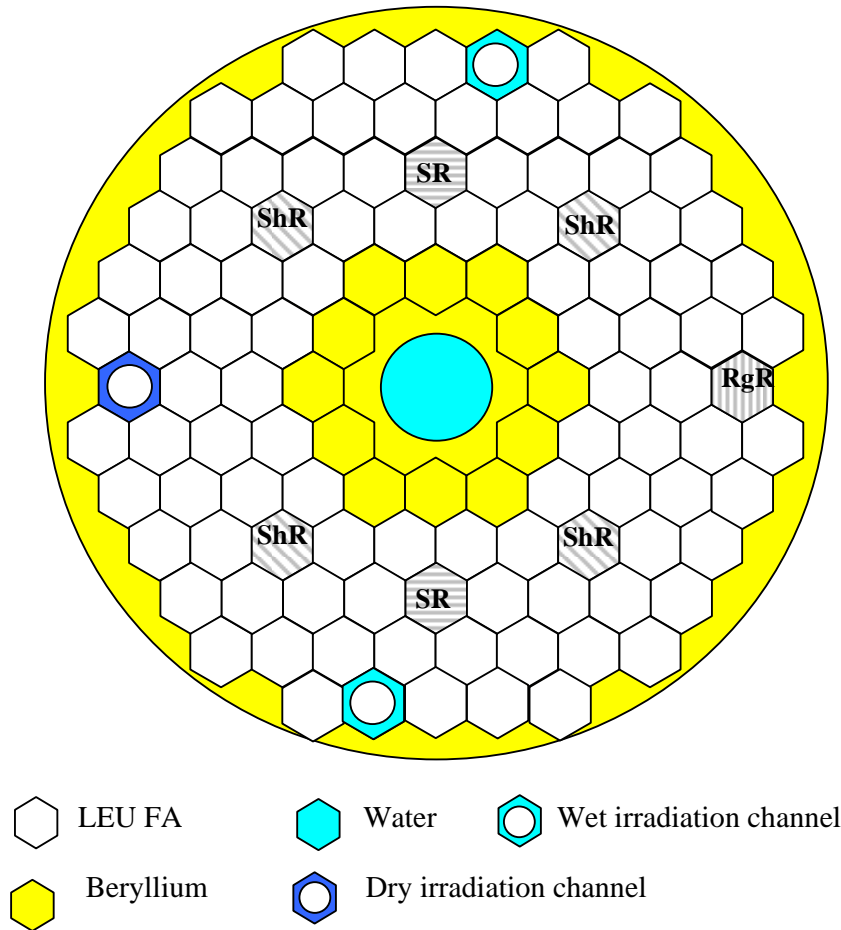
Fig 4. Planned working core configuration of DNRR with 92 LEU FAs

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A ROADMAP FOR MATERIALS REMOVAL AND DISPOSITION THROUGH THE GTRI GAP MATERIALS PROGRAM

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The U.S. Department of Energy (DOE), National Nuclear Security Administration (NNSA) Global Threat Reduction Initiative (GTRI) mission is to to identify, secure, remove and/or facilitate the disposition of high risk vulnerable nuclear and radiological materials around the world. Recently, the DOE NNSA-GTRI has identified a category of material in foreign countries that is not covered by the current programs and policies for the repatriation of the U.S. origin and Russian origin HEU fresh and spent fuel and yet can pose a potential threat to nonproliferation goals because of inadequate disposition options. This category of materials, termed as Gap Materials, includes certain fresh and spent HEU nuclear fuel and plutonium materials. This paper focuses on the project execution steps in Gap Materials removal. It provides the roadmap to Gap Materials removal typically consisting of concurrent phases of Materials Assessment, Materials Characterization, Container/Package Certification, Regulatory Approvals and Mission Execution.

Background

The U.S. Department of Energy (DOE), National Nuclear Security Administration (NNSA) Global Threat Reduction Initiative (GTRI) mission is to reduce and protect vulnerable nuclear and radiological materials located at civilian sites worldwide. GTRI has the goals of: (1) converting reactors from using weapons usable highly enriched uranium (HEU) to low enriched uranium (LEU), (2) removing or disposing of WMD-usable excess nuclear and radiological materials, and (3) protecting at-risk WMD-usable nuclear and radiological materials from theft and sabotage. The GTRI removal programs have focused on the repatriation of U.S. origin HEU and LEU fresh and spent nuclear fuel to the U.S. and the repatriation of Russian origin HEU fresh and spent nuclear fuel to Russia. Recently, the NNSA-GTRI has identified a category of material in foreign countries that are not covered by other removal efforts and can also present a potential threat to nonproliferation goals because of inadequate disposition options. This category of materials, termed as Gap Materials, includes fresh and spent HEU nuclear fuel and plutonium materials. This paper will summarize the steps involved in materials removal and transfer to the U.S. under the NNSA-GTRI Gap Materials program and provide the detailed roadmap of the activities necessary to facilitate Gap Materials removal.

Gap Materials Program

The U.S. DOE NNSA GTRI's U.S. origin and the Russian origin HEU fresh and spent materials removal programs are very mature and have successfully repatriated approximately 91% and 66% of the identified HEU materials to U.S. and Russia respectively and continue to maintain an aggressive schedule. The Gap Material program on the other hand is at its infancy. Gap material includes: (1) Non-U.S.-origin and U.S.-origin fresh HEU not covered by existing programs or policies; (2) Spent Nuclear Fuel (SNF) containing non-U.S.-origin HEU; (3) SNF containing U.S.-origin HEU that was not previously addressed in the Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Acceptance Program Environmental Impact Statement (EIS), and (4) separated weapons-usable plutonium.

Since 2005, GTRI has worked towards obtaining the necessary approvals associated with the necessary environmental and legal reviews per the National Environmental Policy Act (NEPA) of the United States to facilitate the return of certain Gap Materials to the U.S.. The approval to return additional forms of U.S. origin and non-U.S. origin HEU materials to the U.S. was obtained in 2006. A record of decision was obtained under NEPA in 2009 to return non-U.S.-origin HEU spent fuel to the United States as well as U.S.-origin HEU spent fuel that was not previously addressed in the Foreign Research Reactor Spent Nuclear Fuel (FRR SNF) Acceptance Program Environmental Impact Statement (EIS) [Ref.1]. Most recently, approval was obtained in 2010 to return up to 100 kgs of separated plutonium materials.[Ref.2] In order for Gap HEU spent fuel and separated plutonium materials to be returned to the U.S. under these approvals, the materials would have to a) pose a threat to national security, b) be susceptible to use in an improvised nuclear device, c) present a high risk of terrorist threat and d) have no other reasonable pathway to assure security from theft or diversion. In all cases, the materials have to meet the U.S. receipt facility acceptance criteria.

Roadmap to Gap Materials Removal

The roadmap to Gap Materials removal typically consists of concurrent phases namely Materials Assessment, Materials Characterization, Container/Package Certification, Regulatory Approvals and Mission Execution.

Materials Assessment: The first step in initiating a Gap Remove Program project is to perform a technical assessment of the candidate materials inventory. Upon receipt of materials data from the facility of origin, the GTRI team works with the facility experts to assimilate the detailed characterization and storage configuration data in order to conduct a preliminary evaluation of the materials eligibility under the Gap Materials program. The assessment includes data mining from facility reports and facility process knowledge to assimilate characterization data for the candidate materials. The characterization data typically includes but is not limited to the following:

- *Inventory*: Number of items, element and isotope weights
- *Materials Form*: geometry, dimensions, weights, materials of construction
- *Fabrication Data*: accountability records, fabrication drawings, photographs
- *Process History*: manufacturing history, irradiation history, manufacturing records

- *Radiological*: Gamma, beta, deep dose, surface contamination of primary container
- *Current Storage Configuration*: current packaging/containment
- *Physical Condition*: Mechanical damage, corrosion damage,

The characterization data helps determine potential disposition options and the materials eligibility for return to the U.S. under the program. In cases where U.S. NEPA approvals do not currently exist, alternative packaging and disposition strategies may also be assessed.

Materials Removal/Disposal Planning: Assuming that the candidate Gap Materials meet the eligibility requirements for return to the U.S., a Remove Project can be initiated and a project scope of work and schedule developed. A Material Characterization document is developed. The packaging options are evaluated for the specific type and characteristics of the materials. Transportation modes are also identified and schedules for materials removal developed. Formal contracting protocols are implemented upon agreement on the packaging and removal process. Shipping container certification and transport modes will be subsequently finalized and regulatory approvals obtained to facilitate the removal project execution.

Materials Characterization Documentation: Following a positive assessment, a detailed Materials Characterization document will be assembled. The Material Characterization Document will help guide the handling, shipping, storage, and processing of materials. This document will be a collaborative effort between the shipping and receipt facilities. The Material Characterization Document includes description, dimensions, weights, history; references as a basis to drawings, manufacturer reports, facility usage reports. The document also provides the basis for Transportation, MC&A, Radiological, & Criticality Control Limits. It provides the necessary background/historical information for the identification of alternative disposition options.

Materials Stabilization and/or Packaging: The handling and packaging of materials for removal may be as simple as shipping in existing packages to complex processing schemes requiring stabilization and repackaging of the Gap Material. Packaging and transport of the Gap Materials may involve additional characterization, stabilization and analyses prior to packaging in order to meet transport requirements and/or receipt facility acceptance criteria.   Materials and content specific flow sheets are developed if necessary in conjunction with the facility of origin and the receipt facility. In some cases, the GTRI team works with the facility to guide potential modifications to the existing facility equipment in order to facilitate stabilization of the materials, e.g. conversion of HEU solutions into oxides or stabilization of plutonium materials to meet transport and receipt facility requirements.

Shipping Container/Package: Type B nuclear materials transportation packages utilized in the international transport of HEU fresh and spent nuclear fuel and plutonium materials are designed, constructed, maintained, loaded and sealed in accordance with performance standards recommended by the IAEA and as adopted and enforced by national competent authorities to consensus standards. After review by a national competent authority, e.g., U.S. NRC, a certificate of approval, sometimes called a Certificate of Compliance (CoC), is issued.   The Certificate details the permissible content for the package or cask and its functional parameters. Additionally, a Certificate of Competent Authority (CoCA) or Competent Authority Certificate (CAC) must be issued

in the country that will receive the material if not the country in which the CoC was issued. A CoCA must also be obtained for all other countries through which the material must transit.

The certification process for the transport of HEU fresh and spent fuel is quite mature and existing DOE and/or commercial packages have been routinely used for the U.S. origin and Russian origin HEU fresh and spent fuel return programs. These processes are readily adapted to the Gap Materials program needs. Examples of containers and/or casks used for fresh HEU include ES3-100 Type B and for HEU spent fuel include NAC-LWT, TN-BGC1, TN-MTR and Skoda casks etc. The certification process for international transport of plutonium materials to the U.S. on the other hand is at relative infancy. Plutonium materials are typically packaged in welded or non-welded containers which are then placed in the containment vessel within a transport package. Potential transport package options include the DOE-9975, Croft's Safekeg and TN-BGC1 packages. However the package will need to be certified by the competent regulatory authorities. It is very important for the above reasons to perform cask selection and begin the regulatory approval process as early as possible in order to allow time to address any issues that may arise.

Shipping: Development of a shipping plan includes consideration of modes of transportation namely ground, sea, or air and in many cases is developed in conjunction with commercial shipping companies selected by the shipper country/facility. Scheduling national or international shipments of radioactive spent nuclear fuel requires coordination of all involved of federal, state, and local government agencies from each affected country, including those with the shipping and receiving facilities, those included in the shipping route, and the shipping company. Routes and ports of exit / departure are identified. IAEA Category I, II, III shipments require different security levels and the security requirements follow INFCIRC 225, Rev. 5. Multiple shipments may be warranted depending because of availability of shipping packages or to reduce security levels.

Transfer of Gap Materials: Transfer and shipment of the materials will commence once the readiness of the shipping and receipt facilities is assured. It entails facility safety and technical reviews including validation of nuclear criticality safety of the facility, approvals of all technical procedures for on-loading, off-loading, storage, and handling, Emergency Management Assessment

Gap Materials Removal: A notional project execution roadmap for the idealized case is shown in Figure 1 for HEU, SNF and plutonium materials removal. The schedule builds on the above mentioned activities necessary to implement a Gap Materials removal project. The schedule may be as little as 12 to 18 months e.g. for fresh HEU materials removal to greater than 36 months for plutonium materials due to potential stabilization and special packaging needs.

Summary: The U.S. DOE NNSA-GTRI has initiated a new program for the removal of Gap Materials. Gap Materials include HEU fresh and spent fuel and plutonium materials that are outside the scope of existing GTRI materials removal programs and policies. The scope of Gap Materials and its eligibility for repatriation to the U.S. was delineated in recent NEPA documents. The specific steps and activities involved in the materials removal activities are outlined along with a notional schedule. As part of the Gap Materials program, fresh HEU materials could be removed in as little as 12 to 18 months

whereas spent fuel and plutonium materials would require longer times, of up to 36 months.

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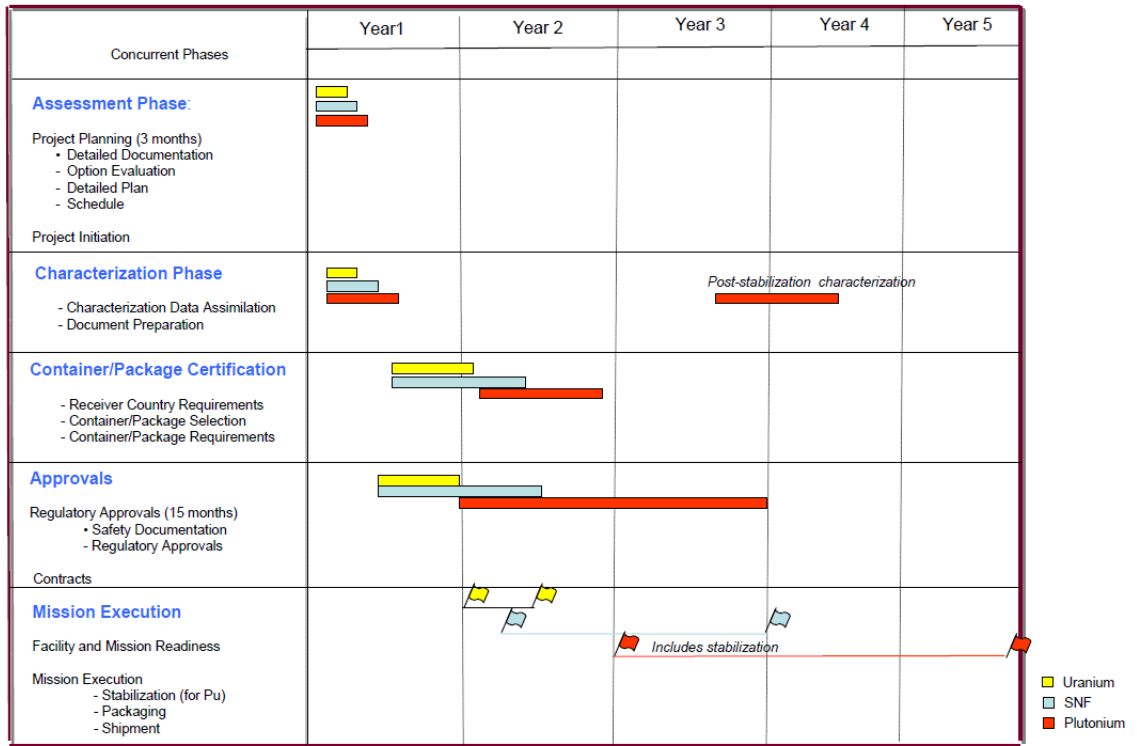
Figure 1: Notional roadmap of activities and schedule for Gap Materials Removal

CONVERSION OF REACTOR LVR-15 IN CZECH REPUBLIC FROM HEU TO LEU FUEL

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ABSTRACT

Accordingly to the IAEA recommendations and RERTR program, the LVR-15 reactor started the process of conversion from fuel enriched to 36 % to fuel enriched up to 20 % U^{235}. As the most suitable fuel for the reactor was chosen the IRT-4M fuel enriched to 19.7% U^{235}, fabricated in NZCHK Novosibirsk. The most important requirements, the fuel had to fulfill, were attainability, constructional continuity with the old type of FAs and operational experiences.

The conversion procedure began in January 2010 with testing irradiation of 3 IRT-4M FAs. Test irradiation took 9 reactor operation cycles. During this period were done visual inspection and sipping tests of FAs. An experiment with the aim to compare the influence of the fuel changing to neutron flux and the reactivity and to verify basic physical characteristics of the new fuel was performed at the start of test irradiation. The conversion itself is going on since January 2011. In this time, every cycle will be replaced 1 or 2 burned-up IRT-2M FAs with fresh IRT-4M FAs. This period will take 14 cycles and at the end should be in the core used only IRT-4M fuel.

1   Introduction

1.1   LVR-15 research reactor

The research reactor LVR-15 is situated in Rez and is operated by Research Reactor Division at Research Center Rez Ltd. The reactor LVR-15 is light water moderated and cooled tank reactor with forced cooling system and has maximal power of 10 MWth. The reactor is used as a multipurpose facility and it enables to perform research and services in the following areas:

- Material irradiation and testing in reactor loops and rigs,
- Production of radiation doped silicon,
- Regular production of radioisotopes for the radio pharmaceuticals and technical radiation
- sources,
- Irradiation devices for special irradiation,
- Pneumatic rabbit for activation analysis,
- Development of boron neutron capture therapy at the thermal column channel,
- Neutron physics research (e.g. neutron diffraction for different purposes, SANS) at reactor
- Horizontal channels.

Loop experiments and material testing in rigs belongs among the main research activities performed at LVR-15 reactor. Material research on the loops and rigs is mainly used for material embrittlement, corrosion, and material-coolant interaction studies that are very important for nuclear power plant utilization, plant life aging, and general safety.

The irradiation facilities are complemented with well-equipped hot cells that are also used for removing of irradiated specimens from experimental devices and their post-irradiation examination. More special post-irradiation examinations of irradiated specimens are performed in either hot or semi-hot cells situated in another building near the reactor.

1.2 Fuel and enrichment reduction

Since 1989 research reactor LVR-15 used fuel IRT-2M with 80% enrichment. During 1994 - 1995 fuel was converted to new one with 36% enrichment. Conversion to the fuel with 36% enrichment was determined by the fact that fuel of low-enriched uranium was not manufactured at that time in NZCHK Novosibirsk.

Fuel assemblies have shape of tubes with cubic cross section. At research reactor LVR-15 fuel assemblies with 4 or 3 tubes are used (in the 3-tubes FA control rods are positioned). Fuel elements are "sandwich" type meaning the layer in the middle is made of UO_2 and has aluminium cladding on both sides. Average burnup of IRT-2M FA is 60%. Parameters of the fuel are given in Table 1 and Figure 1, where also comparison with the new IRT-4M fuel is presented.

New fuel was already tested in the core of the reactor in 2010 and in the future the whole active core would be changed to low enriched fuel IRT-4M type.

Table 1 Comparison of IRT-2M and IRT-4M fuel assembles

	IRT-2M, 36% U-235	IRT-4M, 19.7% U-235
Fuel material	UO_2-Al	UO_2-Al
Cladding	SAV1 (min 97.1%)	SAV1 (min 97.1%)
Total length	882 mm	882 mm
Active length	590 mm	600±20mm
Total mass of the assembly 4/8 tubes	3.755 kg	6.0 kg
3/6 tubes	3.2 kg	5.2 kg
Mass of ^{235}U 4/8 tubes	230.0 g	300.0 g
3/6 tubes	198.0 g	263.8 g
Tube wall thickness	2 mm	1.6 mm
Cladding thickness	2 x min. 0.4 mm	2 x 0.3 mm
Section square - head	71.5 x 71.5 mm	71.5 x 71.5 mm
Section square	67.0 x 67.0 mm	69.6 x 69.6 mm
Fuel plate thickness	0.64 mm	0.70 mm

2

Figure 1 The IRT-2M fuel assembles with 4 tubes

In accordance with the IAEA recommendations and the RERTR initiative (Reduced Enrichment for Research and Test Reactors) reactor LVR-15 was prepared to change to fuel with the enrichment below 20%. In assessing the appropriate type of low-enriched fuel for future reactor operation thinking in terms of reactor design to the existing fuel types availability, it was chosen that the fuel from the same manufacturer will be used. The new fuel assembles IRT-4M were developed by Novosibirsk NZCHK with enrichment of 19.7% U235 and is presented in Figure 2. To maintain as far as possible equivalent neutron and physical characteristics of the new fuel it was necessary to substantially increase the total amount of uranium inside the fuel assemblies. The manufacturer has increased the number of fuel tubes in the FA to 8, respectively to 6 in the fuel assemblies with control rods, while maintaining the same external dimensions of the fuel assembly. An important advantage is that with the transition to new fuel there is no necessity to change the design of reactor or any instrumentation and control systems. Fissile material remained UO2. Fuel IRT-4M is also used in the VR-1 reactor at Czech Technical University in Prague, in the reactor in Tashkent and the reactor in Libya.
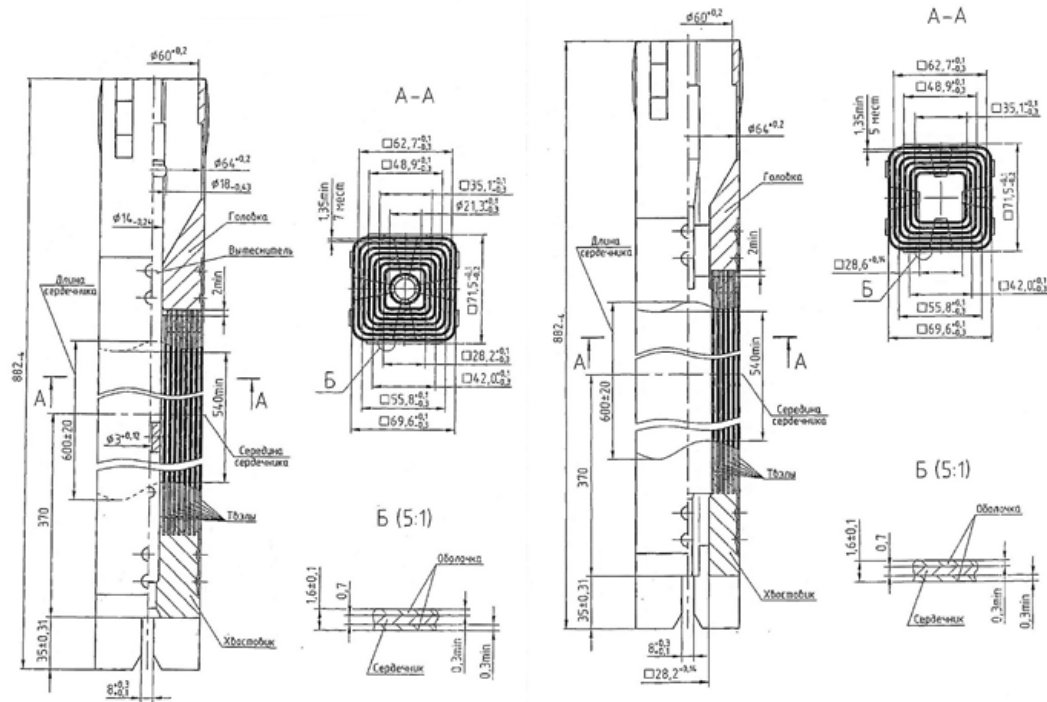
Figure 2 The new IRT-4M fuel assembles with 8 and 6 tubes

Conversion of the fuel at LVR-15 reactor to IRT-4M fuel will be blended with active core configurations in which they are currently used both IRT-2M and IRT-4M FAs. IRT-2M spent fuel assembles in the actual core will be gradually replaced by FA IRT-4M with enrichment of 19.7% until whole core is replaced. In order to obtain some operational experience with new fuel assemblies, three IRT-4M FAs were mixed into the core with IRT-2M fuel and were irradiated for 9 operating campaigns. During that period, all spent IRT-2M FA from the core were routinely replaced with the same IRT-2M fuel type. Only after evaluation of the three irradiated IRT-4M FA and irradiation tests in the core it will be possible to start with step by step replacement of spent IRT-2M FAs with the new IRT-4M FAs. Right now it is expected that the last old IRT-2M fuel element will be taken out from the core in 2012 and LVR-15 reactor will start its operation only with ITR-4M fuel assembles.

2   Testing of the three IRT-4M fuel assembles in the reactor core

Research reactor LVR-15 will go through sequence of mixed cores operation during the process of fuel changing where two types of fuel will be present. In these mixed cores IRT-2M fuel enriched to 36% will be gradually replaced with fuel assemblies of IRT-4M with an enrichment of 19.7%. IRT-2M elements which exceeded the allowed burnup will be replaced with adequate number of IRT-4M assembles after each operation cycle. During the 14 different reactor campaigns it is expected to create the core consisting of only low-enriched fuel elements. The implementation of the mixed cores is to be rigorously followed in order to respect all the specific characteristic of the new fuel assembles, especially in terms of its relative impact on the reactivity, the tendency of local peaks in the distribution of power and different hydraulic characteristics of the new fuel. The transition to new fuel through mixed cores is more economical, but takes much longer time than changing all the old fuel elements with new ones in a single step.

Prior to the gradual construction of mixed cores certain tests were carried out by irradiation of three IRT-4M fuel assemblies for 8 operational reactor campaigns. At that time three FAs type IRT-4M were inserted in the core and spent fuel assemblies IRT-2M during operation (the once which reached maximal burnup) were change to fresh IRT-2M with 36% enrichment. Irradiation was carried out 184 days at the reactor power 9-10 MW in the period 2-12/2010. The objective of this test was to gain experience with the new fuel in long-term operation and associated fuel handling. During the irradiation and reactor operation fuel leaks and FAs surface conditions were monitored carefully. New fuel elements reached 27% burnup. Since during the irradiation in the core no problems occurred, it was decided to continue in 2011 for the gradual replacement to low-enriched fuel.

In order to test new fuel assemblies in the core three IRT-2M FAs were replaced with new IRT-4M FAs which were positioned instead of old fuel (in the same positions inside the core). Core configuration was marked as K116 mix.01.01. and new FAs were in positions B6, F2 and G6. The purpose was to execute the series of experiments to verify the physical calculations and predictions of changes in neutron-physical parameters (spectra) of the core with a new type of fuel. Figure 3 presents a diagram of core configurations prepared to implement the program of experiments for the first irradiation of the IRT-4M fuel.

After irradiation of three IRT-4M FAs in positions B6, F2 and G6, these three assembles were moved to positions B6, C6 and C7 (see Figure 3). The experimental configuration was adopted to measure the size of neutron flux density and its spectrum in the fuel and surrounding positions during the 4 hours reactor operation at 5 MW. Measurements were performed by using activation foils which were evaluated in program STAYNL. Also the influence on the reactivity by FAs rotation was followed. Results are presented further in the text.
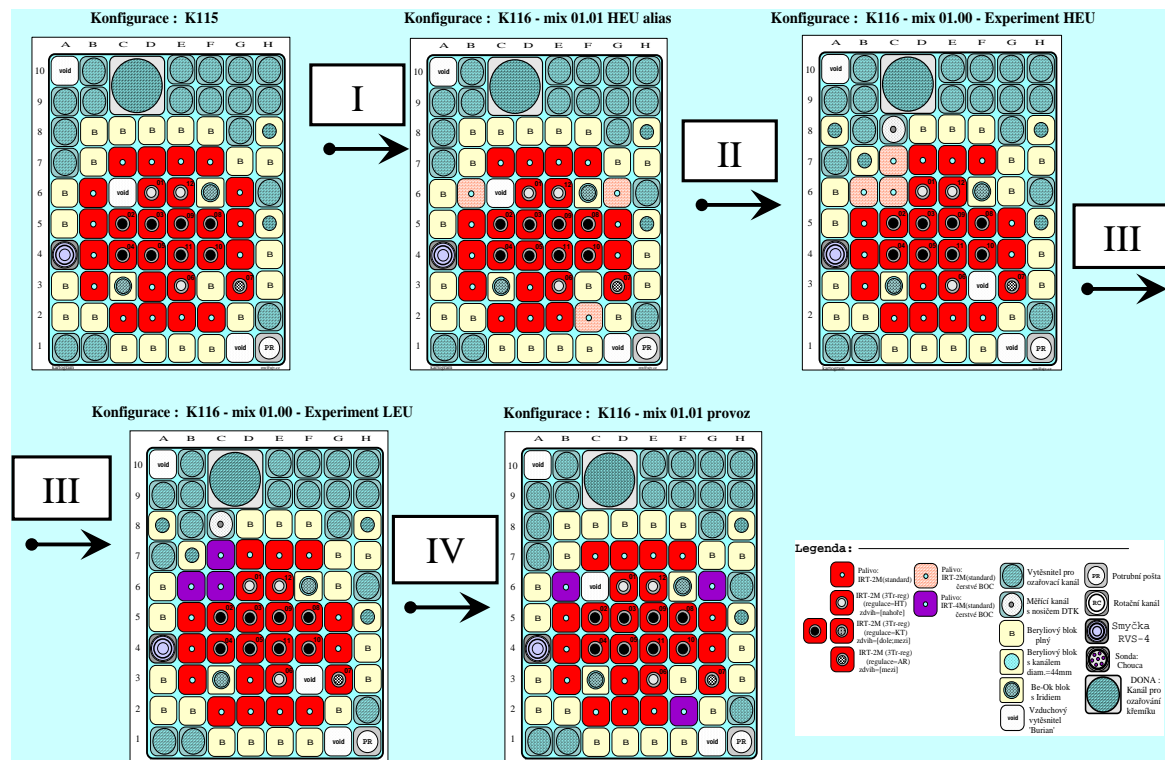

Figure 3 Sequence of configurations of the core for the first irradiation of low enriched uranium (LEU) fuel

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Experiment with low-enriched fuel IRT-4M showed 30% decrease of the neutron flux density of thermal neutrons in the fuel and about 10% around the fuel elements. On the contrary, there was a slight increase in epithermal neutron flux density. Results are presented in Table 2. Effect of high-and low-enriched fuels substitution in the core on reactivity was insignificant.

Table 2 Results obtained after irradiating three LEU FAs IRT-4M

Neutron energy	Position C6, Fuel			Position B7, Beryllium		
	IRT-2M	IRT-4M	IRT4M/IRT2M	IRT-2M	IRT-4M	IRT4M/IRT2M
(0.0, 0.5 eV)	5.32E+13	3.51E+13	0.66	7.89E+13	6.74E+13	0.85
(0.5 eV,10 KeV)	5.03E+13	4.72E+13	0.94	2.26E+13	2.23E+13	0.98
(10 keV, 0.132 MeV)	1.85E+13	1.78E+13	0.96	6.70E+12	6.66E+12	0.99
(0.132 MeV, 20 MeV)	7.77E+13	7.48E+13	0.96	2.40E+13	2.28E+13	0.95
Total fluence	2.00E+14	1.75E+14	0.88	1.32E+14	1.19E+14	0.90

Neutron energy	Position C8, Air			Position A8, Beryllium		
	IRT-2M	IRT-4M	IRT4M/IRT2M	IRT-2M	IRT-4M	IRT4M/IRT2M
(0.0, 0.5 eV)	4.55E+13	4.05E+13	0.89	2.73E+13	2.70E+13	0.99
(0.5 eV,10 KeV)	1.75E+13	1.79E+13	1.02	2.86E+12	3.01E+12	1.06
(10 keV, 0.132 MeV)	6.11E+12	5.49E+12	0.90	7.93E+11	8.80E+11	1.11
(0.132 MeV, 20 MeV)	2.29E+13	2.25E+13	0.98	2.63E+12	2.75E+12	1.05
Total fluence	9.20E+13	8.63E+13	0.94	3.36E+13	3.36E+13	1.00

Irradiation tests with three FAs IRT-4M confirmed the suitability of this fuel for reactor LVR-15. Some factors associated with the transition to LEU fuel IRT-4M should be followed in the future:

- Due to the small cladding thickness and small gaps between fuel assemblies special care during the fuel handling is needed.
- During the transition phase with mixed cores it is expected that the coolant flow will decrease as a consequence of higher hydraulic resistance of IRT-4M FAs. This will require some adjustments of security documentation and limits and conditions for LVR-15 operation.
- In the future experiments with LEU it will be necessary to take into account the reduction of thermal neutron flux density. This influence can be eliminated during the time by optimizing the design configuration for each campaign.

3   Conclusions

Following the good results of the tests with three LEU FAs, in the year 2011 it is planned to continue the conversion process at reactor LVR-15. As of January 2011 reactor operated with another two IRT-4M FAs which replaced two IRT-2M spent fuel assembles. This configuration has been marked as K126, mix 02.11 and is shown in Figure 4. In order to increase the neutron flux density in the centre of the core special central water and beryllium trap was positioned for Iridium samples irradiation (samples are made of highly enriched uranium for the production of Mo99). In March 2011 new 48 FAs will be delivered to Nuclear Centre Rez Ltd. from NZCHK Novosibirsk and conversion program will continue as planned.
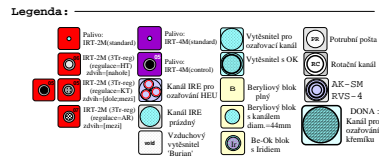
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Konfigurace: K126 mix 02.11
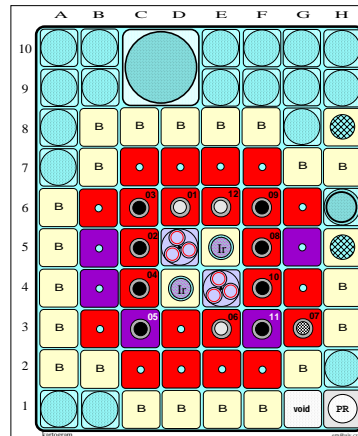
Figure 4 Reactor core configuration marked as K126 mix 02.11

Conversion to low enriched uranium at LVR-15 is carried out with the support of the Ministry of Industry and Commerce Czech Republic, DOE - U.S. Department of Energy and under the supervision of the State Office for Nuclear Safety.

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7

ACTIVITIES RELATED TO TRIGA 14 MW RESEARCH REACTOR UTILIZATION FOR ^{99}MO PRODUCTION TECHNOLOGY DEVELOPMENT.

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ABSTRACT
Institute decision to be involved in medical isotopes production using low enriched Uranium targets have been taken in the context of the need to increase research reactor utilization and in order to have a contribution at the improvement of medical care of the population.
Domestic efforts have been sustained by an existing infrastructure consisting of 14 MW TRIGA reactor, hot cell facility,   Radioactive Waste Treatment Facility and the Center for radioisotopes production in Bucharest. We had in view the following areas to carry out : target fabrication using imported metallic Uranium foil, target irradiation, target disassembly and dissolution, product recovery and purification, waste treatment. The main results obtained in developing technology for ^{99}Mo production are presented in the paper.
IAEA and ANL-DOE provided assistance in the frame of CRP Developing Techniques for Small Scale Indigenous Molybdenum-99 Production using Low Enriched Uranium (LEU) Fission or Neutron Activation.

1. Introduction

Initiation of the IAEA co-ordinated research project (CRP) "Development techniques for small scale indigenous Mo-99 production using LEU fission or neutron activation " during 2005 permitted to the Institute for Nuclear Research to become a contract holder. Participants to the CRP with US Department of Energy financial support and Argonne National Laboratory assistance developed techniques for fission Molybdenum–99 production based on LEU modified CINTICHEM process. The Agency's role in this field was to assist in the transfer and adaptation of existing technology in order to disseminate a technique, which advances international non-proliferation objectives promoting sustainable development needs while also could contributing to extend production capacity for addressing supply shortages from the last years.

The Institute intends to   develop the capability to respond to the domestic needs in cooperation with the IFINN – HH from Bucharest able to perform the last step consisting in the loading of fission Molybdenum on chromatography generators and dispensing to the final client.

Primary scope of the project is the development of the different technological steps and chemical processing steps in order to be able to cover the entire process for fission Molybdenum production at the required standard of purity.

Molybdenum production [1], [2] can be broken up into the following areas:

- Target fabrication and irradiation of the target in reactor at optimal conditions;
- Target dissolution in nitric acid and separation of ^{99}Mo using alpha benzoin oxima;
- Filtering and washing   for decontamination of traces amount of uranium and fission products;
- Running ^{99}Mo solution through a silver coated charcoal purification column and once more purification through resin column of silver coated charcoal, HZO and activate charcoal;
- Transportation of sodium molybdate from Institute to the Center for Radioisotopes Production in Bucharest for loading ^{99}Mo in their generators and   dispensing of the product.
- Waste treatment and disposal;

In parallel with actions deployment for fission Molybdenum –99 production, plans to explore n –gamma based Mo-99 production of high-specific activity making use of high epithermal neutron flux in TRIGA reactor are under way.

2. Target material and size

Modified CINTICHEM target is made [1] of a piece of low enriched Uranium foil covered with nickel foil, held and compressed between two aluminum cylinders having welded ends. The annulus target components have the following dimensions:
- concentric aluminum tubes have 165 mm length and diameters of 27.99 mm and 29.77 mm;
- uranium foil has 7 cm x 5 cm x 0.014 cm and weighs approximate 9.25 grams;
- coating Ni foil has 0.0015 cm thickness and weighs approximate 1.5 grams.

Fig.1 Target components

3. Target irradiation

The target is mounted into the irradiation device and loaded in TRIGA pool type reactor irradiation location. In the central cavity of the present irradiation device a number of maximum three annular targets can be loaded. The amount and the specific activity of ^{99}Mo obtained, depend on the target loading, target power and irradiation time. The un-perturbed neutron thermal flux in LEU core is $(2.7\text{-}2.9)\times10^{14}$ n/cm^{2}s . Approximate 40 Ci of Mo-99 and 4680 Ci of fission products per gram of U 19,75% enrichment are obtained at the end of 5 days irradiation at 10 MW reactor power.

Fig.2 Irradiation device and target holder with 3 annular targets

4. Neutronic calculations of the target

Key parameters for target irradiation are the power developed in the target fission process and temperatures on the target in thermal hydraulic conditions existing in irradiation device.
TRIGA neutron calculation system is based on the following codes;
-transport code cell – WIMS permitting to obtain microscopic cross sections of the nuclides, collapsed on 7 energy groups for different burn up steps.
-3DDT diffusion code supplemented with a burn up loop used for neutron flux distribution calculation and for fuel burn up calculation of the reactor. It is also used to calculate the power and reactivity induced by an experiment in TRIGA core. Based on these

codes and using the irradiation device model from the fig. 3 have been determined effective macroscopic fission cross section of Uranium (Σ_f=3.57 cm^{-1}), calculated linear power and reactivity change associated with Uranium target inside of the irradiation device introduced in the G7 core location. This is the most powerful irradiation position in our reactor.
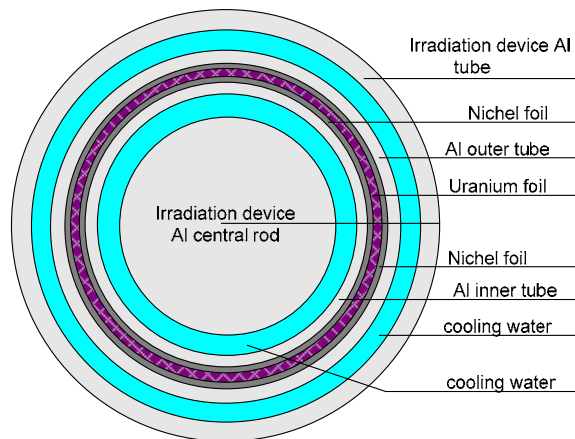

Fig. 3 Cross section through the irradiation device containing the target

Power density in Uranium foil is ($\phi_{thermal}$= 1.6x10^{14}n/cm^2.s, G_f=180 MeV/fission):

q$^{'''}$ =Φ_{th} Σ_f G_f = 1.6x10^{14}x3.57x180=1.028x10^{17} MeV/s.cm^3 = 16.448 kW/ cm^3
P_{target}=8060 W/target;

Positive reactivity induced by irradiation device and target is 0.43$ and this is determined mainly by Aluminium water removal and much less by Uranium foil.

5. Thermal hydraulic assessment of the target and irradiation device.

Heat transfer computations have been based on the following assumptions [3]:
1. The full power developed by the Uranium target is 8.0 KW;
2. Uranium foil covered with Ni foil is tight pressed between the two aluminum cylinders and is considered with no gap. However, because Uranium foil thickness proved to be non-uniform [4], calculation were done taking into consideration a hypothetic uniform gap between Ni foil and Uranium foil of 0.003 mm and also of 0.005 mm.
3. Heat generated in metallic uranium foil is propagating through conduction in nickel foils, gaps and aluminum tubes and through forced convection to cooling agent circulating in inner aluminum tube and around outer aluminum tube;
4. The cooling fluid is light water (H_2O) at the temperature of 35 °C and gauge pressure of 78000 Pa, which correspond to a water column of approx. 8 m.
The water flow through irradiation device containing target is equal with 1.9E+03 m^3/s and is circulating through the two channels of the annular target   with the same velocity;
The main goal was the evaluation of flow thermal-hydraulic characteristics in the annular flow area around target in the irradiation device, see Fig. 3. Temperature field was calculated using FLUENT code.
A gap must be maintained between foil edges to permit a longitudinal cut of the outer Aluminum tube in order to recover irradiated uranium foil for chemical processing. From thermal point of view the unheated gap region provide asymmetry and temperature gradient which was not evaluated. Turbulence   model   was considered to be suitable for flow conditions. These conditions are summarized:
- inlet average fluid velocity: 3.88 m/s
- Reynolds numbers for annular flow cooling channels are: Re1=10508.71, Re2=8850.335
- turbulence Intensity for annular flow cooling channels are: I1=5.0%   , I2=5.1%
- operating Gauge pressure: P=78000 Pa

The following temperatures are very important for target irradiation in the cooling condition from irradiation device from safety point of view:
   -temperature on the Aluminum target inner surface;
   -temperature on the Aluminum target outer surface;
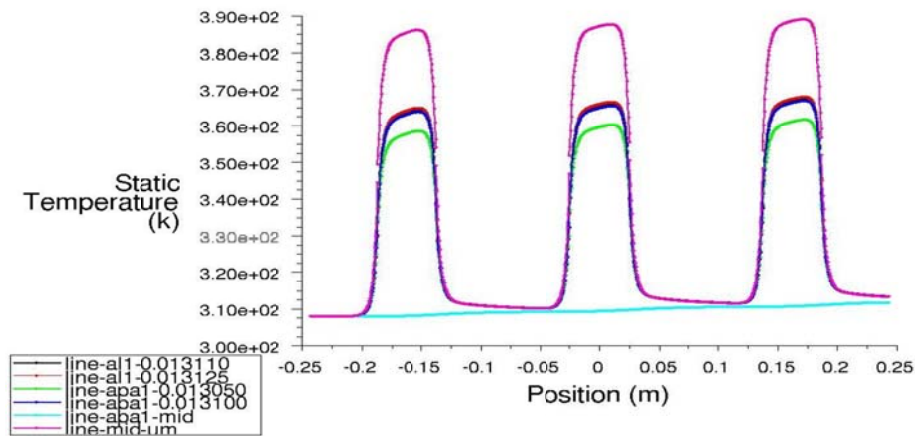   -Uranium foil central temperature;

Fig.4 Temperatures at key locations (water, water-Al interface, uranium) – no gap case

All these parameters were evaluated starting from boundary conditions specified above. The analysis shows the influence of gap size on temperatures in uranium foil and near the interface between aluminum and water.

These figures depict the temperatures in the target as function on the longitudinal position. The origin corresponds to central position in the central annular target. Fig. 4 shows the temperatures in water, water-Al interface and uranium considering no gap between different interfaces. For example "line-al1-0.013125" represents temperature profile across a line situated at a distance of 0.013125 m from central axe of annular target (see fig.3). This line is near the interface Water–Al situated at 0.013105 m, 20 μm inside Al. The temperature profiles are quite uniform in cross-section of the annular target containing Uranium. However there is a slight temperature increase downwards water flow across section containing Uranium. The maximum temperature at aluminum water interface is 365 K for 8 kW of uniform heat dissipation. The temperature profile at a given cross-section of the annular target is quite uniform and does not spread towards the extreme ends of the aluminum tubes. This maximum temperature is well below the saturation temperature of water (388K) at the depth of approximately 8 m where the target are submerged. When the gap is present the maximum temperature at aluminum water interface   is practically unchanged - 365K, but the impact on uranium foil temperature is high. If in no gap case the maximum uranium temperature is 390K (117°C), in 3μm gap case maximum temperature increases to 525K (252°C) and 625 K (352°C) for 5μm gap. A uniform gap is not a credible situation because after foil inserting between two tubes outer and inner cylinders are drawn to obtain good thermal contact.

6. Isotope extraction in the Hot Cell Facility

In the vicinity of the reactor building there is the Hot Cell building containing two big concrete cells (receiving cell and examination cell). A   water filled channel provides a communication way between reactor tank and   Hot Cell, facilitating underwater transfer of irradiated samples after suitable cooling time from rector pool directly into the receiving hot cell. In this location, irradiated target will be disassembled, Uranium foil withdrawn and afterwards, transferred into radiochemical cell for chemical processing. To create the condition for chemical processing we had to design and execute modifications of an existing radiochemical cell so that to become a process dedicated cell.

7. Chemical processing of 8 g Uranium samples labeled with small amount or irradiated Uranium

Uranium foil recovered from the target will be dissolved in a stainless steel dissolver using nitric acid. A fission gas recovery system, provided with cryogenic traps will permit to recover radioactive iodine and xenon.

Experimental work was focused on traversing all steps involved in chemical processing using equipment designed and fabricated in institute. Before dedicated equipment to be introduced in hot cell is needed testing outside cell in "cold" and "semi cold" conditions using small amount of radioactive material (irradiated Uranium). We performed three such tests and finally we have been successfully almost in every aspect.

During this year, complete tests in hot condition will be done.

Fig.5 Images from dissolving and purification stages and final solution of sodium molybdate

8. Waste treatment and disposal

The treatment and elimination of gaseous waste is solved by an efficient off gas system. Noble gases and Iodine isotopes are trapped separately on special materials for decay before elimination in atmosphere.

Volume reduction of solid waste is carried out through cutting and subsequently, they are conditioned in special containers for disposal.

Liquids containing enriched Uranium will be stored. After 5 to10 years of cooling time, Uranium from liquid waste could be extracted and remaining waste treated and conditioned for storage in National Storage for Radioactive Waste (a former Uranium mine). Another option is to convert high radioactive waste solution in solid form and to use for disposal a similar way as for spent fuel resulted from nuclear power plant.

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ENHANCEMENT OF IRRADIATION CAPABILITIES OF MOLY TARGETS IN THE OSIRIS REACTOR

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ABSTRACT

MOLY targets containing high enriched uranium are irradiated in the core of the OSIRIS reactor (CEA/Saclay) in order to obtain enough activity for molybdenum-99 allowing the generation of technetium-99m used as a universal marker for medical diagnostics. This paper presents an overview of the irradiation of MOLY targets in the OSIRIS core and the recent studies performed to enhance irradiation capabilities of MOLY targets by using additional irradiation devices in the core.

1. Introduction

MOLY targets containing enriched uranium (with U-235 isotope), are irradiated in the core of the OSIRIS reactor in order to obtain enough activity for molybdenum-99 allowing the generation of technetium-99m (by radioactive decay of Mo-99). Tc-99m is a radioisotope used as a universal marker for medical diagnostics. The very short half-life of the Tc-99m (6 hours) makes it mandatory to produce it in situ starting from a generator containing Mo-99 with necessary and sufficient activity. The short half-life of the Mo-99 (66 hours) also implies a renewal of the generator every week. The production of Mo-99 is thus subjected to the same just-on-time requirements. It is carried out in a cooperation framework with other European reactors in order to ensure a regular supply of hospitals in Europe. During spring 2010, medicine had to face a great shortage of Mo-99 production, due to the simultaneous shut down of several research reactors throughout the world.

In this paper, we present an overview of the irradiation of MOLY targets in the OSIRIS reactor and its constraints on the reactor operation. Then, we present the neutron studies recently performed to enhance irradiation capabilities of MOLY targets by using additional irradiation devices in the core. Finally, we present the future steps to perform the irradiation of these additional devices in the OSIRIS core.

2. OSIRIS Reactor

OSIRIS is a material testing reactor located at the CEA-Saclay site and operational since 1966. It is a 70 MW_{th} pool type light water reactor with an open core [1]. The core is a compact unit ($60 \times 70 \times 70$ cm^3). The core vessel contains a centrally located rack with 56 cells (a square-lattice pitch of 8.74 cm) loaded with 38 standard fuel elements, 6 Control Elements (hafnium absorber in the upper part and fuel in the lower part) and up to 7 beryllium elements. The remaining cells are dedicated to experiments and equipped with water boxes, each one containing up to 4 experiment rigs. Three experimental cells are available with the 2T-core configuration (reference) in the positions 24, 44 and 64. The 3T-core configuration offers an additional experiment cell in the position 52 (see figure 1).

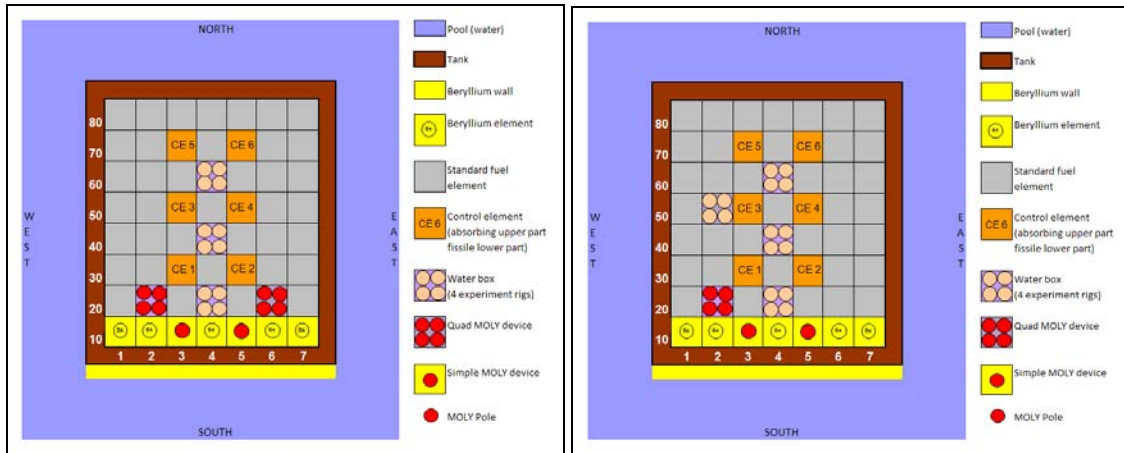[(*)] Corresponding Author (fadhel.malouch@cea.fr)

Fig 1. Horizontal cross-section of the OSIRIS core, on the left 2T-core configuration (reference) and on the right 3T-core configuration

3.   Irradiation of MOLY targets in the OSIRIS core

3.1   Irradiation devices

Tubular MOLY Targets are irradiated in the core of the OSIRIS reactor by group of three targets for about one week. The three targets are supported by a target-holder (called MOLY pole) inserted in an irradiation channel. The in-core irradiation of MOLY poles is carried out using (see figure 2):

− Simple MOLY devices (used since 1981), located in cells 13 and 15 (1 irradiation channel by device, 1 pole by channel),
− Quad MOLY devices (used since 1989), two being located in cells 22 and 26 for 2T-core configuration and only one in cell 22 for 3T-core configuration (4 irradiation channels by device, 1 pole by channel). In a Quad MOLY device, MOLY poles can be irradiated simultaneously and independently in the 4 irradiation channels of the device.
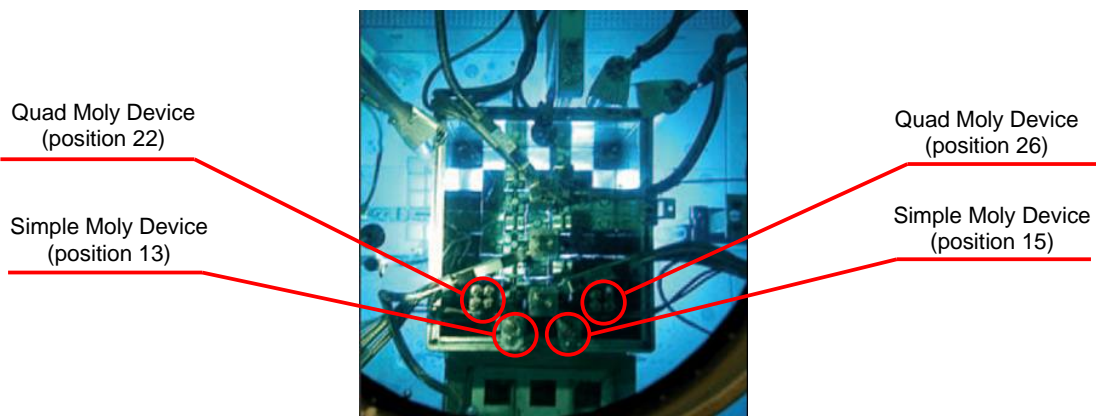

Fig 2. View of MOLY devices irradiated in the OSIRIS core (2T-core configuration)

3.2   Mo-99 production performance

As shown in figure 3, Mo-99 can be considered as a direct fission product of U-235 with a global fission yield taking into account those of all the short-lived precursors (such as yttrium-99, zirconium-99 and niobium-99).

Fig 3. Simplified chain of reactions for Mo-99 production and decay

According to this model, the activity of Mo-99 at the end of irradiation of a target can be approximated by the following formula:

$$A_{Mo99} \; = \; \frac{1}{k} \cdot \frac{N_A}{M_{U235}} \cdot \gamma_{Mo99} \cdot \sigma_{f,U235} \cdot \phi_{th} \cdot \left(1 - e^{-\frac{Ln2}{T_{1/2,Mo99}} \cdot T_{irr}} \right)$$

Where:
- A_{Mo99}: activity of Mo-99 at the end of irradiation of a target (Curie/gU-235$_{initial}$),
- k: unit conversion factor from Becquerel to Curie (3.7×10^{10} Bq/Ci),
- N_A: Avogadro constant (6.022×10^{23} atoms/mol),
- M_{U235}: molar mass of U-235 (235 g/mol),
- γ_{Mo99}: global fission yield of Mo-99 ($\approx$ 6%),
- $\sigma_{f,U235}$: thermal-neutron fission cross section of U-235 (581 barns = 5.81×10^{-22} cm^2),
- ϕ_{th}: thermal neutron flux in the target (n.cm^{-2}.s^{-1}),
- $T_{1/2,Mo99}$: half-life of Mo-99 (2.75 days),
- T_{irr}: irradiation time (in days).

The assumptions of this formula have been validated by neutron calculations using the APOLLO-2 2D Lattice code [2] and the DARWIN/PEPIN-2 punctual depletion code [3] developed by CEA (Saclay center). Figure 4 shows the variation in the Mo-99 activity at the end of irradiation of a target as a function of the thermal flux for different irradiation time.
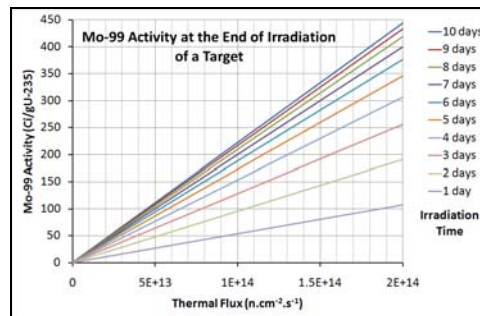
Fig 4. Mo-99 activity at the end of irradiation of a target

3.3 Irradiation constraints

There is no specific circuit used for the cooling of the MOLY targets irradiated in the OSIRIS core. This function is provided by the primary circuit of the core. So, the targets can be loaded and unloaded during the reactor operation. However, there is a limitation for the linear power of targets under irradiation (safety criterion). This condition has to be checked by neutron calculations (modelling several core configurations during the cycle) performed before each operating cycle. If this condition is not satisfied, the reactor power could be limited during all or part of the cycle to satisfy this requirement.

After irradiation in the OSIRIS core, the MOLY targets are stored first in the reactor pool and then in a water channel during the time necessary to the decreasing of the residual heat, before being transferred in one of the two hot cells of the facility, and then loaded by group of three targets in transportation casks (up to 4 casks a day are transported outside the facility). The total activity of the targets transported in a cask has to be lower than 2.1×10^{15} Bq (safety criterion). This condition is also checked by neutron calculations.

4. Enhancement of irradiation capabilities

4.1 Feedback of 2010

During first half of 2010, there was a world-wide shortage of technetium-99m because of a great lack of Mo-99 production, due to simultaneous unforeseen outages of two research reactors throughout the world, one in Canada (NRU), the other in Netherlands (HFR).

To face this shortage, OSIRIS schedule of operation was adapted to this exceptional situation:
- The renovation works of the facility initially planned for spring 2010 had been shifted for two months, with the agreement of the Nuclear Safety Authority, to avoid the simultaneous shutdown of OSIRIS, NRU and HFR,
- One or two additional Simple MOLY devices were used in row 10 of the core instead of beryllium elements to increase the number of irradiated targets,
- The 2T-core configuration was privileged to be able to load permanently a Quad MOLY device in the cell 26 of the core (this cell is loaded by a standard fuel element in the 3T-core configuration),
- The irradiation schedule of targets and the duration of the operating cycle were optimized in order to increase the number of irradiated targets by cycle.

The implementation of these measurements required coordination between the operators of the OSIRIS reactor and the different irradiation experiments. A high number of MOLY targets have already been irradiated in six months equivalent to a year production in normal periods. OSIRIS had thus answered temporarily about 20% of the Mo-99 world needs.

4.2 Recent neutron studies

In order to increase irradiation capabilities of MOLY targets in the OSIRIS reactor, new irradiation configurations of MOLY devices have been studied, in particular using one additional Quad MOLY device in cell 24 of the core instead of a water box. Thus, we can have Quad MOLY devices with a 3T-core configuration as much as using with the current 2T-core configuration (i.e. two Quad devices). With the 2T-core configuration, we could have simultaneously up to 3 Quad MOLY devices in the core, so 36% increase of channels available for the irradiation of MOLY targets.

In spite of the localization of cell 24 in row 20 of the core, the thermal neutron flux in this cell is still relatively higher than in cells 22 and 26. The linear power calculated for the MOLY targets in this cell often exceeds the authorized limit.

In order to minimize the power peaking in cell 24, the fuel loading of the core was adapted by refuelling high burn-up assemblies in the surrounding cells, in particular in cell 34 (north side of cell 24). This adaptation allows a local modification of the power distribution, and a decrease of the linear power in the targets, and so meeting the authorized limit (cf. figure 5).

However, with a MOLY-adapted fuel loading, the average fast neutron flux in the experimental cell 44 (north side of cell 34) becomes lower by about 15 to 20% (see figure 6). This result is less acceptable for the nuclear material damage experiments carried out in this cell, for which the irradiation duration depends on the fast neutron fluence. The fast flux remains appreciably the same in the experimental cell 64. There is a compromise to find between target irradiation in cell 24 and experiments in cell 44.

The neutron calculations related to these studies have been performed by using the OSIRIS core two-dimensional diffusion code developed by CEA (Saclay center).
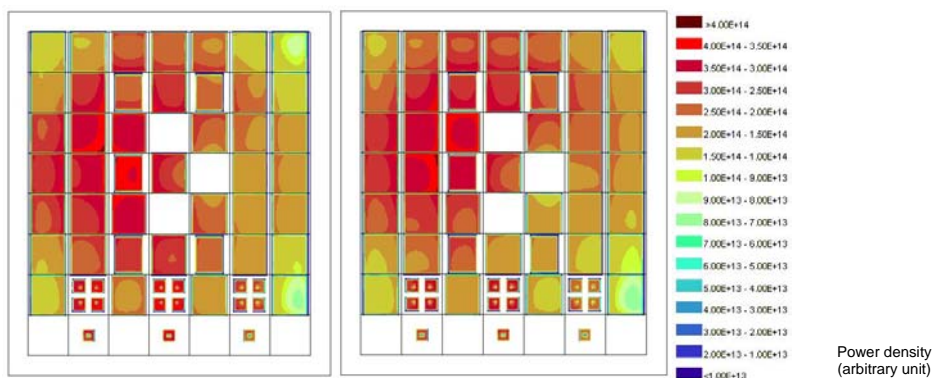
Fig 5. Power map with an additional Quad MOLY device in cell 24 (CE 4 inserted), using a standard fuel loading (on the left) and a MOLY-adapted fuel loading (on the right)
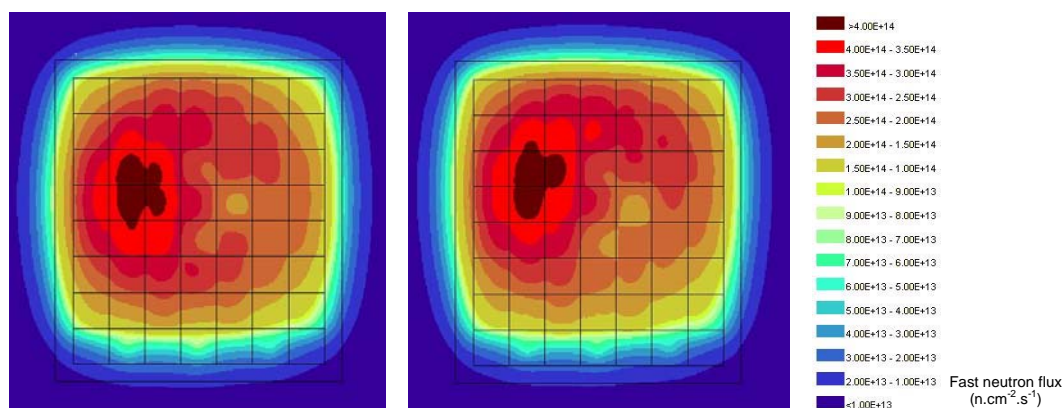
Fig 6. Fast-flux map with an additional Quad MOLY device in cell 24 (CE 4 inserted), using a standard fuel loading (on the left) and a MOLY-adapted fuel loading (on the right)

5.   Conclusion and prospects

The use of an additional QUAD MOLY device in the OSIRIS core will increase the current irradiation capabilities of MOLY targets by 36%. The manufacturing of this device and the related safety analysis are ongoing. It could be operational by the end of 2011.

For the future, the CEA works closely with its European partners NRG (Netherlands), SCK•CEN (Belgium) and the Technical University of Munich (Germany) to ensure an optimal coordination of the research reactors. In parallel, the CEA continues construction, on the site of Cadarache, of the Jules Horowitz research reactor (JHR). The JHR should gradually take over OSIRIS for the production of radioisotopes with medical use from 2015.

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DEVELOPMENT, QUALIFICATION, AND MANUFACTURING OF LEU-FOIL TARGETRY FOR THE PRODUCTION OF MO-99

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This paper provides an overview of the work in progress to develop low enriched uranium (LEU) foil targets for the production of the medical isotope molybdenum-99 (^{99}Mo). The development strategy utilizes analytical and numeric simulation methods, and in-situ testing to establish design rules, that include manufacturing constraints, for a safe and cost-effective ^{99}Mo production target.

A project team, sponsored by the U.S. Department of Energy - National Nuclear Security Administration, Office of Global Threat Reduction, has been formed to support the efforts of the large-scale ^{99}Mo producers to convert from highly enriched uranium (HEU)-based to LEU-based Mo-99 production. This work also directly supports the International Atomic Energy Agency's (IAEA)) International Working Group on Conversion Planning for Mo-99 Production Facilities from HEU to LEU, and the IAEA Coordinated Research Project (CRP); entitled "Developing Techniques for Small-Scale, Indigenous Production of Mo-99 Using Low-Enriched Uranium (LEU) or Neutron Activation." This IAEA sponsored project supports the utilization of research reactors worldwide by providing them the technology to develop and implement a viable radioisotope production capability that is consistent with non-proliferation objectives.

The project team is comprised of personnel representing Argonne National Laboratory (ANL), B&W Y-12, LLC, and the University of Missouri's College of Engineering. Additional support will be provided by the Institute of Nuclear Research - Pitesti Reactor Facility (Romania), and the Korean Atomic Energy Research Institute (KAERI).

The objectives of the project are to:

1. Establish a target qualification methodology that is bounding for all ^{99}Mo target irradiators.

2. Develop the target qualification methodology by building upon the annular LEU-foil target design work and testing previously performed by ANL and ANSTO/CERCA.

3. Develop a final product in the form of a "generic" LEU-foil target qualification document that can be used by any Mo-99 target irradiator to support their facility specific "safety case."

4. Evaluate the technical feasibility of developing, qualifying, and manufacturing LEU-foil targetry in two distinct geometries (annular and plate).

5. Develop a set of "universal" target material specifications and target manufacturing quality control (QC) test criteria that are acceptable to all current target irradiators and potential future ^{99}Mo producers.

6. Optimize the target design, considering both reactor and processing facility safety, and the economics of manufacturing a cost-effective target to offset the inherent economic disadvantage of using LEU in place of HEU.

In summary, this paper provides a review of analytic, numeric, and experimental activities that have been completed on annular and plate LEU-foil target geometries to date. Preliminary results do not identify any technical barriers to the development and qualification of an LEU-foil based target as a viable alternative to HEU or LEU dispersion type targets.

1. Introduction

Both HEU and LEU dispersion type targets have a well established pedigree as they are manufactured to industry accepted materials test reactor (MTR) fuel specifications. Furthermore, HEU dispersion type targetry has a long history of successful irradiation without any significant cladding failures. Historically, thousands of dispersion targets have been safely irradiated to produce fission product ^{99}Mo.

In comparison, LEU-foil targets have a very limited irradiation history. This target type has been successfully irradiated in Argentina, Indonesia, Australia, and the United States (University of Missouri Research Reactor). It is estimated that less than fifty (50) LEU-foil targets have been irradiated to date. Furthermore, foil type targets are not currently manufactured to an industry accepted standard or specification.

As a consequence, a manufacturing standard or specification must be developed for this target type and a corresponding target qualification program must be successfully implemented. A manufacturing specification must be established and a qualification program must be completed before this target type will be considered as an HEU to LEU target conversion option by any large-scale ^{99}Mo producer. Sections 3.0 and 4.0 summarize the activities currently in progress to support manufacturing specification and qualification program development..

2. LEU-Foil Targetry Advantages

An LEU-foil target has several distinct advantages in comparison to an LEU dispersion type target. These advantages are:

1. On a per target basis having the same uranium mass, the time to chemically dissolve a foil target is significantly less than that of a dispersion target. As the foil is removed from its aluminum cladding, only the foil component of the target is dissolved in the first stage of the ^{99}Mo production process.

2. On a per target basis having the same uranium mass, the volume of liquid radioactive waste generated during the target dissolution phase is significantly less because only the foil component of the target is dissolved. The LEU-foil target's aluminum cladding is removed during target disassembly, allowed to decay, and disposed of as low-activity solid radioactive waste.

3. The cost per gram of LEU-foil target material is less than the cost per gram of LEU dispersion target material.

4. On a per target basis, the cost to fabricate an LEU-foil target is expected to be less than the cost to fabricate a dispersion type target that has the same uranium mass.

5. The uranium loading of a typical LEU dispersion target is in the range of 2.5 to 3.0 g U/cm^3. The density of the LEU metal foil is approximately 19 g/cm^3. As a consequence, a foil target of the same geometry can contain much higher amounts of uranium than the HEU or LEU dispersion type target. The uranium content will be limited by the ability of the target's cooling system to remove heat (target power) during irradiation.

3.   Analyses to Support Specification Development

Preliminary evaluations have been performed by the University of Missouri's College of Engineering to determine the thermally induced stress on the target's cladding which is caused by non-uniform heating. A series of analytic, numeric, and experimental tools have been developed in order to evaluate the annular target geometry and to establish a proof-of-concept for the plate geometry. More advanced 2-D analytic models have been developed to establish the property and geometry groups that will influence the target behavior during irradiation. A representation of the stresses developed in an annular target is shown in Figure 1 The colors represent the Von Mises stress that results from interfacial heating. The stresses generated as a result of macroscopic U-growth and fission gas pressure will also be analyzed. A preliminary analysis applicable to both the annular [1] and plate [2] target geometry has been performed.

Fig. 1 ANL's LEU-foil annular target, thermal stress field

The results of the analyses and numeric simulations are being validated by experiment. A mock target, containing a heater element to provide an internal heat source, is placed in a flow loop test section as shown in Figure 2. The deformation of the target's cladding is measured by laser displacement.
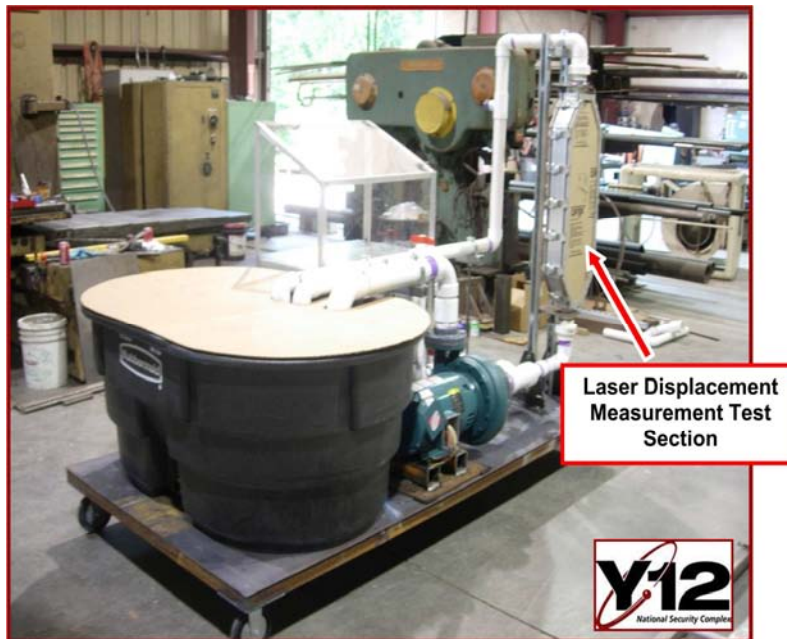

Fig. 2 Flow loop with test section

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The temperature of the LEU-foil will depend upon the thermal contact resistance between the foil and the aluminum tube cladding. Contact resistance, in general, arises due to imperfect mating of components due to micro-scale surface roughness and macro-scale warpage. Any gaps that form between the target's components (aluminum cladding, LEU-foil, and nickel fission recoil barrier) contain relatively low thermal conductivity gases. The thermal conductance between the heat source (LEU-foil) and coolant (water) is thus reduced relative to perfect material contact between the components. A higher contact resistance implies a lower thermal conductance and a correspondingly higher LEU-foil temperature.

Experiments are currently being performed to evaluate the surface state (i.e., roughness) of the LEU-foil so that the thermal contact resistance for the annular target geometry can be characterized. Following this characterization, an envelope of thermal contact resistance for the target geometry can be established.

The surface texture of the foil is measured prior to target assembly and compared to the surface texture measurements following target disassembly to assess the magnitude of thermal contact resistance. The measurements are obtained using the instrument shown in Figure 3. A representative sample of uranium foil manufactured by KAERI using their cooling-roll casting method is shown in Figure 4. The surface texture of the uranium foil is visible.
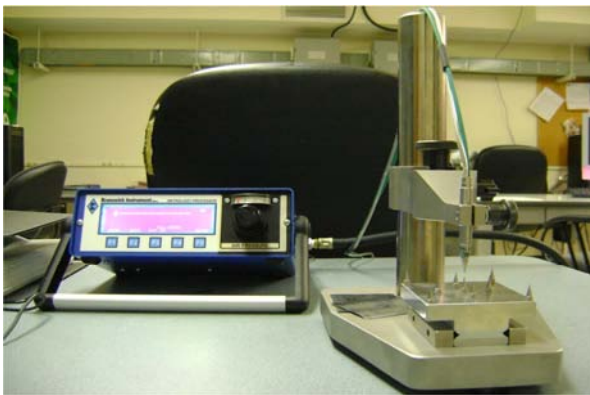
Fig. 3 Surface contact profilometer
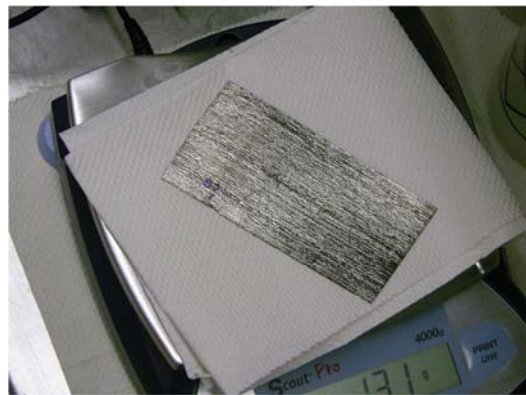
Fig. 4 LEU-foil supplied by KAERI

4. Testing to Support Qualification Strategy

There are limited experimental data published that characterizes the physical behavior of thin uranium foils during irradiation. Macroscopic U-swelling and fission gas generation data is needed to construct models capable of predicting the behavior of LEU-foil Mo-99 targets during irradiation. The models will be used to demonstrate the structural integrity of LEU-foil targets during irradiation.

A test plan has been developed that outlines a strategy for acquiring LEU-foil irradiation behavior data using the Pitesti reactor's post irradiation examination (PIE) facility shown in Figure 5. The PIE data will be used to support the development of a universally qualified LEU-foil based target that can be used by any current or future Mo-99 producer who desires to evaluate the LEU-foil targetry option. A series of tests will be performed under irradiation conditions that represent the maximum irradiation parameters (i.e., thermal neutron flux and irradiation time) of dispersion type targets that are now being irradiated to produce Mo-99. Parameters such as macroscopic U-swelling and fission gas generation will be quantified.

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Fig. 5 Pitesti Reactor PIE Facility

5.   Acknowledgements

We would like to thank Dr. George Vandegrift of Argonne National Laboratory and Mr. Lloyd Jollay of B&W Y-12, LLC.

6.   Technology Readiness Assessment

A Technology Readiness Assessment (TRA) is an industry recognized and accepted method used to assess the technological maturity of a product (in this case, an LEU-foil target) under development. This assessment is formal, systematic, and metric-based. The metrics are defined by the Technology Readiness Levels. A TRA: 1) provides a common language to communicate the maturity of a technology, 2) enables a disciplined approach to evaluate technology readiness, and 3) provides an effective tool and metrics to assess technology risk. It is simply an analysis for determining the technology maturity with respect to meeting product realization goals.   Using this assessment methodology [3], the LEU-foil target is assigned a maturity of TRL 5 for the high-volume production (meaning mass production in sizable lots) [4].

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THE U.S. GLOBAL THREAT REDUCTION INITIATIVE (GTRI) GAP MATERIAL PROGRAM STATUS UPDATE

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ABSTRACT

On May 26, 2004, the National Nuclear Security Administration established the Global Threat Reduction Initiative (GTRI). GTRI's mission is to reduce nuclear and radiological threats worldwide by removing or disposing of excess weapons usable nuclear and radiological materials. While GTRI had several established programs to remove and dispose of these types of materials, there were a number of foreign facilities (primarily research reactors) that had quantities of both U.S. and non-U.S.-origin weapons usable spent nuclear fuel (containing HEU) and separated plutonium, for which there was no identified disposition path. Therefore, GTRI established the "Gap Material" Program to address these types of high risk nuclear materials. This paper will discuss the materials that are covered by the Gap Material Program, accomplishments to date, ongoing international cooperation and the future of the program.

GTRI History

The National Nuclear Security Administration established the Global Threat Reduction Initiative (GTRI) in 2004 to identify, secure, remove and/or facilitate the disposition of high risk vulnerable nuclear and radiological materials around the world, as quickly as possible, that pose a threat to the United States and the international community. GTRI works to reduce and protect vulnerable nuclear and radiological material located at civilian sites around the world.

GTRI Removal Programs Overview

A cornerstone of GTRI efforts is to remove and eliminate excess nuclear and radiological materials. These efforts result in permanent threat reduction because they eliminate bomb making material at civilian sites. Each kilogram or curie of this dangerous material that is removed and dispositioned reduces the risk of terrorists developing a nuclear explosive or dirty bomb. Since 2004, GTRI has removed more than 120 nuclear bombs worth of highly enriched uranium and plutonium, secured more than 775 bombs worth of HEU and plutonium associated with the BN-350 reactor in Kazakhstan, and secured more than 960 radiological sites around the world containing over 20 million curies, enough for thousands of dirty bombs.

At its inception, GTRI had two fuel removal efforts – the U.S.-origin fuel removal program and the Russian-origin fuel removal program. Each program has been incredibly successful with

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the removal of over 2800 kilograms of highly enriched uranium since the programs were initiated.

However, it was quickly recognized that many vulnerable nuclear materials were not covered by the existing programs, so GTRI began efforts to expand its program to address additional materials that could be used for a nuclear weapon. Specifically, GTRI expanded efforts to address the following so-called "Gap Materials"

- Non-U.S.-origin fresh HEU and U.S.-origin fresh HEU not covered by existing programs or policies
- Spent Nuclear Fuel (SNF) containing non-U.S.-origin HEU
- SNF containing U.S.-origin HEU that was not previously addressed in the Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Acceptance Program Environmental Impact Statement (EIS)
- Separated weapons-usable plutonium

Gap Material Authorizations

The first step towards expanding fuel removals efforts required environmental and legal reviews per the National Environmental Policy Act (NEPA) of the United States. These environmental and legal reviews were divided into three separate actions – HEU fresh fuel, HEU spent fuel, and plutonium.

The first issue to be addressed was the return of additional forms of U.S.-origin fresh HEU as well as the shipment of non-U.S-origin fresh HEU to the United States. As the recipient of any fresh HEU shipments, the Y-12 Nuclear Security Complex took the initiative to prepare the documentation needed to bring additional forms of fresh HEU to the United States for disposition. In August 2006, a "Supplement Analysis for the Air and Ocean Transport of Enriched Uranium between Foreign Nations and the United States" was signed allowing for such shipments.

Second, GTRI began to address the possibility of bringing non-U.S.-origin HEU spent fuel to the United States as well as U.S.-origin HEU spent fuel that was not previously addressed in the Foreign Research Reactor Spent Nuclear Fuel (FRR SNF) Acceptance Program Environmental Impact Statement (EIS). GTRI performed a supplement analysis (SA) to the FRR SNF EIS to determine if bringing this material back would have any additional impact on the environment. It was determined that the environmental impacts would be insignificant and the final SA and associated record of decision (ROD) were signed by the NNSA Administrator on January 13, 2009 allowing such shipments under the following limited circumstances:

The material must:
1) pose a threat to national security
2) be susceptible to use in an improvised nuclear device
3) present a high risk of terrorist threat and
4) have no other reasonable pathway to assure security from theft or diversion

Acceptance of Gap Material SNF can also only occur if the material complies with the acceptance criteria of the Savannah River Site facility receiving the Gap Material SNF and provided sufficient storage capacity exists at the facility.

Finally, GTRI began the process of completing the necessary reviews to allow for the importation of separated plutonium. This authorization was received in May 2010 with the issuance of a Finding of No Significant Impact and associated "Environmental Assessment for the Receipt and Storage of Gap Material – Plutonium". As with the supplement analysis

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for HEU spent fuel noted above, any plutonium sent to the United States must meet the same strict acceptance criteria, including that it meet the acceptance criteria of the Savannah River Site and that sufficient storage capacity exists, and is limited to only 100 kilograms. The authorization to receive such quantities of plutonium materials into U.S. is the first ever and considered a significant milestone for GTRI.

Gap Material Program Accomplishments

Since its inception, the Gap Material Program has removed, or verified the disposition of, 228 kilograms of HEU and plutonium at or to the following locations:
- 130 kilograms to commercial entities
- 80 kilograms to Y-12
- 18 kilograms to SRS

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HEU Fresh Fuel Removals

Since 2006 when approval was received to begin shipments of additional quantities of HEU fresh fuel to the United States, GTRI has worked with the Y-12 Nuclear Security Complex, our domestic and foreign counterparts, and commercial entities to identify and eliminate additional excess quantities of fresh HEU. GTRI facilitated disposition of HEU at various commercial entities as well as the return of fresh HEU from both domestic and international locations to Y-12 for storage and disposition. Key shipments included return of fresh HEU from three university reactors in the United States as well as 8 foreign shipments of 77.5 kilograms from our foreign counterparts.

Fig. 1 - Cutting TRIGA rods at Oregon State

Fig. 2 - Packaging of U.S.-origin fresh HEU

HEU Spent Fuel Removals

GTRI's first shipment of non-U.S.-origin HEU spent fuel to the United States was from Chile. This shipment, which included over 18 kilograms of HEU was completed in April 2010 – just prior to the International Nuclear Security Summit. This was a major accomplishment not only because it was the first shipment of non-U.S.-origin HEU spent fuel to the United States under the Gap Material Program, but also because it completed the removal of all HEU from the country of Chile. In addition, the operation was performed under challenging circumstances as the loading operations were performed during the period when a devastating 8.8 magnitude earthquake hit Chile.

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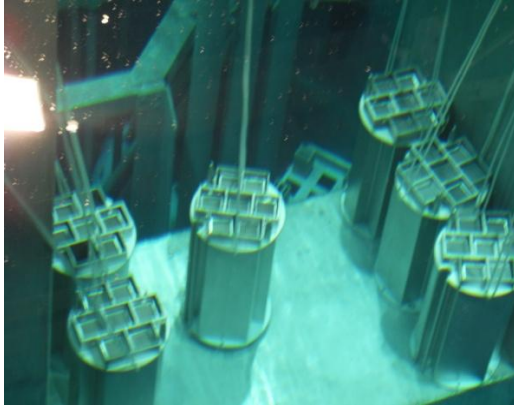
Fig. 3 - Chilean spent fuel at RECH-1   Fig. 4 - HEU spent fuel loading in Chile

Other Disposition Assistance

GTRI is also working with foreign countries – most notably Sweden, Switzerland, Japan, Canada and Argentina - to find alternate disposition solutions for excess nuclear materials. Activities include:
- Developing downblending and processing options for HEU
- Assisting with stabilization and packaging of Pu materials
- Facilitating discussions with commercial entities for disposition

HEU Downblending:

GTRI is providing technical expertise via the Y-12 Nuclear Security Complex and the Savannah River National Laboratory to help countries identify cost effective domestic downblending and processing solutions for HEU materials that cannot be returned to the United States for disposition. For example, GTRI has been working with Yayoi research reactor since 2008 to identify a disposition solution for its U.S.-origin HEU. After careful consideration, and analysis of both technical issues and costs, Japan has decided to downblend this HEU in country and then fabricate fuel elements for its zero power reactor using the LEU.
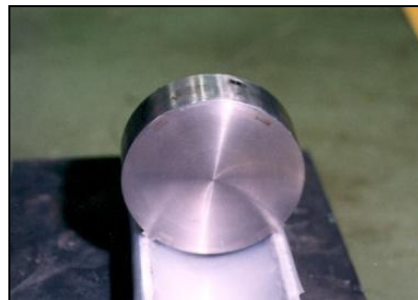
Fig. 5 - Stainless steel clad HEU in Japan

In addition, technical experts are working with Argentina to downblend several kilograms of U.S.-origin HEU that can not be sent to the Unites States for disposition. This included downblending multiple material forms containing HEU (U-Al alloy, U metal, Uranium in liquid form, and contents from legacy UF-6 cyclinder), as well as upgrading hot cells to process irradiated Mo-99 filters. After downblending, this material will be used in target and fuel fabrication.

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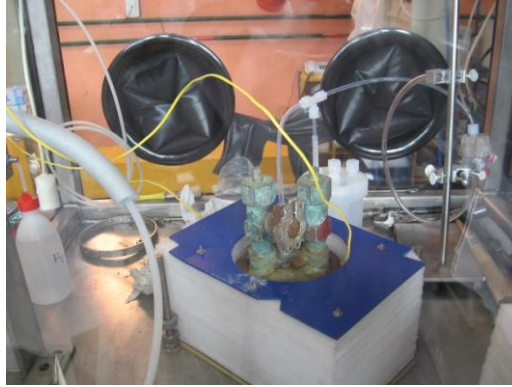
Fig. 6 - UF6 in Argentina

Plutonium Disposition:

GTRI continues to discuss options for plutonium removal and disposition with several countries. This includes technical support to guide the stabilization and packaging of plutonium, review of possible domestic storage options, commercial disposition solutions, and the possible shipment of this material to the United States for disposition. Because of the infancy of this program, GTRI is currently only in preliminary discussions with most countries that have excess separated plutonium.

Future of the Gap Material Program

GTRI is dedicated to continuing to work with our foreign counterparts to find disposition pathways for excess vulnerable nuclear materials, whether by finding a domestic solution, sending to a commercial partner, or sending to the United States for disposition.

If you would like to discuss possible disposition solutions, please contact Sarah Dickerson at sarah.dickerson@hq.doe.gov.

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SERBIAN SNF REAPATRIATION OPERATION.
ISSUES, SOLVING, LESSON

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ABSTRACT

For now the removal of SNF from RA reactor site (PC NFS, Serbia) is the most time-consuming and technically complicated operation under RRRFR Program. The most efficient techniques and lessons learned from other projects of the RRRFR Program as well as new unique technical decisions were used.

Two big challenges were resolved during implementation of Serbian Project: (1) preparation of damaged fuel located in the packages unsuitable for transport, taking into account insufficient infrastructure of RA reactor site and (2) removal of large amount of fuel in one multimodal shipment through several transit countries. The main attention was paid to safety justification of all activities. All approvals were obtained in Russia, Serbia and transit countries.

Special canisters were designed for transportation of specific RA reactor fuel (of small dimensions, unidentifiable, damaged due to corrosion). The canister design was selected to be untight – it was the most expedient decision for that case from safety perspective.

The technology and a set of equipment were designed for remote removal of the fuel from the existing package (aluminum barrels and reactor channels) and placing of the fuel into the new canisters. After fabrication and assembling of the equipment theoretical and practical training of the personnel was performed. Fuel repackaging took about 5 months. SNF was transported in TUK-19 and SKODA VPVR/M casks. The baskets of large capacity were designed and fabricated for SKODA VPVR/M casks. Special requirements to drying the packages and composition of gaseous medium inside were justified to ensure fire and explosion safety.

Specialized ISO-containers and transfer equipment designed under Romanian Project were used together with TUK-19 casks. A forklift and mobile rail system were used to handle SKODA VPVR/M casks under conditions of low capacity of the cranes at the facility.

Due to the tight schedule of RRRFR Program as well as geographical peculiarities of RA reactor site location all types of transport means were used in this transportation: empty casks were delivered by road and air, SNF was transported by road, rail and sea.

The success of the Serbian Project is the result of professionalism and efficient cooperation of many national and international organizations. The lessons learned are useful and could be used in other projects of fuel removal and decommission of nuclear facilities.

1. Development of New SNF Packaging

The major peculiarity of the Serbian SNF removal project is the degraded state of the fuel. An aluminum alloy was used as a structural material of the RA reactor fuel assemblies. The SNF was stored in the untreated water with a high electrical conduction and content of Cl and SO_4 ions for 30-40 years. Due to this fact, the major part of the SNF has significantly corroded causing a contact of the fuel composition with the water environment.

The SNF have been stored in long-length (more than 6 m) aluminum reactor channels and aluminum barrels, which in the process of the long-term storage have significantly corroded, too. Such big and damaged containments could not be used for the shipment and receipt of the SNF at the Russian fuel reprocessing plant.

Since the plan for the SNF transport to the Russian Federation was rather complicated (see the description below), one of the most important requirements was to arrange a single shipment of all the SNF (8030 SFAs).

This caused a necessity to develop a new packaging for the SNF that would be appropriate for the SNF shipment to Russia and have a high capacity. For this purpose, TUK-19 and SKODA VPVR/M casks were selected, and special canisters and baskets were developed (Fig.1). The canisters designed are untight. This significantly facilitated handling operations at the RA reactor, declined

strict SNF drying requirements, and ensured high fire and explosion safety. The oxygen and hydrogen generated as a result of radiolysis and chemical reactions released freely during storage of the SNF canisters in the basin and, when loaded in tight casks, were evacuated by regular blowing of the internal volume of the casks.

ALB   TC2   Basket TC3   SKODA VPVR/M

RC   TC1   Basket 51   TUK-19
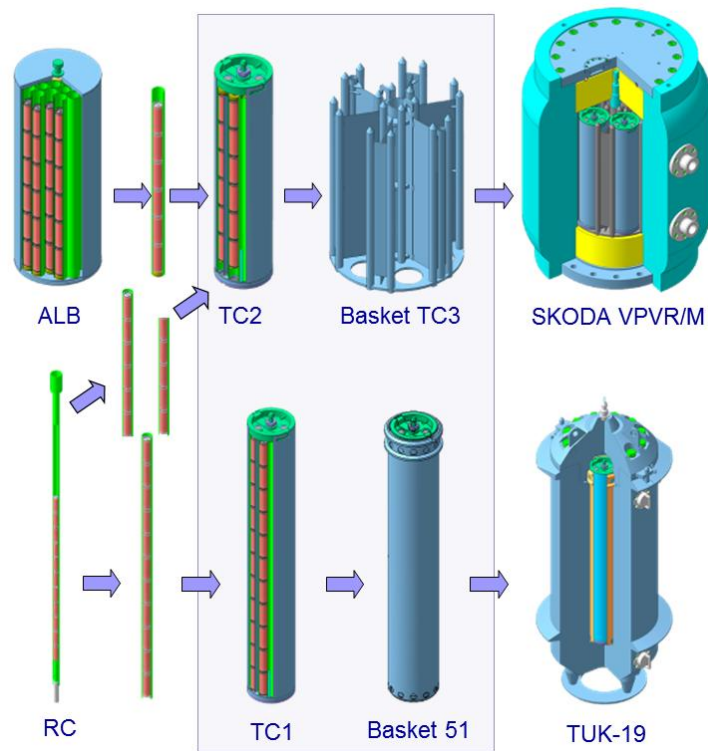
Fig.1. New high-containment SNF packaging

2. SNF repackaging

The SNF removal from the original packagings and its loading into new canisters required development of a special technology and fabrication of equipment and tools. The basis of the technology is remote drilling and cutting of the original aluminum barrels and reactor channels.

The limited space in the RA reactor hall required that all the equipment be deployed compactly. Almost all the repackaging operations were carried out underwater using a working frame, while cutting of the long-length reactor channels was done from a shielded room in the reactor. The prepared SNF in new canisters was stowed on a special underwater rack in the basin.

The entire process was surveyed and controlled by an underwater video system. The set of equipment developed, fabricated and delivered to Serbia (about 100 pieces) included both simple long-length tools, and complicated electric and pneumatic units and large-scale structures (a working frame, racks, frames, etc.). Fig.2 presents some of the SNF repackaging equipment.

In total, the SNF repackaging operations took about five months. Over that period, no emergencies happened causing an overdose or injures to the personnel or a radioactive release into the environment exceeding the approved design limit.

Fig.2. SNF repackaging equipment

3. Cask Handling Operations

The small working area and the low-capacity crane in the SNF storage room required development of a special system to transfer heavy casks inside the facility. A mobile railway system similar to that used in the Czech SNF removal project was utilized. A forklift transferred the casks inside the facility. A unique system for fixing the casks on the fork was developed (Fig.3). A safe deployment of 32 casks was calculated taking into account the allowable load limit for the reactor hall floor.

The experience gained during handling operations with TUK-19 casks during the Romanian SNF removal project was applied to the Serbian campaign. The transfer cask delivered from Romania and upgraded with account taken for peculiarities of operations at the RA reactor was used to load the fuel into the TUK-19 casks.   The TUK-19 casks were shipped in ISO containers that were designed and fabricated under the Romanian project and showed themselves positively in other RRRFR shipments (Libya, Poland).

Other equipment for handling operations with the TUK-19 and SKODA VPVR/M casks was either modernized, or newly fabricated. For instance, the SNF was loaded into canisters using a special above-basin frame to install the casks and a unique self-balancing tool to put heavy canisters into the basket under the canister underwater.

Dynamic accumulation of oxygen and hydrogen in the casks filled with SNF was calculated to ensure fire and explosion safety. The calculation results formed the cask drying requirement, a special procedure for regular gas replacement in the casks, and safety measures for opening of the casks at FSUE "Mayak" PA. The calculations were verified by actual measurements of the gas mixture in the casks during temporary storage at the RA reactor facility.

Fig.3. Railway system and forklift

4. Shipment of SNF Packages

It was impossible to use only the overland transport to ship the SNF to Russia due to the peculiar geographical location of Serbia and a political standpoint of some countries of transit. So, the choice was made in favor of the "sea" route used for the SNF removal from Hungary. Across Serbia, the SNF packages were transported by trucks from the RA reactor to the Serbian-Hungarian border, where they were reloaded onto a train. The train with the SNF packages went through Hungary and Slovenia to the sea port of Koper, where the packages were loaded on a ship. Then, the packaged were shipped over the Mediterranean Sea and the Atlantic and Arctic Oceans to the Russian port of Murmansk, where they were loaded onto a train again and transported to FSUE "Mayak" PA across Russia.

Since the empty casks were delivered to Serbia from Russia by plane, we can state that the Serbian project involved all types of civil transport for the shipment of the Class 7 cargo.

5. Safety Issues and Approvals

The long-term, complicated and potentially hazardous SNF repackaging and loading operations at the reactor facility could not have been possible without an appropriate safety analysis. Jointly with the Serbian operator and Russian experts, R&D Company "Sosny" made a comprehensive analysis of all technologies, procedures and equipment used. Several safety reports were prepared; they were reviewed and approved by the Serbian regulatory body and Slovenian and IAEA experts. All necessary procedures and documents for the safe performance of the operations and correct emergency response were prepared.

Under the guidance of the Sosny experts, all the equipment delivered to the RA reactor was assembled, adjusted and tested.

Theoretical and practical training of the Serbian personnel in performing the operations was conducted. The training was carried out in the step-by-step manner, i.e. every single procedure was trained right before its implementation. So, the total cycle of training included five stages of theoretical training alternating with practical operations.

Over the entire period of practical operations, the Sosny specialists, who developed the technologies and equipment, were present at the facility to control the operations and provide prompt consultations.

Prior to the shipment, all necessary documents were prepared and all required Russian, Serbian and transit countries' approvals were received. In addition to getting the shipment approvals, we, in Russia, had to go through a lengthy and complicated procedure of approval of the foreign SNF import. The Unified project documents including special ecological programs were developed, all approvals of Russian authorities were obtained, and a positive expert assessment of the State Ecological Expertise Committee was received.

6. Conclusions

The Serbian SNF removal campaign took about five years involving dozens of organizations from different countries, in particular, IAEA, Serbian and Russian operators, carriers, developers and

fabricators of the equipment, expert organizations, competent, authorized and regulatory bodies of Russia, Serbia, Hungary and Slovenia, and other organizations providing technical support (Fig.4). For R&D Company "Sosny", it has become the 27th international shipment of nuclear materials and the culmination of the accumulated experience.

The Serbian project has demonstrated the following:

- a feasibility of the safe preparation and transport of significantly degraded SNF;
- a possibility to handle the casks at facilities with small working areas and low-capacity cranes;
- the universality of the TUK-19 and SKODA VPVR/M handling equipment developed under the RRRFR program and a possibility to accommodate it to specific conditions of a certain facility;
- advantages of untight canisters for some cases;
- a possibility to arrange a multi-modal shipment through several countries of transit;
- a possibility of effective interactions between many organization under a single project.

No doubt, the Serbian campaign has become a success due to the high competence of its participants and a result of an ample application of the experience gained from similar activities. In its turn, the Serbian project has generated several perspective engineering solutions that can be called for by other fuel removal and nuclear facility decommissioning projects.
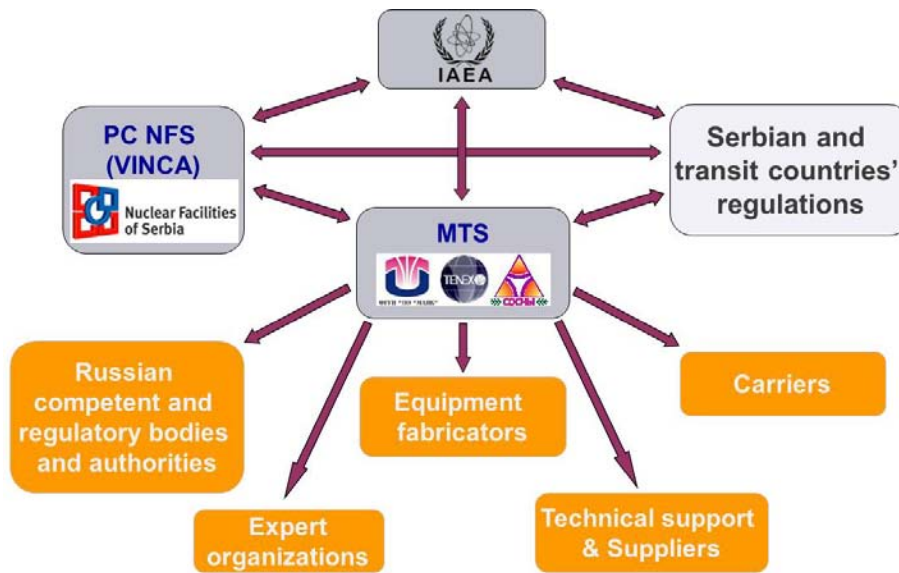

Рис. 4. Organizations involved in the project

NUCLEAR FUEL REPATRIATION
FROM THE REPUBLIC OF SERBIA

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ABSTRACT

At the beginning of 2002, the Government of the Republic of Serbia issued a directive to permanently shut down the RA research reactor at the Vinča Institute of Nuclear Sciences in Belgrade and decided to repatriate all fresh HEU fuel elements and all spent LEU and HEU fuel elements being irradiated during entire exploitation of this reactor. The fresh fuel elements were shipped to the Russian Federation in August 2002. Packed in special containers, the fuel was transported by truck to the Belgrade Airport and then sent by commercial cargo aircraft to its final destination in the Russian Federation. This was the first actual step in implementation of the RRRFR Programme. During the RA reactor's nearly 25 years of operation, more than 2.5 tons of heavy metal in form of the irradiated fuel elements were generated and temporarily stored in water pools in the spent fuel storage within the reactor building. Because the storage lacked adequate conditions for long-term storing, a number of fuel elements lost their tightness which led to fission products release into water in the spent fuel pond. Specific design of the fuel elements and the fact that a number of them were in poor condition, necessitated repackaging of these elements into new canisters designed specifically for the transport in TUK-19 and SKODA VPVR/M transport casks. After four years of negotiations and preparations, in September 2006 a contract for the repatriation of the RA reactor's spent fuel was signed. The parties involved included consortium of Russian companies led by the R&D Sosny Company, the Vinča Institute of Nuclear Sciences and the IAEA. In addition to technical and logistical support, international financial support was also provided to help realize completion of this project. Nearly four years later, at the end of 2010, the spent fuel elements left Serbia, bound for the Russian Federation.

1. Introduction
1.1 Reactor Facility

The RA reactor is a Russian-design tank type research reactor, using heavy water as a primary coolant and a moderator. It is located at the Vinča Institute of Nuclear Sciences nearby the River Danube. Its nominal power was 6.5 MW. The reactor went critical in December 1959 and was temporarily shut down in August 1984 (RA reactor block is shown in Fig. 1). During this period of operation, reactor has been successfully used for scientific research, but also for commercial purposes. From its first commissioning in 1960 until 1975, the reactor was using Russian-origin low enriched uranium fuel (2% of ^{235}U). In 1976, the original fuel was gradually replaced by a highly enriched uranium fuel (80% of ^{235}U), developed and qualified in the Soviet Union in the meantime.

Fig 1. RA Reactor block

After temporary shut down in 1984, it was decided to make a series of reconstructions of facility's systems in order to enable safe and continuous operation of the reactor for the next 20 to 25 years. However, planned reconstructions had never been finished and reactor has never been put into operation again. In July 2002, the Government of the Republic of Serbia made a decision to shut down the RA reactor permanently.

1.2 Fuel repatriation background

Following the objectives of the Russian Research Reactors Fuel Return (RRRFR) Programme, launched in 1999 by the Russian Federation, the United States and the IAEA,

Director General of the IAEA, one year later, invited countries in possession of Russian-origin research reactor fuel to examine willingness in returning HEU fuel to the Russian Federation. The Republic of Serbia expressed strong interest in participating in this programme.

At the beginning of 2002, the Serbian government issued a directive to repatriate all fresh HEU fuel elements and also all spent LEU and HEU fuel elements being irradiated during entire exploitation of the RA research reactor.

1.3 Fresh fuel shipment

After several months of negotiations with US DOE and two Russian institutes, all fresh HEU fuel elements at the Vinča Institute were transported to the Russian Federation in August 2002. There were total of 5046 such fuel elements having approximately 48 kg of 80% enriched uranium. Packed into special transport containers, these elements were loaded onto truck and transported to the Belgrade Airport. By commercial cargo aircraft, fresh fuel was sent to the Ulyanovsk Airport in the Russian Federation and then by truck to the RIAR Institute in Dimitrovgrad.

This shipment was the first actual implementation of the RRRFR Programme.

2. Feasibility considerations
2.1 Fuel type and inventory

During exploitation, only the TVR-S fuel elements of the LEU and HEU type, manufactured in the former Soviet Union, were used in the RA research reactor (Fig. 2). Both types of fuel elements have the same geometry, but the mass of ^{235}U is slightly different

Fig 2. TVR-S fuel elements

(7.25 g in LEU and 7.7 g in HEU fuel element). TVR-S fuel element is an empty cylinder with the outer diameter of 37.2 mm having tubular fuel section 2 mm thick and 100 mm long. This section, coated with 1 mm aluminium cladding, is made of metal uranium in LEU and uranium dioxide mixed in aluminium matrix in HEU fuel element. Total length of the fuel element is 113 mm.

There were 8030 TVR-S fuel elements of both types being irradiated until August 1984. All of them were stored in reactor building in four basins (filled with tap water) in spent fuel storage next to the reactor room (basins are interconnected by the channel, which ends up inside reactor block). Fuel elements were positioned in aluminium tubes inserted either in stainless steel containers, or in aluminium barrels. Majority consisted of LEU fuel elements (6656) placed in aluminium barrels and stainless steel containers, while HEU fuel elements (1374) have been put into stainless steel containers only.

2.2 Characterization of the fuel

Average burn-up of LEU fuel elements for all the time of reactor exploitation was approximately 6.9 MWd/kgU, while the maximum burn-up did not exceed 17 MWd/kgU. For HEU fuel elements, average burn-up was about 134 MWd/kgU and the maximum burn-up reached up to 400 MWd/kgU. Total activity of all spent fuel elements did not exceed 4000 TBq. At the beginning of 2002, maximum decay heat was approximately 150 mW for LEU fuel element and 90 mW for HEU fuel element.

Activity measurements of water samples taken from aluminium barrels and stainless steel containers in the early 2000s showed that several hundreds of fuel elements may have had breached cladding. On the other hand, formation of significant corrosion deposits on the cladding of numerous fuel elements, which occurred mainly during last years of reactor operation, made it very difficult to take out these elements from aluminium tubes.

2.3 Spent fuel removal assessment

Taking into account the structure of fuel elements and the containers where they have been positioned, it was clear that all fuel elements had to be repackaged into new containers being suitable for transport. However, pulling out fuel elements from aluminium tubes would have been an impossible or very risky action that could damage their cladding being already weakened by corrosion processes. It meant that appropriate repackaging technique had to be developed.

Calculations performed have pointed out that upon opening of aluminum barrels, some 10^{13} Bq of ^{137}Cs activity might be released into water in the spent fuel pond and that the same amount may be released in one-year time afterwards. Therefore, an efficient system for absorbing radionuclide ^{137}Cs in the storage basins was mandatory if any fuel handling will be carried out there.

3. Legal framework

The Agreement between the governments of the United States and the Russian Federation, which has been signed in May 2004, accordingly provided legal authority for the RRRFR Programme. Four months later, the Ministry of Science and Technology of the Republic of Serbia engaged the Vinča Institute to prepare (in close cooperation with the IAEA) input data relevant for the spent fuel shipment to the Russian Federation. In September 2006, following an international tender, a tripartite contract for repatriating RA reactor's spent nuclear fuel was signed. The parties involved included consortium of Russian companies ("Sosny", "Tenex" and "Mayak"), the Vinča Institute and the IAEA. Work obligations for the Russian companies and for the Vinča Institute were fully determined and distributed among participants. Spent fuel repackaging and loading technology, including design and manufacture of special equipment had to be developed and worked out by the Sosny Company, while all the facility preparations had to be carried out by the Vinča Institute. Transport of all the spent fuel from the reactor facility, as stated in the contract, had to be completed until the end of 2010 and in one shipment only.

Two basic documents for import of spent nuclear fuel to the Russian Federation are obligatory. The first one is the so-called "Government-to-Government Agreement" and such an agreement between the governments of the Russian Federation and of the Republic of Serbia was signed in June 2009. According to this agreement, waste generated by the reprocessing of the RA reactor's spent fuel will be permanently stored in the Russian Federation. The second one, the so-called "Foreign Trade Contract", determines all mutual obligations referring to spent fuel transport including the scope of services to be provided by the Russian Federation. In September 2009, such a contract was signed between the Public Company "Nuclear Facilities of Serbia" (as successor of the Vinča Institute) and the "Federal Centre of Nuclear and Radiation Safety" from the Russian Federation.

4. Preparation activities
4.1 Engineering design

Sosny Company has evaluated all available casks suitable for RA reactor's spent fuel transport and selected the two types: TUK-19 and SKODA VPVR/M. Both types were acceptable at the Mayak reprocessing plant and available number of these casks allowed all RA reactor's spent fuel elements to be sent in one shipment. Fuel handling technology and equipment design, including facility modifications have been directed by the features of these casks. For positioning TVR-S fuel elements into transport casks, special canisters were designed, as well as the corresponding baskets (Figs. 3 and 4). Numerous failed fuel elements and technical difficulties to separate them from the others, as well as long time of storing canisters with such elements

Fig 3. Canister for TVR-S fuel elements

in the storage pond (up to one year), demanded for non-hermetic canister design. One canister for the TUK-19 cask could have been loaded with 132 TVR-S fuel elements and the one for the SKODA cask with 72 such elements. Capacity of the casks was: one canister in TUK-19 and 6 canisters in SKODA cask. Consequently 16 TUK-19 casks and 16 SKODA casks were needed to load all RA reactor's spent fuel.

Fig 4. SKODA basket

Technical preparation of RA reactor's spent fuel for transport was divided into several stages: repackaging of fuel elements into new canisters; loading of canisters into baskets; loading of baskets into transport casks and preparation of these casks for shipment. A great number of sophisticated tools, devices and subsidiary equipment had to be designed and constructed to enable these operations, especially the repackaging ones. All the procedures for repackaging TVR-S fuel elements and loading canisters into transport casks were thoroughly evaluated primarily from the viewpoint of sub-criticality and radiation safety. Safety of loaded casks has also been analyzed for normal and accidental conditions. Obtained results have shown that chosen technology approach and designed equipment satisfied all the requirements of nuclear and radiation safety and were in compliance with relevant spent fuel transport regulations, too.

4.2 Facility preparations

A lot of facility modifications, including new equipment provision, were carried out to enable repackaging and loading activities in the reactor building. Among them, the substantial ones were:

Fig 5. New ventilation pipeline

- underwater metal structures removal (spent fuel storage pond);
- bridge crane replacement in the spent fuel storage;
- bridge crane upgrade in reactor room;
- special ventilation system reconstruction (Fig. 5);
- electric power supply system adjustment;
- adaptation of control room in spent fuel storage;
- adaptation of access roads to the reactor building;
- purchase of 16-ton capacity forklift;
- purchase of casks rail-transfer system (Fig. 6);
- purchase of ^{137}Cs removal system;
- purchase of additional radiation monitoring systems.

Fig 6. Cask rail-transfer device

4.3 Provision of permits and licenses

Considerable effort has been required to provide necessary permits and licenses for spent fuel shipment to the Russian Federation. To import spent nuclear fuel into the Russian Federation a series of documents had to be completed, reviewed and approved by competent Russian authorities. Initial document was certificate for the RA reactor's spent fuel package design for both TUK-19 and SKODA casks. This certificate, issued by Rosatom, enabled elaboration of the Unified Project documents. This project is basically an overall assessment of the radiation, economic, social and environmental impact to the country, especially for the Chelyabinsk region. When positive results of the State Ecological Review have been submitted to Rosatom, import of spent nuclear fuel was granted and the "Foreign Trade Contract" was signed.

Authorities in the Republic of Serbia validated firstly certificates for the spent fuel package design and then issued approvals and permits for radiation protection procedures, physical protection procedures and reactor facility operations programmes and procedures.

After Take-back guarantee and Civil liability insurance documents have been provided, Serbian export license for spent nuclear fuel was issued by the Serbian Regulatory Agency for Radiation and Nuclear Protection.

To provide transport license in transit countries (Hungary and Slovenia), Regulatory bodies in these countries had to validate spent fuel package design, firstly. After all aspects of spent fuel transit have been discussed and coordinated with Serbian institutions, competent authorities in these countries issued transit approvals.

4.4 Operational readiness

To assure safety in executing such a complex and delicate operations, such as RA reactor's spent fuel repackaging and loading, a series of safety analysis reports had to be elaborated, reviewed and approved. Written by the Sosny Company and the Vinča Institute, many international institutions have been engaged in reviewing these reports. Final approval was issued by the Serbian Regulatory Agency for Radiation and Nuclear Protection. Sosny Company also elaborated sequential set of instructions to be used in spent fuel repackaging and loading operations. These instructions have been incorporated into all-inclusive operational procedures prepared by the Public Company and approved together with safety analysis reports.

Reactor staff received extensive training for handling tools and devices both for repackaging activities and for loading ones. Specialists from the Sosny Company have conducted training of reactor staff for repackaging activities and specialists from the Mayak Reprocessing Plant and the NRI in Řež (Czech Republic) joined them in conducting training for loading activities. Special training was organized for persons from supporting organizations being engaged in the spent fuel transportation (several practical exercises have been performed, too).

Special equipment, received by the Sosny Company and other organizations, including all necessary devices and tools, was installed, tested and adjusted prior to executing repackaging and loading operations. Existing containers with spent fuel elements have been prepared for repackaging and relocated in the storage basins according to the pre-defined repackaging sequence

Safeguards Division of the IAEA was informed about the fuel shipment, so they have been prepared to control the flow of nuclear material in all phases of the repackaging and loading operations.

5. Execution of operations
5.1 Repackaging operation

Dismantling of aluminium barrels was performed at the Working platform installed in one of the basins in the spent fuel storage (Fig. 7). Fuel elements were taken out from the barrel and positioned in the new canister. The whole operation has been carried out underwater using long tools and an electric cutting device. Actions performed by operating personnel were monitored in the storage control room (Fig. 8). For repackaging of fuel elements from stainless steel tubes, alumunium tube with fuel elements, being almost 6 m long, was firstly cut in a small compartment in the reactor block. Fragment with fuel elements

Fig 7. Working platform

was brought to the Working platform in the spent fuel storage, where further cutting of the fragment tube with manual cutting device enabled transferring fuel elements into new canisters.
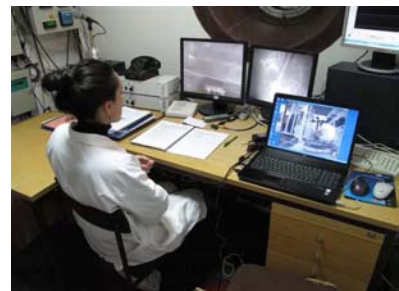

Fig 8. Control room

Repackaging of RA reactor's spent fuel began on December 2nd (2009) when the first aluminium barrel was opened. Working in 3 shifts per day (total of 5 shifts), spent fuel from

all 30 barrels was repacked by the end of February 2010. Four weeks later, repackaging of spent fuel from the stainless steel tubes began. In the meantime, part of the repackaging equipment has been removed and additional equipment was installed, adjusted and tested. Fuel elements from the last of 297 steel tubes, were repacked on May 24th 2010.

When repackaging operation was finished, 110 canisters loaded with 8030 fuel elements were temporarily stored in underwater shelves mounted in all four basins of the storage pond.

5.2 Loading operation

Canisters with fuel elements have been loaded into baskets placed on an underwater stand mounted at the bottom of the storage basin. Special tool was used to carry out this operation. Above the stand, platform for SKODA casks and TUK-19 transfer cask was mounted, enabling direct insertion of baskets into them (Fig. 9). Being loaded, casks were brought into reactor room (using forklift and rail-transfer device), where baskets from transfer cask have been put into TUK-19 cask (Fig. 10). Each transport cask was dried with hot air firstly and then subjected to vacuuming, gas filling and sealing (Fig. 11). Standard SDGL equipment from NRI Řež was used for SKODA casks and slightly modified version for

Fig 9. Loading a basket into SKODA cask

Fig 10. Loading a basket into TUK-19 cask

TUK-19 casks. For the first time, TUK-19 casks have been subjected to vacuuming and the results were quite satisfactory. Finally, leakage tests were performed and IAEA seals have been put onto each cask.

Loading transport casks with canisters has been carried out at three different times, due to casks availability and delivery schedule. In the second half of August, the first 12 SKODA casks were loaded with 72 canisters and in the first half of November, the last 4 SKODA casks were loaded with 22 canisters. In the meantime, in the second half of October, all 16 TUK-19 casks were loaded with 16 canisters.

Fig 11. Drying and vacuuming of TUK-19 casks

5.3 Transportation

Each group of casks, being loaded with canisters, was immediately transferred into ISO-containers placed on the plateau in front of the reactor building (Fig.12). All ISO-containers were properly marked and sealed by the Public Company and the Serbian customs. On November 14th 2010, 8 ISO-containers loaded with 16 SKODA casks, 6 ISO-containers loaded with 16 TUK-19 casks and one ISO-container loaded with equipment for handling TUK-19 casks, were ready for transport.

Fig 12. Loading SKODA casks into ISO-containers

Loading of ISO-containers onto trucks was carried out on November 18th and the convoy left Vinča the next day. Transport route was determined almost one year ago (Fig. 13), but the timetable was fixed a few days before

Fig 13. Transport route

transportation. Near the Hungarian border, ISO-containers were reloaded onto railway flatbed cars and transferred through Hungary and Slovenia to the Koper harbor. The train arrived to Koper on November 21st. Immediately upon arrival, ISO-containers were reloaded again - from the railway cars onto the boat (Fig. 14). A few hours later, the boat left Koper towards Murmansk. It took more than 3 weeks until boat reached Murmansk where ISO-containers were reloaded onto railway cars and on December 22nd, RA reactor's spent nuclear fuel arrived to Mayak.

Fig 14. Loading ISO-containers on a boat in Koper harbor

During loading of ISO-containers at the RA reactor facility and all the way long to Koper harbor, significant police forces were engaged to provide physical protection of the cargo. For all this time, continuous radiation control has been provided, too.

6. Summary

Repatriation of the RA reactor's spent nuclear fuel was a complex and challenging task. Unique fuel elements, bad storing conditions and poor fuel handling capabilities on one hand, combined with requirement to repackage all fuel elements on the other, asked for development of sophisticated operational methods and proper equipment design. Good planning and organization from the very beginning and extremely good cooperation with many international organizations and institutions, followed up with substantial financial help given by several countries, enabled successful completion of this task.

After safe transportations of fresh and spent nuclear fuel, carried out in 2002 and 2010, the Republic of Serbia lined up alongside other countries having no highly enriched uranium material.

THE GLOBAL THREAT REDUCTION INITIATIVE'S RETURN OF HIGHLY ENRICHED URANIUM FROM CHILE

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ABSTRACT

In March 2010, the U.S. National Nuclear Security Administration's Office of Global Threat Reduction (GTRI), in collaboration with the Chilean Nuclear Energy Commission (CCHEN), completed a shipment of 18.2 kilograms of non-U.S.-origin highly enriched uranium (HEU) to the United States. The HEU was in the form of 71 aluminium-clad material test reactor (MTR) fuel elements and was the first GTRI Gap Program shipment that included non-U.S. origin irradiated nuclear fuel. Although shipments of research reactor fuels are not unique, this shipment served as a cornerstone to the first Presidential Nuclear Security Summit held in Washington, D.C., in April 2010. Carrying out the shipment became critical when a severe earthquake struck Chile just one day before the shipment was to occur. As the fuel had already been packaged in casks and the ocean vessels were nearing the port, U.S. and Chilean officials decided that it was most imperative that the shipment continue as planned. After careful analysis of the situation, inspection of the transportation packages, roadways, and port services, the shipment team was able to make the shipment occur in a safe and secure manner. This paper describes the loading activities at both the RECH-1 and RECH-2 reactors as well as the transportation of the loaded casks to the port of departure.

1. Introduction and Background

The RECH-1 and the RECH-2 research reactors are operated by the Chilean Nuclear Energy Commission (CCHEN), the primary national organization for nuclear activities in Chile. RECH-1 is located at the La Reina Nuclear Center in Santiago; RECH-2 is located at the Lo Aguirre Nuclear Center near Santiago. RECH-1 is a pool-type reactor moderated and cooled by light water and reflected by beryllium. RECH-1 achieved initial criticality on October 13, 1974, and operated at a nominal power of 5 MW. RECH-2 is also a pool-type reactor moderated and cooled by light water, but reflected by graphite. RECH-2 achieved initial criticality on February 4, 1977, and operated only intermittently at the design rating of 2 MW. Both RECH-1 and RECH-2 use material test reactor (MTR) fuel elements.

In August 1996, CCHEN and the United States Department of Energy's Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Acceptance Program collaborated to return 28 irradiated fuel elements containing highly enriched uranium (HEU) of U.S.-origin from RECH-1 to the Savannah River Site (SRS). In December 2000, a second shipment of 30 irradiated fuel elements (also HEU of U.S.-origin) was returned to SRS. Although it was recognized that CCHEN possessed other MTR fuel elements containing HEU, those elements were not of U.S.-origin and, therefore, could not be shipped to the U.S. because of limitations imposed by the FRRSNF Programs National Environmental Policy Act (NEPA) authorization.

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[1] Paper Presenter

In 2004, the National Nuclear Security Administration (NNSA) established the Global Threat Reduction Initiative (GTRI), which assimilated the FRRSNF Program into its "Remove" portfolio. The establishment of GTRI resulted in the creation of a Gap Material Program, which would pursue disposition of non-U.S.-origin HEU and U.S.-origin HEU not covered by other already existing programs. The Gap Program covers both fresh and irradiated HEU. NEPA authorization to return materials under the Gap Program was received in two parts. In the first, NNSA was authorized to remove fresh HEU to the Y-12 National Security Complex (2006) and in the second, was authorized to remove additional forms of spent fuel to SRS (2009). Thus, these authorizations opened the door for shipment of non-U.S.-origin HEU from Chile to the United States.

2. Fuel Description

The fuel elements for both the RECH-1 and RECH-2 reactors are standard MTR designs. These elements were fabricated before CCHEN's start-up of a domestic fuel fabrication process for MTR elements containing low enriched uranium. The fuel fabricators for the RECH-1 and RECH-2 fuel elements were, respectively, Dounreay (United Kingdom) and JEN (Spain with HEU supplied by France). The basic details of the elements are: box-type assemblies with 18 flat plates, lifting bail at the top and a flow nozzle at the bottom. Materials of construction are aluminium for the structural components (including plate cladding) and U-Al alloy for the core. Dimensionally, the fuel elements are very similar. The differences are in ^{235}U enrichment (45% for RECH-1 and 89.9% for RECH-2) and ^{235}U loading (183 g/element for RECH-1 and 135 g/element for RECH-2).

The total inventory of non-U.S.-origin HEU elements in Chile: 40 at RECH-1; 31 at RECH-2. Two of the 31 elements at RECH-2 consist of loose plates (36 total).

3. Preparations

Discussions to ship the non-U.S.-origin HEU fuel elements began in earnest in August 2006 at a meeting at the International Atomic Energy Agency (IAEA) in Vienna where CCHEN and GTRI representatives discussed disposition options. These options included shipment to the United States, the United Kingdom, France, or a combination of the three. GTRI and CCHEN analyzed the options, but eliminated the United Kingdom and France as possible disposition solutions. Of course, shipment to the United States would not be possible until the NEPA authorization for Gap materials was approved. As noted above, GTRI performed a Supplemental Analysis to the FRRSNF Environmental Impact Statement (EIS), which resulted in a Record of Decision (ROD) that allowed the shipment of the non-U.S.-origin HEU spent fuel to the United States. The ROD was signed in January 2009 and paved the way for the shipment of Chilean non-U.S.-origin spent fuel to the United States.

To plan and execute the shipment, GTRI and CCHEN carried out negotiations on contracts, safeguards issues, regulatory issues, public affairs, and a host of technical issues. Similar to contract(s) used in the 1996 and 2000 FRRSNF shipments, the new contract included the details of the inventory to be shipped, roles and responsibilities for each party, nuclear liability, and title transfer.   The safeguards issues included removal from Chile-IAEA agreements and placement into US safeguards. Regulatory issues included compliance with Chilean law and international regulations related to the transport of nuclear material.

Upon technical review of the inventory and options for shipment, NNSA prepared to ship 69 spent HEU fuel elements to SRS and two fresh HEU elements (36 loose plates) to the Y-12 National Security Complex (Y-12). The two fresh fuel elements, both from RECH-2, had only been slightly exposed in the reactor in the late 1970s, and analysis of the irradiation history concluded that they could be defined as unirradiated.   CCHEN completed the

necessary fuel description forms for the 69 elements that would go to SRS and for the 2 elements that would go to Y-12. These fuel description forms allowed the respective receiving facilities to verify that the fuels met acceptance criteria. Evaluations included positive identification, operational history, receipt and handling, storage, disposition and compatibility with transport package requirements.

The transport package chosen for shipping the irradiated fuel was the NAC International LWT (Legal Weight Truck) spent fuel cask. The LWT had been used previously in the 1996 and 2000 shipments from RECH-1, therefore CCHEN was familiar with that package and its requirements. The ES-3100 would be used to ship the two elements (loose plates) to Y-12. Safety Analysis Reports and Competent Authority Certificates were sent to CCHEN for validation in Chile.

After signing of the contract, approvals of receipts at SRS and Y-12, and validation of the transport packages in Chile, the project schedule moved forward. The NNSA contractor for transportation, Global Threat Reduction Solutions[2], began implementing the schedule to ship equipment on an ocean going vessel to the Chilean port of San Antonio.

4. Loading Operations

The project plan, as determined by CCHEN and GTRI, was to load the RECH-1 fuel first and then move operations to RECH-2. RECH-1 fuel would take the longest to load because it had more fuel elements and a requirement to crop. Cropping of the elements was necessary to use only one LWT at RECH-1. NAC could use the 42-element basket arrangement (6 baskets/7 positions) and load the entire inventory. Additionally, one element from RECH-2 was sent to RECH-1 for cropping and loading. This would allow the second LWT at RECH-2 to use the 28-element basket arrangement (4 baskets/7 positions) and not perform any cropping operations.

Qualified personnel from NAC arrived in early February 2010 to begin setup of equipment at the RECH-1 reactor. The NAC team and CCHEN operations staff worked together in a deliberate manner to ensure all operations were performed safely. A submersible saw was used to crop the RECH-1 elements to fit the LWT baskets. The cropping and basket operations were performed in the reactor pool. After each basket was loaded, the NAC Dry Transfer System (DTS) was used to transfer the loaded basket to the cask. After loading operations were completed, the cask was leak tested, surveyed and loaded into the ISO.

The NAC team then repeated cask loading operations at RECH-2 without the cropping requirement.

Loading of the two elements (36 loose plates) was performed jointly by CCHEN and Y-12 personnel. The loose plates were bundled into two separate handling units and loaded into the ES-3100s. Two ES-3100 containers were required, and the elements were packaged by hand. Each ES-3100 was sealed, leak-tested, and closed.

5. Shipment

All operations were completed safely with the transport packages and ancillary equipment loaded in the ISO containers on February 25, 2010. Then, on February 26, an 8.8 magnitude earthquake struck off the coast of Chile causing widespread destruction and casualties to the coastal city of Concepción southwest of Santiago. Santiago also sustained damage, although not as severe. With the Chilean government now facing a national

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[2] GTRS is a partnership that includes Edlow, NAC, Energy Solutions and others.

emergency, GTRI and CCHEN needed to quickly answer several difficult questions. Could the shipment proceed as planned? Were the roads passable? Was the selected port of departure still available or would a new port need to be identified? As national resources were occupied with emergency response operations, would security forces be available to support transportation of the nuclear fuel to the port of departure? The CCHEN and the GTRI team members travelled to the reactors as soon as possible to inspect the packages for damage. Fortunately, there was no damage and only a slight movement of the ISO containers was observed.

With the locations considered safe, the CCHEN-GTRI team set out to determine if the shipment should proceed. Unfortunately, the departure port of San Antonio was now closed and a new port would need to be identified. The port of Valparaiso was inspected and found to be in working condition, though the pier crane was deemed unsafe due to a large crack in the support structure. The team decided that since the Valparaiso pier was available, the ship's crane would be used to transfer the loaded ISO containers into the cargo hold.

Fig. 1. Separation of the roadway.

Route assessments performed by the Chilean National Police and CCHEN/GTRI officials identified a safe route to Valparaiso, but required the convoy to bypass a mountain tunnel, thereby adding significant ground-transport time to the shipment. Next, the availability of security forces was confirmed. Since the packages were loaded and ready to go, the Chilean authorities authorized the transport to go ahead.

6. Final Comments

The earthquake was unprecedented and damage to Chile's infrastructure was severe. Several members of the Chilean team lost their homes. Throughout this crisis, the personal losses, and the heightened concern for safety, the CCHEN/GTRI team stayed focus and completed the mission. The team made sure safety and security were a priority before moving forward. Flexibility and adaptability in the rerouting to a new port for shipment to the United States were essential. Interestingly, imbedded reporters who were sent to cover

what was thought to be a typical shipment to the United States were able to capture both the successful efforts of the team and their adjustments to the situation. That coverage spoke volumes for the determination of the CCHEN/GTRI team, the security forces, port authorities and others to complete the mission.

NNSA announced that the shipment had successfully been completed on the eve of the Presidential Nuclear Security Summit in April 2010. The shipment's success, despite the massive earthquake, provided a positive example of nonproliferation collaboration between countries in the spirit of nuclear security.

Fig. 2. Ocean vessel departing Valparaiso.

DISPOSAL OF THE FRESH AND IRRADIATED NUKLEAR MATERIAL OF THE ROSSENDORF RESEARCH NUCLEAR FACILITIES

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ABSTRACT

A 10 MW research reactor, two zero power reactors as well as nuclear fuel research facilities have been operated from 1957 to 1991 at the Rossendorf Research Site next to Dresden, Germany. The deployed nuclear fuel originated exclusively from Russia. When in the beginning of the 90s it was decided to shut down and decommission all nuclear facilities at Rossendorf, considerable amounts of fissile material were at the Site (mainly "fresh" fuel: approx. 500 kgs HEU and approx. 700 kgs LEU; and irradiated fuel: approx. 164 kgs HEU and approx. 186 kgs LEU).
The VKTA-conception for clearing the Rossendorf Research Site from nuclear material defines priorities and the ways of disposal. For the nuclear material in form of fuel elements, the Russian Research Reactor Fuel Return Program (RRRFR Program) has been of extraordinary importance for the return of this fuel back to the Russian Federation. For other kinds of nuclear material like liquids or powder, separate management projects were developed. A short overview about the disposal activities and the results is following.

1. Introduction

The Rossendorf Research Reactor (RFR) was in operation from 1958 to 1989. The power of the RFR was originally 2 MW and 10 % enriched fuel (EK-10) was used. During several upgrading up to a power of 10 MW the fuel was changed to the 36 % enriched WWR-M fuel type. All used spent fuel was unloaded and stored in a wet storage in the reactor hall until the year 2000. Between 1998 and 2000 all spent fuel was transferred by a special transshipment procedure from the wet storage into 18 so called CASTOR MTR2 casks. Every CASTOR MTR2 cask was transferred immediately after loading to a special buffer storage facility at the Rossendorf Research Site. In 2005 the 18 CASTOR MTR2 casks were transported to the central interim storage at Ahaus, near the western border of Germany, and still are stored there [1].
Furthermore two zero power reactors as well as nuclear fuel research facilities and an isotope production facility has been operated at Rossendorf from 1957 to 1991. The reactors used 20 % enriched fuel elements developed in Rossendorf as well as pellets and rings of metallic uranium (natural as well as 36 % enriched).
The main process of the isotope production facility was the production of fission molybdenum. The facility for Mo-production - so called AMOR - used short time irradiated fuel elements of the RFR (36 % enriched). After the Mo-99 separation, a high enriched uranyl nitrate solution had to be treated for storage and for disposal. In the nuclear research facilities have been worked with several kinds of nuclear material e. g. powder, pellets a. s. o. These different nuclear research activities, done in the times before the unification of Germany at the Rossendorf Research Site, caused that the VKTA - after its foundation in 1992 assigned to decommission the nuclear facilities at the Rossendorf site and to clear it from nuclear material and radioactive waste - had to take over more than 200 different nuclear material positions for treatment and preparation for disposal.

2   VKTA-conception for disposal of nuclear material

The VKTA-conception is determined by two main criteria: The definition of priorities and the choosing of the appropriate "disposal ways" to clear the site from the different nuclear materials.

The following priorities were confirmed:

Ranking:	Kind of nuclear material:
Top Priority	- gaseous (UF6) and highly enriched uranyl nitrate solution
Middle Priority	- fresh high and low enriched fuel (HEU, LEU) - spent nuclear fuel (SNF) ; HEU, LEU - Plutonium
Lower Priority	- uranium of natural enrichment - depleted uranium - thorium material
	Total Mass: approx. 10 Mg

The selected "disposal ways":

First objective was to find possibilities for a further utilization of the nuclear material.
For the fresh and spent fuel the preferred way was (and still is) the participation in the Russian research reactor fuel return program (RRRFR). For all the nuclear material below 5 % enrichment as well as for powder and liquids, special disposal projects had to be developed. Only the nuclear material for that no such convenient way for a further utilization is available, the conversion (transposition) of nuclear material into radioactive waste is the only alternative. All this different "disposal ways" the VKTA used contemporarily.

3   Disposal of nuclear material with top priority

In 1994 the first disposal project was finished successfully: As 78 kg of UF 6 were stored in a fixed installed pressure vessel, the vessel could not be used as transport container. For the disposal the UF 6 the gas was transferred in a certificated container of the URENCO Germany, which also took over this nuclear material.

The next project was the disposal of the high enriched uranyl nitrate solution still left on site from the former AMOR-process. After several talks and negotiations VKTA and BNFL agreed upon a disposal way to the reprocessing plant Sellafield. To get a license for the transport of the original high enriched liquid nuclear material and to meet the ingot conditions of the reprocessing plant, the blending of a sum of 4000 L uranyl nitrate solution down to an enrichment of lower than 2 % was necessary. Therefore a special blending plant in two mobile containers was constructed in installed in Rossendorf.

Fig 1. Mobile blending plant

After blending the solution was shipped in 43 high-grade steel drums (200 L each) in two 20ft containers to BNFL/Sellafield by truck.

Fig 2. NCS-Truck ready for shipment

In the Sellafield reprocessing plant the uranyl nitrate solution was converted into depleted uranium trioxide powder. This was shipped back to Rossendorf in 2005 and stored there.

The "disposal way" for fresh low enriched uranium oxide powder and pellets was offered NUKEM.

Two uranium disposal projects were carried out together with the NUKEM as supplier of nuclear material to the ULBA Metallurgical Plant in Ust-Kamenorgorsk in Kasachstan.
As the ULBA Plant accepts only uranium material up to 5 % enrichment, it was necessary to blend our fresh uranium oxide powder down to an enrichment of lower than 5 %. For this blending process a special drum mixer was used.

Fig 3. The special drum mixer for blending

In October 2006 the first shipment started with approx. 1.2 Mg uranium oxide material in 34 BU-D-drums in a 20 ft container to the ULBA Metallurgical Plant. The transport was carried out by truck and ship.

For a second disposal campaign extensive analyses of the uranium material were necessary to prove the acceptance conditions of the ULBA plant. In August 2010 the second shipment was finished with approx. 1.4 Mg uranium material to the ULBA Metallurgical Plant.

4   The participation in the RRRFR-Program

Already in 1999 representatives from the United States, the Russian Federation, and the International Atomic Energy Agency (IAEA) started discussing a program to return of Russian-origin highly enriched uranium (HEU) to the Russian Federation, the Russian Research Reactor Fuel Return (RRRFR) program. The primary goal of the RRRFR program was to prevent the proliferation of weapons-usable nuclear materials. More than 20 research reactors in 17 countries using Russian/Soviet origin HEU fuel were identified. One of these was the research reactor in Rossendorf which used highly enriched uranium.

It took some years to sign the necessary contracts and protocols before the return program started. First priority had the return of all the fresh high enriched fuel following by the return of the irradiated high enriched fuel.

The Rossendorf Fuel Return Project I

It also took some years to clear all the Germany-internal preliminaries and conditions and to get the necessary licenses for the return of the fresh high enriched RFR fuel back to Russia.

One of the largest fresh fuel return projects within the RRRFR-program finally was started in 2005 at Rossendorf: 268 kg of fresh high enriched uranium fuel (HEU) and 58 kg fresh low enriched uranium fuel (LEU) were to be shipped to the Scientific Industrial Association "LUCH" facility at Podolsk near Moscow for down-blending below 20% and further utilization for fuel for Russian power plants.

Fig 4. Monitoring by inspector

Together with VKTA ROSATOM, US NNSA/DOE and IAEA were involved in the Rossendorf fuel return project. The Free State of Saxony, the state of the Federal Republic of Germany where Rossendorf is situated, had to bear the cost of the return.

Transport by airplane was proposed as the safest and most unproblematic solution. A critical point was to convince the German authorities and to get the licenses for the air transport and for the use of the necessary 18 Russian transport casks of TK-S 16 type.

Fig 5. Project team with TK-16 cask

IAEA and EURATOM inspectors have monitored the loading process joined by experts from the U.S. National Nuclear Security Administration (NNSA) and from Russia.

Finally the fuel was airlifted on December, 18th, 2006 from Dresden Airport to Russia.

Fig 6. Airplane with loaded casks

<u>The Rossendorf Fuel Return Project II</u>

After the successfull first return project VKTA and its national and international partners engaged in the return project II. The objective of this Project is to return the irradiated fuel elements of the Rossendorf Research Reactor (RFR) which currently are stored in 18 CASTOR MTR 2 casks in the interim storage facility at Ahaus (approx. 164 kg HEU and approx. 186 kg LEU). The CASTOR MTR 2 casks are supposed to be transported in 20"-containers by truck, by vessel and by train to the Mayak facility. Like at the transport of the loaded casks from Rossendorf to Ahaus, also from Ahaus to Mayak three transports with 6 casks each will be necessary.

To fulfill all the technical, administrative, legal and political conditions and manage the licensing procedures of the transfer of the irradiated RFR fuel proved to be much more complicated than for the fresh fuel return project. Nevertheless supported by excellent cooperation of all the involved national and international partners most of the necessary arrangements were managed during the time and up to November 2010 it seemed to be possible to starting the first transport still before end of 2010. Although not only all the other European countries, who operated Russian type research reactors with high enriched fuel of Russian origin, already returned the irradiated fuel back to Russia, the German Federal Minister of Environment - unfortunately not earlier than late November - declared not to be convinced about the utilization of the material without detrimental effects. As just this is one of the conditions for the German export license, the first transport, scheduled in December 2010, had to be postponed.

How this problem is solvable now is not at last a sensitive political item. Whether and when we can go on with the return project currently is an open question.

5 Current status of the VKTA-conception for disposal of nuclear material

Up to date results concerning the reduction of stock piles of the VKTA nuclear material:

	1992	2011	
	Rossendorf Site	Rossendorf Site	Interim Storage Ahaus
HEU	ca. 600 kg	ca. 1.6 kg	ca. 164 kg
LEU	ca. 1,000 kg	ca. 21 kg	ca. 190 kg
Depleted Uranium	ca. 2,000 kg	c. 1,600 kg	------------
Plutonium	ca. 0.150 kg	0 kg	------------
Thorium	ca. 4,500 kg	ca. 4,500 kg	-------------

Tab. 1: The VKTA nuclear material stock

For the remainders of nuclear material still available at the Rossendorf site there are no prospects to find a possibility for a further utilization. Therefore all this material is treated and prepared for transfer into radioactive waste and will be stored in the respective interim storage at the Rossendorf site for later final disposal when a German final disposal site will be available.

6 Reference

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THE SHIPMENT OF RUSSIAN-ORIGIN HIGHLY ENRICHED URANIUM RESEARCH REACTOR SPENT NUCLEAR FUEL FROM BELARUS

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ABSTRACT

In October 2010, the Global Threat Reduction Initiative and the Joint Institute for Power and Nuclear Research – "Sosny" of the National Academy of Sciences of the Republic of Belarus completed a shipment that returned 43 kilograms of Russian-origin highly enriched uranium (HEU) spent nuclear fuel to the Russian Federation. The spent fuel was legacy material, discharged from the two decommissioned reactors, the Pamir-630D mobile reactor and the IRT-M research reactor. This shipment marked the complete removal of all HEU spent nuclear fuel from Belarus. This paper discusses the planning, preparations, and coordination required to complete this important international shipment successfully.

Introduction

In October 2010, casks containing highly enriched uranium (HEU) in the form of spent fuel assemblies (SFA) from the Joint Institute for Power and Nuclear Research – "Sosny" of the National Academy of Sciences of the Republic of Belarus were loaded onto rail cars and transported by train to the Russian Federation. The shipment was conducted under the Global Threat Reduction Initiative's (GTRI) Russian Research Reactor Fuel Return (RRRFR) Program which was created in 1999 as a tri-partite initiative between the Russian Federation, United States, and the International Atomic Energy Agency (IAEA) to return Russian-origin HEU from Russian-designed research reactors. The shipment of 43 kilograms of HEU SFA marked the successful completion of the accelerated project to deinventory all spent HEU from Belarus.

This paper discusses the various aspects of the shipment in detail including: historical background; organizations involved; preparation activities; coordination; and shipment logistics.

Background

The Joint Institute for Power and Nuclear Research – "Sosny" is located approximately 20 km from the city of Minsk, Belarus in the town of Sosny. The spent fuel that was returned was a product of two research reactors which were operated at the institute. The first reactor was a 4-MW IRT-M research reactor that was built in 1957 and achieved initial criticality in 1962. After years of successful service, the reactor was shutdown in 1988 and fully decommissioned by 1996. The majority of the spent fuel was returned to Russia between 1990 and 1991 while only some failed EK-10 fuel rods and experimental fuel spheres remained. This failed fuel was packaged into five sealed canisters and stored in the spent fuel pool for many years. The burn-up of this fuel was 7 – 20%.

The second reactor was a gas-cooled ($N_2O_4 \leftrightarrow 2NO_2 \leftrightarrow 2NO+O_2$) pilot reactor for the mobile nuclear power plant (NPP) "Pamir-630D". The mobile NPP had an electrical power output of 630 kW and included five basic modules: a reactor; a turbine generator; a system for control and protection; and an auxiliary module. The modules were installed on semi-trailers that could be transported by trucks. A picture of the reactor module on its semi-trailer is provided below in Figure 1.
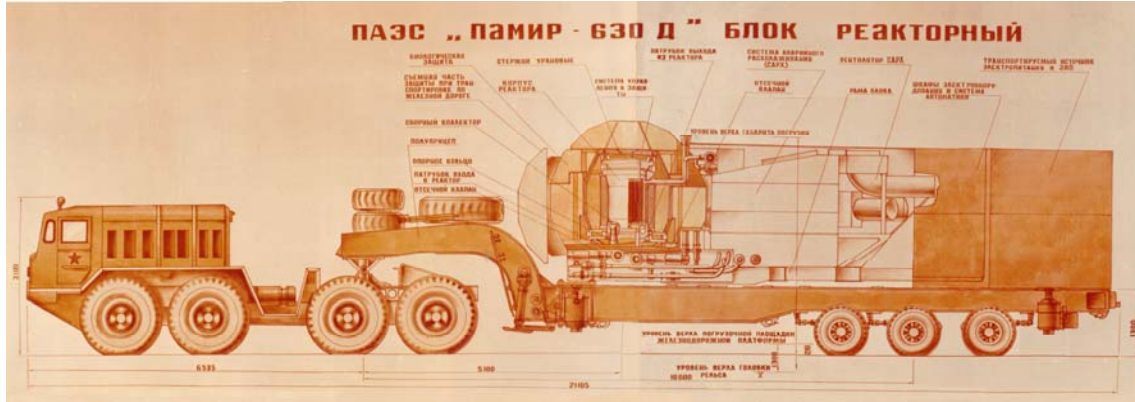

Figure 1 – Pamir-630D Reactor module on semi-trailer

The reactor core consisted of 106 fuel assemblies. Each assembly contained zirconium hydride moderator blocks and 7 fuel rods made of UO_2 particles enriched to 45% ^{235}U in a nickel / chromium matrix (UO_2-Ni-Cr) and clad in stainless steel. The reactor was put into operation in 1985 and testing was halted in 1986. The fuel was discharged in 1991 with an average burn up of 0.78%. All 106 spent fuel assemblies were stored in the spent fuel pool (one Pamir assembly was failed and was placed in a sealed canister).

The spent fuel pools are located in the same building where the pilot reactor was tested. Prior to the start of the project, only one of the two pools was operational with distilled water (pool#1). With a depth of 3.9 m, the pool provided adequate water shielding and had a capacity for 207 spent fuel assemblies and 10 sealed canisters.

Once the Government of Belarus declared its commitment to return the spent fuel, a project kick-off meeting was held in January 2009. From the beginning, the project had two major challenges. The first challenge was to determine where and how the Pamir spent fuel would be dispositioned. The Ni-Cr fuel matrix and stainless steel cladding of the Pamir SFA could not be reprocessed at Mayak using the standard processing line (used for UO_2 / UAl fuel matrix and aluminum clad fuel). A technology research project was conducted in parallel to the shipment preparations to determine how the Mayak processing line could be modified and the flow chart updated to accommodate the Pamir SFAs. The second challenge was the project completion deadline of September 2010, which was accelerated to support commitments made by the National Nuclear Security Administration (NNSA). The average project schedule for shipping spent fuel to Russia had been over 2 years; this shipment was to be conducted in 18 months.

Organizations Involved and Responsibilities

Organization	Country	Description and Responsibilities
NNSA	United States	National Nuclear Security Administration – division of the DOE that manages and funds GTRI and the RRRFR Program.
JIPNR-Sosny	Belarus	Primary contractor with NNSA and primary contractor with FCNRS. JIPNR provided project management and was responsible for all of the activities within Belarus.

Organization	Country	Description and Responsibilities
Gosatomnadzor Gospromnadzor Goscomvoenprom	Belarus	Belarus State Atomic Energy and Industrial Supervisory Authorities – nuclear regulator for Belarus. Approved the transportation Program, cask license validation, export license and other documents related to spent fuel transport in Belarus.
Rosatom	Russian Federation	State Corporation Rosatom - Provided Russian Government Agreements, cask and transport requirements, certificates and licenses.
Rosteknadzor	Russian Federation	Issued approvals of the State Ecological Expertise Reviews of the Special Ecological Programs and Unified Project.
Mayak	Russian Federation	Mayak reviewed spent fuel inspection data and provided transportation arrangements in RF. Provides temporary technological storage, reprocessing of irradiated fuel assemblies and its components, temporary technological storage of radioactive wastes.
FCNRS	Russian Federation	FCNRS – The only organization authorized to import spent fuel into Russia. Provided management of the Unified Project and Foreign Trade Contract.
Sosny	Russian Federation	Subcontracted by FCNRS to perform the safety analyses, prepare the required documentation, and obtain the licenses for the SKODA cask. Sosny also performed the technology research for processing the Pamir SFAs.

Legal Framework
Government-to-Government Agreement (GTGA): Unites States and the Russian Federation – *'Agreement Between the Government of the United States of America and the Government of the Russian Federation Concerning Cooperation for the Transfer of Russian-Produced Research Reactor Nuclear Fuel to the Russian Federation'*, dated May 2004. The GTGA provided the rules for participant eligibility, funding responsibilities, and program approval.

Diplomatic Note: United States and the Republic of Belarus – *'Agreement between the Government of the United States of America and the Government of the Republic of Belarus regarding assurances concerning the provision of technical assistance that may be provided by the Government of the United States of America, through the United States Department of Energy and its contractors, to support the transfer of spent research reactor nuclear fuel from the Joint Institute for Power and Nuclear Research "Sosny" of the National Academy of Sciences of Belarus to the Russian Federation'*, dated October 1, 2010. This Agreement provided liability protection and tax exemption for this nuclear nonproliferation project.

GTGA: Republic of Belarus and the Russian Federation – *'Agreement between the Government of the Republic of Belarus and the Government of the Russian Federation in the Field of Peaceful Use of Nuclear Energy'*, dated October 8, 2010. This GTGA provided the pathway and guidelines for returning the spent fuel and final disposition of the Radioactive Waste (RAW) in Russia.

SKODA VPVR/M Cask Specifications
For several reasons such as fuel specifications, cask availability, and facility configuration, the SKODA VPVR/M cask was selected for the shipment. The cask was specially designed for use by the RRRFR program and, with top and bottom loading capabilities, can be easily accommodated in most facilities. The cask is made of steel and aluminum and has a diameter of 1.2 m, height of 1.5 m, and weighs 12.4 mt fully loaded. Two basket configurations were used for the JIPNR spent fuel, the standard 36 SFA basket and the TS-3 basket which holds six canisters. Four casks were used to transport the 111 items (105

intact assemblies and 6 sealed canisters). The casks were transported in ISO containers, one cask per ISO container. In Russia, the certificate for design and transportation was issued by the Rosatom State Nuclear Energy Corporation on June 11, 2010. In Belarus, the certificate validation was issued by Gosatomnadzor on September 3, 2010.

Facility Preparations

New equipment, repairs, and enhancements were needed to the areas inside Building 40, to the JIPNR grounds, and at the rail station to implement the loading and transportation of the spent fuel. Building 40 contained the spent fuel pools and was where cask loading took place. Due to rules in Belarus, it was forbidden to handle radioactive materials at public railway stations. Consequently, a railway station on the territory of a military base was used. The major repairs or additions performed at each location were:

Building 40 and JIPNR grounds:
- The overhead 50 ton crane was inspected and certified for operation. A 2 ton hoist and trolley were added to meet the speed requirements for raising and lowering the basket.
- The security structure that covered both spent fuel pools was modified so that it covered only pool #1 and allowed for modifications to be completed on pool #2. Pool #2 was inspected, cleaned and filled with distilled water. The work platform surrounding the spent fuel pools was enlarged to allow for loading and leak testing of the SKODA casks. A metal stand was constructed over pool #2 to support the SKODA cask during loading.
- New remote cameras were installed throughout the working areas of Building 40 to assist with loading and observing. A central control room was setup to allow managers, supervisors and operators constant visual access during the loading process. Underwater cameras were installed in both pools to assist with fuel transfer and basket loading.
- New communications equipment was procured.
- All legacy equipment was removed from Building 40 (spent fuel pool area, room 111, and offices). Repairs and upgrades were made to the personnel spaces inside the building.
- To minimize the length of the route taken by the truck within the Institute grounds, a previously shutdown entry gate was refurbished. The roadway was assessed for the load capacity and repaved. The roadway was also widened in some areas to allow for improved truck maneuverability. A security area was setup outside the building for storage of the casks and ISO containers before and after loading.
- A new truck and trailer were procured to transport the ISO containers from the Institute to the rail station.

Rail Station:
- The loading platform at the rail station was cleared of all old equipment and reconstructed. Lighting was added and a backup generator was installed.
- The roadway from the entrance to the loading platform was resurfaced. An inspection area was constructed.
- The rails were inspected and all debris was removed.

Shipment Logistics Summary

Authorization to import the spent nuclear fuel into Russia was received using the procedure mandated by Russian decree. A Unified Project (UP) was developed and approved after a positive conclusion by the State Ecological Expertise Review Board. The UP is a collection of analyses and documents that look at: justifying the safety and assessing the environmental impact of import; implementation of special ecological programs; anti-terrorist measures; cask licenses; emergency response; and the Foreign Trade Contract (FTC). The FTC was the main contract between FCNRS and JIPNR-Sosny to import the spent nuclear fuel. The FTC includes transportation, reprocessing, storage and disposal. Once the UP

was approved, a governmental decree was issued along with the import license. In Belarus, transportation permission was supported by documents looking at: physical protection; emergency response; cask licensing; facility license; and environmental and health concerns. The transportation permission and export license were issued by the State Industrial Supervisory Authority, Gospromnadzor.

By the beginning of September 2010, the last of the four casks had arrived, the final equipment setup and testing and personnel training had been completed and the facility was ready for loading. A truck was used to back each ISO container into the room that was located below the main hall where operations were conducted. The SKODA cask was lifted from its ISO container and positioned in the main hall in its designated storage location. Prior to the loading of each cask, the specified number of SFAs were verified and transferred (via a transfer cask) from pool #1 to pool #2 in preparation for loading. Residual activity measurements were performed on each SFA using detectors installed over the pool. A cask was then lowered onto the working platform surrounding the spent fuel pools where the impact limiters and secondary lids were removed. The SKODA cask was placed over pool #2 and the basket was connected to the hoist and handling rods. The basket was lowered into the pool and the fuel assemblies were loaded into the basket according to the loading schemes. An IAEA inspector performed a measurement on each SFA to record the cesium peak to verify the presence of spent fuel. After the last SFA was loaded into the basket, the basket was raised and secured into the cask. The loaded cask was then placed in the cradle where the secondary lids were installed and the vacuum drying and leak testing were performed. IAEA inspectors placed two seals on the top secondary lid per safeguard requirements. Once the cask passed the leak test, the impact limiters were installed and the cask was placed in its designated storage position on the main level of the main hall. An IAEA safeguard seal was also installed on the impact limiter. The cycle was repeated for all four casks. 35 SFAs of "Pamir-630D" reactor were loaded into standard baskets in the first three casks. The fourth cask contained the TS-3 basket which was used for the 6 sealed canisters (5 canisters with SFAs of IRT-M reactor and 1 canister with SFA of "Pamir-630D" reactor). The loaded casks were lowered from the main hall to their ISO containers and stored outside until the day of transportation to the rail station.

The loaded ISO containers with the SKODA casks were transported one at a time by truck to the area where the rail station was located. Special Forces guards were located approximately every 200 meters along the route from the Sosny Institute to the entrance of the rail station, including extra guards at overpasses and bridges. All traffic was diverted or stopped while the material was on the roadway. Each round trip took about 2 hours to complete. Under heavy guard by Belarusian Special Forces, the train departed for the Belarus / Russian Federation border. Careful considerations were given on all details of roadway/railway transportation from the territory of JIPNR-Sosny up to the Belarus / Russian Federation border, and, as a result of this detailed planning there were no incidences or issues during the transport. At the border, the responsibility for guarding the spent fuel was transferred to the Russian guards. The locomotive was also exchanged for a Russian locomotive. The travel time from the border to Mayak was approximately 3 days. As indicated earlier, modifications are required to the reprocessing process and equipment to dissolve the spent fuel assemblies. The modifications are scheduled to occur in 2011 with reprocessing to follow soon after.

Conclusion
The shipment of 43 kilograms of highly enriched uranium from the Joint Institute for Power and Nuclear Research – "Sosny" marked the end of a successful project that removed all of the spent HEU from Belarus. This project was an excellent example of international cooperation between the various organizations within Belarus, Russian Federation, United States, and the IAEA. It also demonstrated how Belarus continues to support global nonproliferation programs.

THE HOR NUCLEAR INSTRUMENT CHANNEL REFIT

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ABSTRACT

The research reactor in Delft, the HOR was built around 1960. Because of ageing effects, the nuclear instrumentation was completely replaced in 1980, together with the construction of a new control room outside the containment. In 2010, after 30 years of successful operation, it became a real challenge to repair the nuclear channels because of obsolete components. Of course, this problem was identified earlier, and a project was started in 2008 to select and replace the electronics of the nuclear channels. For this purpose a European tender was started to select a manufacturer for the new electronics in accordance with the requirements. The boundary conditions to be fulfilled by the manufacturer were: a) The functionality of the instrumentation and the interface to the plant should remain the same, and b) The proposed type of equipment should have been installed and commissioned successfully at other research reactors of comparable type earlier. Only the electronics should be replaced, detectors and cabling are reused.
Parallel to this tender we started discussions with the authorities to clarify which standards the instrumentation should fulfil.
We selected a digital system based on two microcontrollers, each one checking the other one.
It turns out to be a flexible system. It was easily adapted to our needs, showing adequate provisions for guaranteeing data integrity.
In the summer maintenance period of 2010 the instrumentation was successfully installed and commissioned. This paper will describe the steps taken and the tests performed.

1. Introduction

The Reactor Institute Delft (RID) is part of the Faculty of Applied Physics which belongs to the Delft University of Technology. The RID is the Dutch national centre for multidisciplinary research and education involving the Hoger Onderwijs Reactor (HOR), nuclear radiation and radionuclides. Until 2005 the institute was an interfaculty facility. Since 2005 the interfaculty facility is split in the new research department 'Radiation, Radionuclides and Reactors' (R3), and the Reactor Institute Delft running the research infrastructure. Together they form the national focal point of expertise in the fields of reactor physics, neutron and positron beam research (including radiation detection), as well as radiochemistry. At the heart of the reactor institute is the HOR as a source of neutron, positrons and radioisotopes. The HOR is a pool type research reactor operated 24 hours per day, 5 days a week at a thermal power level of 2 MW, attaining a steady thermal flux of about 2×10^{13} n/cm²/s.

In the sixties, the HOR was built with a control room inside the containment from which the operator had a good view at the reactor. At that time the reactor instrumentation was using vacuum tube equipment. Over the years, the power and the operation time per day were increased considerably. The instrumentation aged and the control room was uncomfortably small for 24-hour shift operation. In the early eighties a new control room was built outside the containment with new instrumentation, detectors and cabling. A computer for data acquisition was introduced. With this instrumentation, the reactor was operated successfully for 30 years until 2010. Ultimately, it became more difficult to calibrate and repair the equipment because of components becoming obsolete. To cope with the ageing phenomenon a project was started in 2008 to select and install new instrumentation.

2.   Project HOR nuclear instrumentation refit

2.1   Scope

The scope of this project is confined to renewing only the aged electronics in the control room and the preamplifiers in the "field". The detectors are still in good condition and spare parts are available on stock. New detectors of the same type could be procured if necessary. The cables are reused and for ensuring their good condition, they have been retested during the commissioning phase.
The design of the safety system is not changed. All new nuclear channels have the same functionality as the channels of the old system. The reactor protection system is not renewed in this project, so the interface of the new channel system to the reactor protection system remains unchanged.
A no-break installation for the power supply of the instrumentation is not required for safety functions because of the "failsafe" design, but for uninterrupted data registration purposes it is preferred to have this option. Therefore the manufacturer was also required to implement Uninterruptable Power Supply's.

The nuclear channels that have been replaced are listed in table 1.

Channel name	Channel type	Number of channels	Safety class
Neutron Flux level	Safety	4	1
Pool Gamma Monitor	Safety	1	1
Bridge Gamma Monitor	Safety	1	1
Stack Off Gas Activity	Safety	1	1
Pool Outlet Gamma Monitor	Safety	1	1
Primary Coolant Flow	Safety	1	1
Differential Pressure Heat Exchanger.	Safety	1	1
Lin/Log Channel	Process	1	2
N16 Reactor Power	Process	1	2
Fission product activity (forced cooling)	Process	1	2
Area Gamma Monitor	Process	3	3

Table 1: Channels replaced

2.2   Requirements

As the Reactor Institute is part of Delft University of Technology, according to Dutch rules a European Tender Procedure is obligatory for selecting a manufacturer/supplier. For this purpose, general requirements and instrumentation requirements were formulated and incorporated in an invitation for tendering. The most important requirements to be fulfilled are:

- Every instrument channel should have the same functionality as the old channel
- Interface to the plant should be kept the same, using the existing detector and reactor protection system
- The proposed type of equipment should have been installed and commissioned successfully at other research reactors of comparable type earlier
- Response time of the new equipment should be equal or faster
- Standards: KTA 3501/3505 or equivalent, IEC80660 for software purposes, ISO9001 QA system
- Flexibility to adapt to future needs, for instance a power increase

- Calibration and testing should be user friendly, e.g. without the need for disconnecting cables
- Installation should take place in the summer maintenance period of 2010

2.3 Project milestones

- June 2008: approval of project proposal by the Dean of Applied Physics.
- June 2008: start of preparation for the European tender.
- 2008 Start of discussion with the supervising authorities: In good cooperation, it was agreed to accept the qualifications of the manufacturing country of origin.
- October 2008 Specifications are written.
- March 2009: Selection of the manufacturer, contract with the manufacturer is signed.
- 2009: Several meetings with the authorities: According to KTA regulations the Factory Acceptance Test is witnessed by an independent third party. For this purpose and with the consent of the authorities we selected the TUV-Nord.
- The software of the safety channels is certified and approved by the TUV-Nord by a dedicated, independent software department.
- November 2009: Detailed proposal sent to the authorities.
- June 2010 Proposal accepted by the authorities.
- June 2010 Factory acceptance test (2 weeks): Performed by the manufacturer and witnessed by the TUV-Nord, RID and the authorities.
- July 2010 Start of installation, by the manufacturer and RID-employees.
- July 2010 (end): Site Acceptance Test: Performed by the manufacturer, the RID and witnessed by the authorities.
- August 2010: start-up (2 weeks): Started with "cold" test, followed by the "hot" test with step increases in reactor power.
- August 16 2010: Reactor at its nominal power for regular 24-hours/day operation as scheduled.

2.4 Old versus new

Following the tender procedure, Mirion Technologies (MGPI H&B) GmbH was selected as the manufacturer/supplier of the new nuclear instrumentation. All the channels with a radiation detector are based on the Digital Signal Processing Channels TK250. This system is a flexible modular system that could be adapted for the functionality we needed in every single channel.
These channels have remote signal generators and signal simulation built in and are qualified according to KTA 3501/3505, IEC 60 880.

One of the big differences beside the digital implementation is the auto ranging feature in most of the channels. In the old system the range was set with a range switch. The trip is set to 100% of the selected range and thus depending on the position of the range switch. In this case it's not possible to switch to a more sensitive measuring range because the trip will also have a different value.
The new channels have a fixed trip which is stored in EEPROM and is not changeable by the operator. The range switching is done automatically by the software. The advantage is a better view of the process parameters during start-up of the reactor due to the use of the more sensitive measuring ranges.

For power control of the reactor, a combination of N16 for the integral absolute power measurement and a wide range channel for measuring the fast power changes is used. This wide range channel has a logarithmic and a linear output (saw tooth) with 18 ranges over the total power range. Switching to a lower range for the linear channel is done automatically during power decreases. However, for power increases the operator has to activate range up switching by pressing a button. The operator is guided with a lamp in the button if the linear

signal exceeds the 80% level in the current range. At 135% of each linear range, a trip signal for the Automatic Rundown of the control rods is generated, in conformity with the requirements. With full automatic range up-switching the 135% level trip would only be reached in the highest range. In order to comply with the specific requirements in force for controlling the HOR during power increases, we implemented this kind of semi-automatic range up switching.

3   Implementing Flow Blockage Protection

The HOR is an open pool reactor. From this point of view there is a possibility of a flow blockage of a cooling channel caused by an object dropping into the pool during operation of the reactor [1]. Subsequently, the object may be trapped on a fuel element upper inlet section, thereby blocking the coolant flow into one or several channels. As a consequence, the flow in the affected cooling channels will either be reduced or completely interrupted. The channels will dry out and the plates will heat up rapidly. If the reactor is not scrammed in time the melting temperature of the fuel plates is reached. Voiding of coolant channels causes the neutron flux to decrease by the feedback mechanism on reactivity.

3.1   The previous analogue implementation

The previous HOR protection system was equipped with a power Shut-Down Amplifier with a special feature. In addition to the usual protective function regarding the overall neutron flux level, they respond to sudden changes in neutron flux. The current signals are received from ionization chambers (four-redundant). If the input current deviates excessively from an automatically adjusted internal reference level, a trip condition is generated.

To generate a trip, the current from the ionization chamber is converted to a proportional voltage representing the power signal. From this power signal a power reference signal is generated internally as a parameter that is electrically adjusted in a closed loop control system. The system maintains a fixed ratio between the power reference signal and the power signal at steady state. Both signals are compared and the resulting difference signal is amplified to provide the Margin signal. This signal is a sensitive indicator of the input current variations which is used for detection purposes.

As the power signal varies, the power reference signal will vary in sympathy with it, and produces the margin signal equal to the "demanded" margin. Under transient conditions with power decreasing, the power reference signal can only follow the power signal within the limits of the maximum decreasing tracking rate of the system. If the power decreases faster the margin signal deviates from its quiescent state until the excess margin trip condition is reached, generating a reactor scram. The requirement for this system is to generate a trip within one second at a power decrease of 10%.

3.2   The digital implementation

The replacement for this special shutdown amplifier should be a fast system. The manufacturer selected the DGK250 from their TK250 range. It is a neutron flux safety channel for the power range with the possibility to have 2 signal paths for two detectors. One signal path is used for the power signal and the other one is used to calculate the margin signal by the software. This HOR system uses only one detector per channel. The response time is <9ms. The signal flow diagram is given in figure 1.
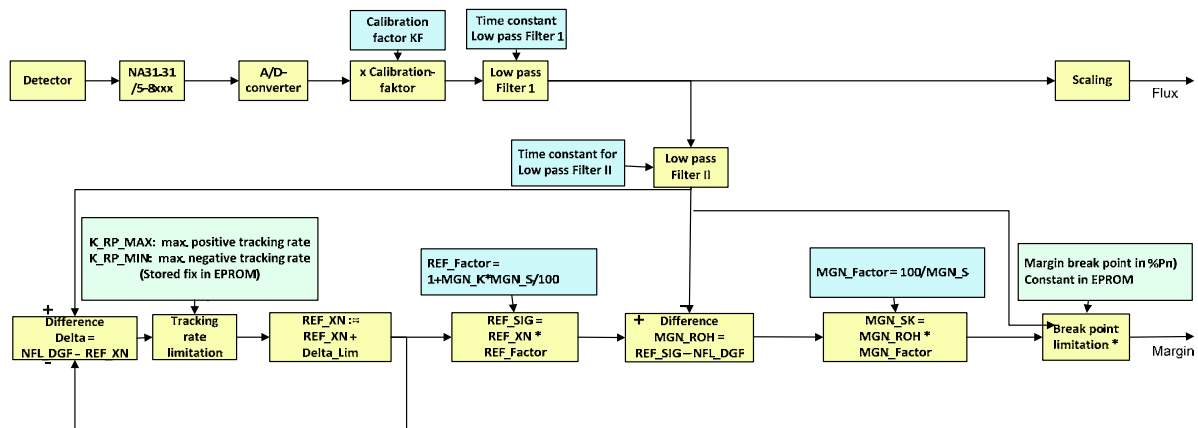
Calibration factor KF

Time constant Low pass Filter 1

Detector → NA31.31 /5-8xxx → A/D-converter → x Calibration-faktor → Low pass Filter 1 → Scaling → Flux

Time constant for Low pass Filter II

Low pass Filter II

K_RP_MAX: max. positive tracking rate
K_RP_MIN: max. negative tracking rate
(Stored fix in EPROM)

REF_Factor = 1+MGN_K*MGN_S/100

MGN_Factor= 100/MGN_S

Margin break point in %Pn) Constant in EPROM

+ Difference Delta = NFL_DGF − REF_XN −

Tracking rate limitation

REF_XN := REF_XN + Delta_Lim

REF_SIG = REF_XN * REF_Factor

+ Difference MGN_ROH = REF_SIG − NFL_DGF

MGN_SK = MGN_ROH * MGN_Factor

Break point limitation * → Margin

Figure 1: Signal flow diagram DGK250 (provided by courtesy of MGPI)

The reference and margin signals are calculated according to their analogue electronic equivalent. In the previous analogue electronics version the set points like the tracking rate are adjusted by potentiometers. In the new digital version the safety critical parameters are stored in EEPROM and can not be changed by the operator. Calibration factors can be changed but are protected by procedure and a special key to unlock parameter writing.

To test the system performance during the design phase a software simulation provided by the manufacturer was applied. The proper functioning of the software was tested with this tool, also checking if the requirement of a trip in 1 second at a sudden decrease of 10% power could be met.

During the start-up phase following the cold test, a live test was performed to check if the trip on "excess power differential" was excited, thereby scramming the reactor. In accordance with the test protocol, at a power of 1 MW one of the 4 control rods was dropped into the core to create a sudden power decrease of at least 10%. Indeed, as a result the other 3 remaining rods dropped into the core instantaneously following trip excitation.

4   Lessons learned

- We started in an early stage discussion with the authorities. This gave us the possibility to respond to their questions in due time. For instance the authorities asked to have a third party analyse the suitability of the new equipment for the Delft application. For this purpose we hired the TUV-Nord to do an assessment and check if the old and new channels have the same functionality.
- In the discussions with the authorities we discussed the way how the manufacturer handles the protection against Common Cause Failures. For this purpose IAEA Technical Report NP-T-1.5 is used [2]. With the overview in this document the additional points of CCF (software) were checked against the measures taken in the channel design. The results were discussed with the Reactor Safety Committee and with the authorities [3]. The conclusion is that the set of measures that is provided in the channels are adequate in preventing CCF as stated in IAEA Technical Report NP-T-1.5.
- For the channel Pool Outlet Gamma Monitor the response time is of importance. This channel is the second line of defence in the safety system and trips in the case that fission products are released to the primary coolant in significant quantities. This channel plays a role in the safety analyses reported by Siemens [4]. Due to the signal filtering with a time constant depending on the input signal, the response time became too long. We decided to use the unfiltered (raw) signal for the trip generation

and to use the filtered signal for the analogue output. The analogue output is used only for data registration. We didn't expect spurious trips by the non-filtered signal because the signal is quite large during normal operation. In this way the safety analyses are still valid.

- The range switching in the gamma channels is done by software and is more or less artificial. It's not needed for the sensitivity of the system but implemented to make the difference between the old and the new system smaller. Therefore also the number and the end of the ranges are kept the same as in the old system. It turned out to be very practical; the operators still have the same "feeling" with the reactor.

- The European tender process is a time consuming process caused by certain pre defined time steps. We gain benefit from an internal Procurement department which helps us with the legal issues for the tender.

- In cooperation with RID, the manufacturer organised training sessions for the operational staff. These sessions were held before the actual commissioning. Test channels were used for this purpose, with the same software and Dutch menu structure as the channels that have been installed. The staff has been trained in signal processing in the different channels, maintenance and periodical testing, including practical exercises. The training effort turned out to be very effective for a smooth transfer from the old to the new situation. All staff received a certificate after successful course completion.

- The new instrumentation is qualified according to the strict rules of KTA 3501/3505. Therefore every module that needs repair or testing is sent back to the manufacturer. This is a new requirement with respect to the old situation. To cope with this new situation we have 2 spare modules of every different module.

5.   Conclusions

- The MGPI system turns out to be a very flexible system that could easily be adapted to our needs.
- Testing during checkout is user friendly by using the built-in signal sources and output simulations.
- The time for the operating staff to adapt themselves to the new situation was minimized due to good preparation and training.
- The project was done in time and within the budgetary planning.
- Until now the equipment showed good performance without any errors in the channels.

6.   References

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THE REPAIR OF THE HIGH FLUX REACTOR IN PETTEN

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ABSTRACT

On August 20, 2008 a bubble jet was discovered in the primary cooling water system of the HFR (High Flux Reactor) at Petten. The possible impact of this phenomenon on safety was unknown. Hence NRG, the operator of the nuclear reactor, decided to stop normal operation and started a concerted action to investigate the root cause and to develop a return-to-services programme. The programme included a plan to restart the HFR and to prepare for a full final repair of the cooling water system. The time needed to prepare the repair was estimated at one year. Additional technical and organisational measures were put in place to maintain safety and after internal, national and international review, the plan was approved. In February 2009 the HFR was back in operation for one year on a temporary license and NRG prepared for the final repair.

The approach for the final repair project was to involve specialists from day one and to develop and test each step of the project at work bench scale and then in 1:1 scale models. Accurate engineering and thorough testing proved its value: the project proceeded well and according to plan. The execution of the project started February 19, 2010 and the HFR returned to services September 9, 2010.

1. Introduction

Since the early sixties, the High Flux Reactor (HFR) in Petten has been a vital link in the chain of nuclear facilities in the Petten dunes. Nuclear Research and consultancy Group (NRG) operates these facilities and employs 360 people. NRG develops and supplies sustainable nuclear technology, with applications in power generation, the environment and healthcare.

Utilizing the HFR, NRG has grown into a key supplier of medical radioisotopes; with about 60 % of European demand produced in the HFR at Petten. The HFR runs around the clock, with on average 10 cycles of 28 full power days per year.

2. Deformations in the cooling water system

Each summer the HFR is shut down for maintenance and inspection. In summer 2005, deformations were detected in two parts of the reactor's primary cooling water system. These parts are called "reducers", conical pipes connecting the wider diameter of the reactor cooling water outlet with the smaller diameter of the cooling water pipes.

The reactor is located in a concrete pool with floor and walls some meters thick, lined with aluminium. For cooling and radiation protection purposes the pool is filled with water 9 meters deep and the top of the pool is open to enable reactor loading and unloading during operation. The reducers and connecting cooling water pipes are embedded in the concrete at the bottom of the reactor pool.

3. Investigations to find the cause

Investigations were started to find the cause of the deformations in order to determine their possible impact on the safety or reliability of the HFR. We also decided to increase the frequency of the periodic
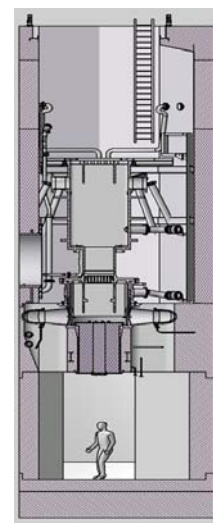

Figure 1. Cross section of the HFR

inspections of both the reducers and the deformations. The hypothesis was developed, that on the outside of the reducers, a combination of carbon steel, moisture and aluminium initiated a galvanic corrosion process causing a decrease of the reducer's aluminium wall thickness. The corrosion product has a volume 2 to 3 times greater than the volume of aluminium. As the concrete was unlikely to deform, it was likely that the reducer wall would be pushed inwards, thus causing the observed deformations. From 2005 onwards additional investigations were executed and we involved experts from many different fields in order to avoid "tunnel vision".

4. Safety and reliability

The very first question we raised after detection of the deformations was: "What is the impact on safety?" Only when safety is ensured, can the reactor be operated. To ensure sustainable supply of medical isotopes, the next very important question was the reliability of the reactor, both short and long term.

To answer these questions we frequently updated our safety analysis with new data from our investigations. From this we learned that the risks from the deformations were well within safety limits; however, the risk for unscheduled downtime was increased.

We decided to set-up a project to look for solutions. At first we considered repair to be very difficult or even impossible and wanted to investigate options other than total repair. We hired a British engineering company with broad experience in the nuclear industry. One of the first steps in the project was to define a decision matrix and decision criteria.

The position of our isotopes customers also played a crucial role. Both minimum down time of HFR for repair and speed of availability of solutions were high on the priority list. The latter was important, because early availability of a repair solution could prevent a long down time in the case of an unexpected failure of the reducers.

With these criteria at hand, we reviewed many repair options. A few, potentially successful solutions were engineered further in order to substantiate the decision making process. By this time it was Summer 2008.

5. A gas bubble jet and a decision with major impact

In agreement with the coordinated operating schedules of the main medical isotope production reactors, July 28, 2008 was the first day of the HFR summer maintenance stop. The planned date for the restart was August 24 and inspection of the reducers was scheduled. At this time we were still working on repair options for the reducers.

The inspection progressed according to plan; the results confirmed again our view of the situation. The deformations had grown slightly; but wall thickness had decreased very little. We were glad that we had already started to prepare for repair of the reducer.

On Wednesday, August 20, 2008 at lunchtime, with a flushed face, the inspection team leader interrupted a project progress meeting. The inspection team were performing final measurements when unexpectedly they saw a gas bubble jet released from one of the deformations. With some disbelief their first action was to keep the camera focussed on the spot and start the video recorder.

By the next day we had learned that the gas bubble jet emerged at a frequency of once every 10 – 20 minutes and lasted 10 to 30 seconds each time. What followed was a discussion about these observations, a first exchange of ideas about the origin and cause of the gas bubbles, decisions about possible extra investigations, preliminary considerations about the possible consequences and finally decision making within the management team about "go or no-go" for the planned HFR restart. We were fully aware of the impact a "no-go" decision would have on all our customers and on the patients who are dependant on the radioisotopes from Petten. At that point in time, the cause, extent and possible safety impact of the observed phenomena on the reactor were unknown; so we decided not to restart HFR.
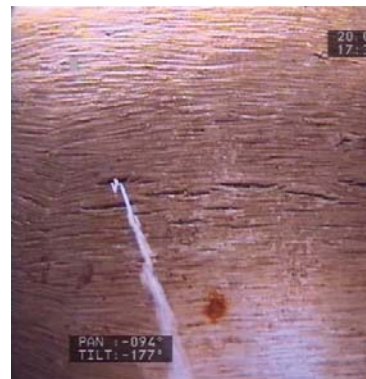

Figure 2. Gas bubble jet on the inside of the reducer

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6. Mitigation of consequences

During the days after this decision, all possible powers within and outside NRG were mobilised to try to mitigate the consequences. The highest priorities were communication with customers and colleague reactors in order to minimise impact and to immediately set up of an HFR return-to-services project.

This project consisted of 3 main elements: 1) Investigation and analysis of the condition of the reducers. 2) Find a solution where the HFR could re-start on the shortest possible time path. 3) Find a long term solution ensuring a reliable HFR.

It was clear, that whatever the solution, there would be no compromise on safety. For a short term solution we were prepared to compromise on operational reliability, if necessary.

7. HFR return to services

Prior to the discovery of the gas bubble jet, we had been developing potential repair methods. At this point we changed our approach based on the experience we had gained; our new strategy was now to find a company with extensive experience with this kind of repair and to quickly develop a temporary repair solution. In parallel they should also develop the final repair method. We selected a Swedish company who specialised in repair projects in the nuclear industry.

Quite quickly we decided to start detailed engineering on the so-called "sleeve option" for the temporary repair; an option already considered earlier that year. The idea was to install a specially designed pipe inside each reducer. However, during detailed design we encountered insurmountable technical problems.

In the mean time, it became clear that for the final repair, it would be necessary to either locally repair the corroded parts of the reducers or to replace both reducers completely. The only way this could be done successfully, would be to make the reducers accessible by cutting two large cavities in the concrete in which the reducers were embedded. Initial analysis showed that two cavities in the concrete could be accommodated from a constructional point of view. Preliminary calculations indicated that radiation levels at the repair sites could be reduced to an acceptable level by installing extra shielding material in the reactor to compensate for the radiation shielding effects of the primary water, which would need to be drained.

Confidence in the "concrete route", as it was called by that time was growing; while confidence in the "sleeve option" had decreased rapidly. Most importantly, our understanding of the actual status of the reducers and their impact on safety and operational reliability had increased a great deal since the first discovery of the gas bubbles. An important new fact was the clear confirmation that there was no water leakage at the reducers.

We estimated that we needed about one year to fully prepare for the "concrete route". We also judged that with additional detection systems for the possible but unlikely leakage of water from the reducers, safety could be maintained. This would allow the HFR to be returned to services temporarily while we prepared for the full repair. Comprehensive safety analysis confirmed our assessments.

Our plans and supporting analysis were reviewed by national and international experts and the International Atomic Energy Agency (IAEA) sent an expert mission to assess in great detail all the safety aspects of the project.

Late on February 12, 2009, we received a temporary license to start and operate the HFR. Three important conditions were connected to the license. The HFR would only operate when necessary for medical needs; if any water leakage from a reducer was detected, then HFR would be immediately shut down and we must start execution of the project of the definitive repair of the reducers by March 1, 2010.

As all preparations for start-up were already in place, we were able to start the HFR that evening. Before midnight we were running at full power and had resumed production of radio-isotopes.

8. Prepare for final repair

The first step towards the repair was setting up a dedicated project organisation. The author of this article acted as the project executive and appointed an experienced project manager

to create and lead the project team. Specific expertise and strong teamwork characteristics were the main selection criteria.

People both from within NRG and from outside companies were appointed project team members. The project objective was to safely repair or replace the two reducers by gaining access through the concrete ceiling of the sub pile room, the space directly under the reactor vessel.

The repair should start no later than March 1, 2010 and in order to minimise the negative impact on the isotopes market, the reactor outage time for the repair project should be as short as possible without jeopardizing safety or quality of work. The quality of our planning and ability to accurately forecast the date of return to service of the HFR at the end of the project was also of utmost importance.

A major decision early in the project was to drain the reactor pool at an early stage of the work as this concerned the safety of the workers. Calculations had indicated that the pool bottom would be strong enough after removal of the concrete from the cavities, but to be absolutely sure there was no risk of flooding by the pool water, we decided to drain the pool. This decision had far reaching consequences, as we needed to install alternative radiation shielding material on the pool bottom. We selected a tungsten alloy with a density of 18 000 kg / m^3. Prior to draining the pool, we also needed to cover the open top of the pool with a concrete cover in order to reduce overall radiation levels within the reactor building.

Our approach during the design and engineering phase was to involve sub contractors from day one and to develop and test, preferably several times, each step of the project with 1:1 scale mock-ups.

For the concrete removal and restoration we built a replica of the section with one of the reducers. We found a Swedish mine which was able to supply the same high density concrete filler (magnetite) that was used 50 years ago. By practising concrete removal on the replica, we learned which tools and work processes were the best and staff doing the work gained experience in working in a limited space like the sub pile room.

For the actual repair of the reducers, several techniques were investigated and tested at both work bench scale and on 1:1 models. We tested a relatively new technique for increasing material wall thickness by means of "cold spraying", however this method could not be qualified for application in a nuclear facility, because there were no relevant standards. Welding was selected as the repair method and criteria were developed for the decision process to either repair locally or totally replace one or both reducers. The qualified welders practiced both repair options in models of the narrow cavities.

9. Internal and external reviews

Our repair plan, the safety plan, the radiation protection plan and the detailed plans of the sub projects were modified and reviewed several times as part of the engineering process. To ensure the quality of all aspects of the project, internal and external reviews were a continuous part of the process. To mention a few: at Petten there are two permanent committees, the HFR Safety Committee and the Petten Reactor Safety Committee. Specifically for this project we also installed a Technical Expert Advisory Group. In December 2009, an international Peer Review Team was invited to Petten for in depth discussions and project review. In early 2010, an IAEA mission was in Petten again for thorough discussions and project review. During the whole process, the Dutch national authority, the Kern Fysische Dienst (KFD) supervised each project phase and all the reviews.

By mid January 2010 all documents were approved and the project was ready for execution.

10. The repair

The repair started February 19, 2010 and after removing the reactor internals, like fuel elements, control rods, isotope production rigs and experimental facilities, the first major test for the project team was the installation of the radiation protection equipment inside the reactor and shielding blocks at the bottom of the reactor pool. Accurate engineering and thorough testing proved its value and the installation went according to plan. After covering the pool top with the concrete slabs and draining the water, the radiation levels were

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measured. It showed that the shielding had reduced the radiation levels by the required order of magnitude, but at some locations the levels were still too high. With these measurements in hand, additional shielding was designed, made and fitted. This proved to be adequate and after draining the reactor and pool, we started removing the concrete. Using a hollow-drill-technique holes were cut of various lengths depending on the position of the reducer inside the concrete. The total length of the holes was more than the height of the Eifel Tower. For this type of drilling, practise and "touch-and-feel" are the key to success, as we had learned by training on the mock-up. On one occasion during drilling, a flange on the outside of the reducer was hit. The flange position was slightly different from the 50 year old drawings. We had already planned to replace the flange and we were happy to have real evidence of the exact location and condition of the reducer and flange. The last of the concrete was removed by cutting with hand tools, to avoid damaging the reducers.

After cleaning, the next step was thorough inspection and measurement of the reducers. Until this point in the project, we had been forced to rely on our hypothesis of the cause of the deformations and had to be alert that it could prove to be incorrect. We were able to confirm that the cause of all the deformations was galvanic corrosion, initiated by the combination of steel, aluminium and moisture. The reactor pool has an aluminium liner which is designed in such a way, that when small amounts of pool water are released due to thermal movement, they are transported to a drain system through small channels in the concrete. Some of this moisture migrated through the concrete and reached the reducers.

A very important point was that we found that the rest of the reducer material was in perfect condition and the criteria for local repair were met. In both reducers we cut a larger hole at the bottom and a smaller hole at the top. All four holes were closed with pre-fabricated plates of new aluminium. The top plates were welded from within the reducers through the larger holes and then the bottom plates were welded from outside. Despite many trials, the welding in situ proved to be more difficult than expected. After several improvement actions we succeeded in producing welds which passed the tests in accordance with the relevant standards.

After welding was completed, the cavities were filled with reinforcement bars and concrete. To prevent air being trapped during the pouring of the concrete, we installed four vents in each cavity and a camera was inserted through a vent pipe to enable monitoring the concrete filling process.

The final project phase was recommissioning the HFR equipment and controls, followed by extensive testing of all systems prior to reloading the core and starting up the HFR. September 9, 2010 we started the reactor, reached full power and resumed the isotopes production.

Figure 3. Welding

11. Evaluation and outlook

The repair project proceeded well and according to plan. We have had no safety incidents. The individual and collective radiation dose has been about half of the quite stringent limit we put on ourselves at the start of the project. Only the individual doses of the welders were close to the set limit. We created a learning attitude within all teams involved and during execution, several safety improvement proposals were made and implemented. Our project planning involved of all parties in interactive sessions leading to a project plan and detailed work plans of high quality. During project execution, at the times when reality deviated from planning, we were able to maintain the same attitude. We solved the problems with focus on safety, quality and a proper decision making processes.

To secure the long term medical radioisotopes supply, we have intensified our maintenance and reliability program for all nuclear facilities at Petten. We expect the end-of-lifetime of the HFR to be around 2020. To secure the strength of the existing chain of the nuclear facilities at Petten for the long term, we are developing the replacement of the HFR by our new reactor, PALLAS.

5

THE IMPLEMENTATION OF AN INTEGRATED MANAGEMENT SYSTEM FOR TRIGA RESEARCH REACTOR AT LENA (LABORATORY OF APPLIED NUCLEAR ENERGY) - UNIVERSITY OF PAVIA (ITALY) –

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ABSTRACT

The Laboratory of Applied Nuclear Energy ("LENA") is an Interdepartmental Research Centre of the University of Pavia which operate, among other facilities, a 250 kW TRIGA Mark II Research Nuclear Reactor. The reactor is at the disposal of researchers from Pavia University and of other users, both public and private, for research activities, training and education and other services. The Centre itself carries out research and training activities and provides services for private enterprises, encouraging the transfer of the results of nuclear technology research to the production system, including the education and training of specialists in nuclear technology.

The prerequisite for the management of the reactor is the satisfaction of all stakeholders requirements, among which safety constraints, efficiency and effectiveness in the delivery of the services. In order to continuously improve the safety and quality of reactor management and the accomplishment of the stakeholder requirements, LENA decided to implement an Integrated Management System in accordance with International Standard ISO 9001:2008. This choice allowed to satisfy both national and international compulsory requirements (i.e. safe reactor operation and maintenance) and typical ISO 9001 requirements (as e.g. continuous improvement, users/stakeholders care and satisfaction).

In addition, through this systematic and graded approach, that led to the standardization of all processes involved in reactor operation and maintenance, all the aspects of the reactor management mentioned in the IAEA publication *The Management System for Facilities and Activities* (IAEA Safety Standards Series No. GS-R-3 [1]) were also satisfied. This publication, in facts, provides a guidance for establishing, implementing, assessing and continually improving a management system for facilities and activities that integrates safety, health, environmental, security, quality and economic elements.

1. Introduction

LENA, the Applied Nuclear Energy Laboratory (http://www.unipv-lena.it/) is an Interdepartmental Research Centre of the University of Pavia which operates, among other facilities, a 250 kW TRIGA Mark II Research Nuclear Reactor.

The reactor is at the disposal of researchers from Pavia University and of other users, both public and private, for research activities, training & education and other services. The research reactor is used for many different purposes (fundamental research, applied research and technologic transfer, education and training, public information), and addresses various types of users (schools and universities students, researchers, companies, workers, individuals).

In order to continuously improve the safety and quality of reactor management and the accomplishment of the stakeholder requirements, LENA decided to implement an Integrated Management System in accordance, among others, with International Quality Standard ISO 9001:2008.

2. Stakeholders requirements and Quality Management System (QMS) scope

In relation to management and operation of a nuclear facility, stakeholders are usually defined as groups or individuals who may participate, benefit or have a specific interest in a given issue or decision: they may be both internal (those involved in the management and operation processes) and external (those which may be affected by the potential outcome of facility management) stakeholders.
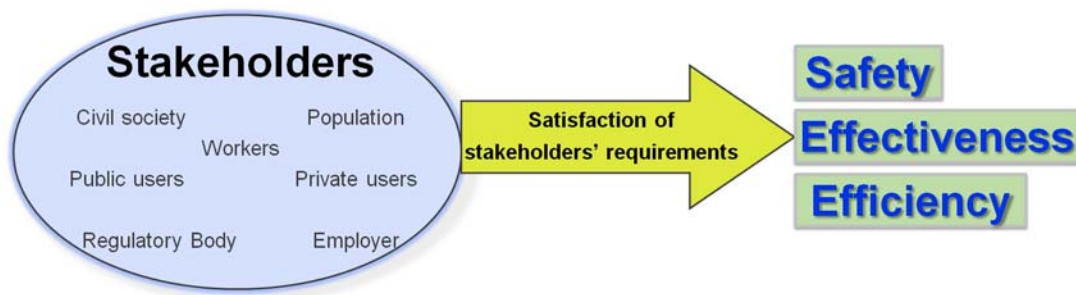
Fig 1: Stakeholders and requirements identification

In order to develop the Management System, LENA first task was to identify its stakeholders (e.g. Workers; Owner; Users; Civil Society; Population; Regulatory Body; National Governative organizations; International Organization). In fact, only by knowing exactly what the stakeholders are, it is possible to identify all requirements that must be met by the Organization and plan the necessary actions to ensure the management of the involved processes.

Each stakeholder brings several requirements, which may be legal requirements (as in the case of the Regulatory Body) or not (as in the case of users/customers).

The prerequisite for the management of a research reactor is the satisfaction of all stakeholders requirements, among which safety constraints, efficiency and effectiveness in the delivery of services as well as in reactor operation and maintenance activities.

Furthermore, it cannot be ignored that in Italy, a research reactor of nominal power above 100 kW is considered by law, under all respect, as a nuclear power plant. Thus, this category of research reactor (such it is LENA reactor) undergoes to all applicable laws and regulations in the field of nuclear safety and protection of environmental, population and workers as a nuclear power plant.

Based on the above, the reactor organization needs to manage its processes both to meet the demands arising from specific purposes and users, and to ensure the highest mandatory safety standards: this is the aim and scope of the Management System, as it is intended at LENA reactor.

The so understood Management System (MS) is therefore referred as Quality Management System (QMS): it aims to be the equivalent of what is usually called an Integrated Management System (IMS), which takes into account all quality, safety and environmental issues.

In fact, the Organization unequivocally puts nuclear safety in first place over all other needs: the management system is intended as a tool to enable the continuous development of safety culture and achieve higher safety levels.

All aspects of nuclear safety, occupational health protection and environmental protection are included in the Management System and safety culture is present and integrated in all activities, eg:

- Definition of responsibilities: both Direction and staff at all levels are aware of their responsibilities and that their actions may affect the safety of the activities connected with the reactor operation. Furthermore, Direction continuously promote safety culture though training and updating of staff, though implementation of an effective internal communication, and by encouraging personnel to suggest innovative solutions and continuous improvement.

- Definition of working practice: working practices are documented in manuals, procedures and instructions, which constitute the guide for the personnel to carry out the safety related activities. Based on a graded approach, the documentation may have different levels of detail depending on the complexity of the operations to be performed and on the competence of the staff: this approach allows the reproducibility, traceability and control of activities.

- Staff competence and training: staff adequacy is a key factor for the implementation of the Management System. The activities carried out are based on personnel skills and technical capability. The Organization manages its personnel training according with the following criteria:
  • Definition of the staff skills necessary to carry out the assigned activities that affect safety;
  • Identifying, planning, implementing and verifying training activities;

- Ensuring that staff is aware of the importance of their activities and how activities could affect safety;
- Maintaining appropriate records of training, experience and skills acquired by personnel.

- Audit and assessment: the Organization developed an internal audit system applicable to all processes of the MS, including all the safety related activities aimed to satisfy mandatory conditions for both nuclear safety and health protection.

- Top management commitment is realized through:
  - raising awareness of the Organization about the importance of satisfying safety requirements;
  - establishing the Quality Policy and raising consciousness, motivation and involvement among the staff;
  - setting safety objectives;
  - carrying out the management reviews;
  - ensuring the availability of resources;
  - systematically monitoring management system improvement and verifying the continuous requirements satisfaction.

3. Analysis of Reactor processes and implementation of the Management System

One of the first steps in the set up of the MS has been the specific and detailed analysis of all the processes involved in the management of the Organization activities, with particular regard of Reactor safety operation and maintenance aspects.

According with the typical *process approach* scheme, stakeholder's requirements, considered as input elements, play a key role in the MS since they are undoubtedly determined by the Users, but also by all the national and international legislation in the field of nuclear facility management.

For this reason, the main goal to achieve was to ensure safety reactor operation by controlling a primary process and its correlated support processes; therefore, based on requirements analysis, the processes needed for achieving the intended outputs were accurately defined.

For a better development of the system, LENA defined a Primary process as a starting point, focusing on a process which could comply with specific requirements and which could also encompass all the other processes involving safety.

All the processes are described in specific documents, such as documented procedures, work instructions, manuals, etc. which have to be considered, along with their records, as an unavoidable process control tool. A schematic diagram of what described above and how processes are linked together with safety aspects is shown in Fig 2.

A "Primary" is usually defined as a process that constitutes the core business and the main purpose of

Primary Process

Operation of the TRIGA MARK II Research Reactor for design and delivery of irradiation service

Measurements & Monitoring

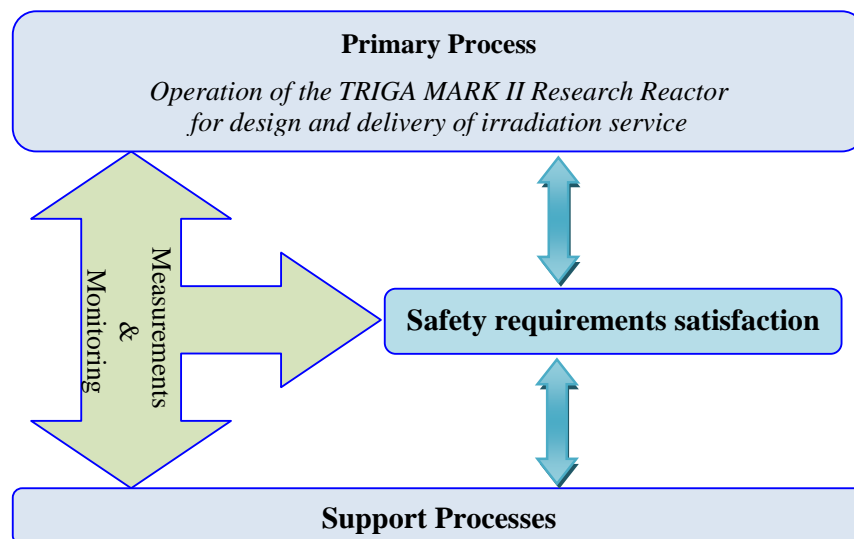**Safety requirements satisfaction**

Support Processes

Fig 2: Primary and support processes relationship

an organization. In the case of a research reactor this typically consists of research, irradiation and experiments activities.

Hence LENA defined, as its *Primary process*, the "Operation of the TRIGA MARK II Research nuclear Reactor for design and delivery of irradiation service".

This approach is a consequence of the consideration that, in order to provide irradiation and/or experiments implying reactor operation, it is unavoidable to implement also safely reactor operation and maintenance, that means, for the reactor operation, that all safety aspects have to be covered and complied.

The so understood *Primary process* includes all the stages of design, planning and delivery of samples irradiation and/or experiments at the reactor.

As mentioned above, there are many other processes related to the *Primary process*, usually mentioned as "Support processes". These processes are not part of the *Primary process*, but they are essential to enable it to take place.

In the case of a research reactor, all those activities necessary to provide irradiation and/or experiments implying reactor operation constitute the so called *Support processes*, that is all the activities connected with safely reactor operation and maintenance.

Support processes have been analyzed, developed and implemented according with the process approach, and can be grouped as showed in Tab 1.

Process
Safety related processes
Health Physics surveillance
Environmental & waste monitoring
Maintenance (Mechanical & Electrical)
Instrumentation
Purchasing
Abnormal events - Accident and/or Emergency
QMS processes
Top Management (Commitment, Authorities and Responsibilities; Top Management reviews)
Work environment
Control of documents
Control of records
Resource management
Internal and external communication
Internal Audit
Control of Non Conforming
Corrective Action and Preventive Action
Processes Monitoring & Measurements

Tab 1: LENA *support processes* definition (Safety related and typical Quality processes)

In order to accomplish the implementation of the MS with a rational logic and in the scheduled times, the entire Management System was designed in accordance with the internal document *General Plan for Design and Development of the Quality Management System*, which defines a specific and precise strategy for the system set up. Main steps of system implementation were: Project Team identification; budget and schedule development; processes analysis; Quality Policy definition; inventory of existing documentation; system implementation (development of documents and their application); system assessment (Internal audit); third party certification.

After its completion, LENA QMS was certified as ISO 9001:2008 compliant on October 2010. An independent Certification Body performed the assessment (external Audit) and released the certification, after about two years of work carried on by the Project Team and all the Organization's staff.

4. Conclusions

The global research reactor (RR) community includes a highly diverse number of facility designs, organizational structures and technical missions. RR power levels range from tens or even over 100 MW down to zero power, critical facilities. Some RR operating organizations are large and comprised of teams of operators, maintenance technicians, safety and radiation control officers, managers, licensing and other support staff. Other reactors are successfully operated by a relatively small team of around ten permanent staff or less. Similarly, technical missions can be diverse and complex or more limited including a limited number of very specialized applications.

Producing guidance on the development and implementation of an integrated management system requires examples from the ends of the aforementioned spectrum. Integrated management systems for larger RR organizations will have similarities with systems developed for nuclear power plants with comparable organizational structures. Significant differences will relate to a more diverse number of costumers managed by an organization operating a large, multipurpose RR. Here, customers range from technical industries, educational institutions, national and international research organizations, water and agricultural management programmes, and even a national nuclear energy programme. Smaller research reactor organizations also often have a diverse community of customers; but execute a more straightforward utilization programme managed by a much smaller team.

In the latter case above, an integrated management system developed for a smaller RR operating organization applies a graded approach to ensure all requirements are fully satisfied, but with significantly fewer staff. In this case, responsibilities are distributed among available staff with a single person filling the role of several different positions. The system, and organization designed to use it, are established to ensure independence; high standards of safety, quality and reliability; and continuous improvement. Since many research reactors are managed by smaller organizations, finding an example of the development and implementation of a management system for this case was key to producing comprehensive and effective guidance for the RR community.

The specific experience here described, constitutes an example of a Management System implementation in a typical medium-size research reactor according with legal and users requirements.

Among other, some aspects, typically related with such kind of Organizations, have been considered during the MS implementation, e.g.:
- Limited staff number
- Multiple responsibilities for the same unit of personnel
- Need for multi-disciplinary skills
- Capability to develop the correct graded approach in each specific situation
- Capability to perform processes measurements and assessment
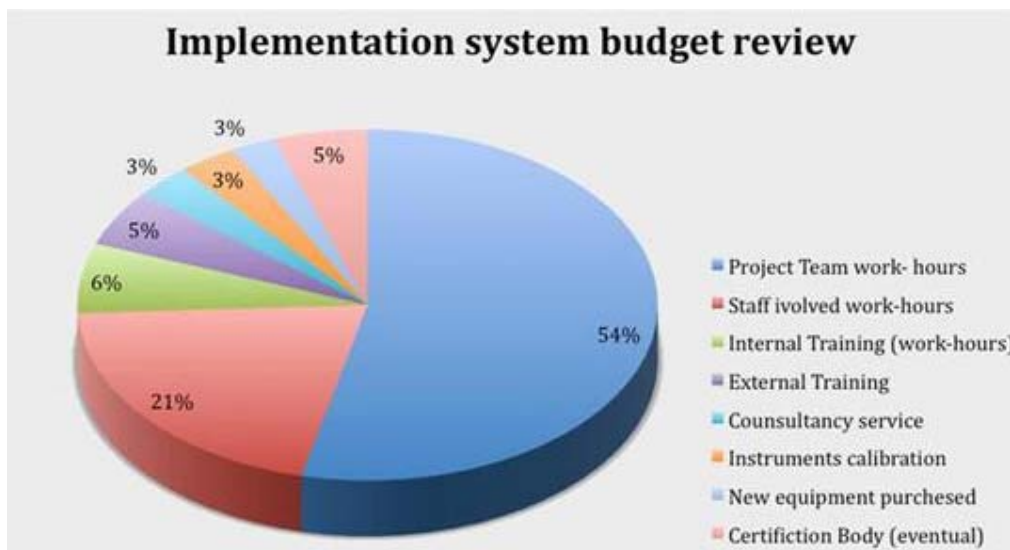- Limited financial resources

Fig 3: costs afforded by LENA to implement the system (in percentage of total outcome)

In Fig 3 is given the graph that summarizes costs afforded by LENA to implement the system. The aspects involved are here parameterized and represented as a percentage of the total outcome for the system implementation.

As for these types of Organizations, costs can be a tight issue concerning the decision to implement a management system , it is interesting to notice how most of the efforts, both in terms of cost and of time, can be ascribed to employees' involvement. That means that, when the Organization holds internally the necessary skills, the implementation process is more a matter of time and staff commitment than financial effort (for example in new equipment acquisitions or consultancy services).

During the implementation process, LENA indentified, for each process involved, a set of *Performance indicators*, i.e. parameters that provide a measurement of processes efficiency and effectiveness. *Performance indicators* are an unavoidable tool to manage, asses and improve the MS. In particular, it is interesting to observe values obtained for indicators related to LENA reactor operation process (Tab 2) and how those indicators match with ones indicated in the publication *Optimization of Research Reactor Availability and Reliability: Recommended Practices* (IAEA Nuclear Energy Series No. NP-T-5.4). This indicators choice aims to provide evidence of reactor operational and safety related processes performance, together with reactor efficiency, reliability and availability.

Process	Indicator	Expected value	Calculated value
Reactor operation	*Reactor availability[1]*	*70 – 80%*	*77%*
	Reliability[1]	*70 – 80%*	*89%*
	Reactor Utilization[1]	*>50%*	*52%*

Tab 2: Performance indicators values related to reactor operation at LENA

In conclusion, the choice of implementing an Integrated Management System in accordance with International Standard ISO 9001:2008 allowed to satisfy both national and international compulsory requirements (i.e. safe reactor operation and maintenance) and typical ISO 9001 requirements (as e.g. continuous improvement, users/stakeholders care and satisfaction).

In addition, through this systematic and graded approach, that led to the standardization of all processes involved in reactor operation and maintenance, all the aspects of the reactor management mentioned in the IAEA publication *The Management System for Facilities and Activities* (IAEA Safety Standards Series No. GS-R-3 [1]) were also satisfied. This publication, in facts, provides a guidance for establishing, implementing, assessing and continually improving a management system for facilities and activities that integrates safety, health, environmental, security, quality and economic elements.

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[2] *Application of the Management System for Facilities and Activities* - IAEA Safety Guide No.GS-G-3.1

[3] *Optimization of Research Reactor Availability and Reliability: Recommended Practices* - IAEA Nuclear Energy Series No. NP-T-5.4

[4] EN ISO 9001:2008 *Quality management systems - Requirements*

[5] EN ISO 9000:2005 *Quality management systems - Fundamentals and vocabulary*

IAEA'S TECHNICAL ADVISORY SUPPORT
OF VINČA'S SNF REPATRIATION
IN THE FINAL PREPARATORY STAGE OF THE SHIPMENT

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ABSTRACT

The IAEA has been assisting Serbia since 2001 to reduce radiological hazards at the Vinča Institute due to extended shutdown of the 6.5 MW heavy water RA research reactor and its spent nuclear fuel (SNF) that were inadequately stored in water pools within the reactor building. In 2006, under the umbrella of the Russian Research Reactor Fuel Return (RRRFR) programme, a contract was signed with a Russian consortium for safe removal and return of the RA reactor SNF to the Russian Federation, the fuel's country of origin. Given the large number of SNF elements (8030 in total) and taking into account the anticipated poor condition of the fuel, it was necessary to repackage all of the fuel to ensure safe transport. During shipment preparations, the IAEA offered two types of support: (1) safety and readiness reviews that were requested by the Serbian Regulatory Commission; (2) technical advisory support upon demand of the Operation Organisation (Vinča Institute and later Public Company "Nuclear Facilities of Serbia") in preparation of licensing and technical documentation, to advice on operational safety and technical aspects of fuel repackaging and preparation for transport. This paper focuses on the technical advisory support provided by the IAEA: most of the discussion focuses on the final preparatory activities carried out prior to transport; there is less focus on the preparatory work accomplished before 2009. Finally, the paper makes some concluding remarks about lessons learned through the project.

1   Introduction

The main nuclear and radiation safety issues at the Vinča Institute of Nuclear Sciences, Belgrade, were: (1) the 6.5 MW heavy water RA research reactor that had been in the stages of an extended shutdown since 1984; (2) spent nuclear fuel (SNF) from the RA operation inadequately stored in the water pools within the reactor building; and (3) insufficient storage facilities for the low and intermediate radioactive waste (waste storage hangars). Although the radiological hazard issues were reported in 1996 by the Vinča Institute the first TC project was launched in 2001 (shortly after the political situation settled down) following three further specific projects. The strategic objectives of these projects were firstly to survey the site, identify the real conditions of the SNF and waste storage facilities, then letter develop a nuclear characterization of the site. One of the main objectives was to regain control of the stored SNF, stabilize the conditions (prevent the escalation of further degradation as much as possible and achieve a long term safe and stable state) and find an ultimate solution. Firstly two possible solutions were envisaged to manage the SNF problem: (a) ship the SNF elements abroad, or (b) create the conditions for the long-term storage in the country. Due to the announcement of the Global Threat Reduction Initiative (GTRI) announced in May 2004 and then the Bratislava Agreement[1] the second option of the long-term storage in the country was dismissed and from 2005 the main goal was to ship the SNF elements back to the country of origin. Now the projects are

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[1] Since 1999, and continuing to the present, the US, the RF, and the IAEA have been working on a programme to return to Russia stockpiles of Soviet- or Russian-supplied HUE fuel currently stored at foreign research reactors. On 27th of May, 2004, the US and RF governments signed an agreement in Bratislava (Bratislava Agreement) on the repatriation of research reactor fuel of Russian origin.

managed in the frame of a coordinated and interrelated programme under the title of VIND (Vinča Institute Nuclear Decommissioning) Programme a major part of which was the SNF repatriation project, which accomplished successfully in 2010.

2 SNF repatriation Project

The real risk mitigation measures started in 2001 immediately the site was stabilized. The identified potential nuclear and radiological safety phenomenon was raised at the governmental level. As a consequence of this the Serbian Government decided to shut down the RA reactor permanently and to participate in the international nuclear non-proliferation efforts associated with reducing the amount of HEU in international commerce[2]. The decision resulted in broad international cooperation and the effective provision of service assistance, as well as donations.

2.1 Legal framework

Upon the invitation of the IAEA, in May 2005 an international consultancy meeting was held in the Vinča Institute with the participation of a few invited international enterprises. The main goal of the meeting was to draft the outlines of an international bid for the removal and transportation technology of the seriously corroded and leaking SNF assemblies being stored in the storage pool accommodated by the reactor building. At the conclusion of this meeting an international tender was issued by the IAEA in the summer of 2005. Following the selection procedure, an RF consortium (Sosny-Mayak-Tenex: hereinafter referred to as Sosny after the leading company of the consortium) was selected and an international tripartite contract between IAEA, Sosny and the Vinča Institute was signed in September 2006 for the safe removal of spent nuclear fuel (SNF) from the Vinča RA Research Reactor in Serbia and return to the RF.

Two further frame agreements were signed in 2009: (1) in June 2009, at the governmental level, a "Government-to-Government Agreement between the Government of the RF and the Serbian Government concerning cooperation for the transfer of research reactor irradiated nuclear fuel to the Russian Federation"; (2) in September, at the enterprise level, the "Foreign Trade Contract" between Federal State Unitary Enterprise "Federal Centre of Nuclear and Radiation Safety" (FSUE "FCNRS") and the PC NFS for the importing irradiated nuclear fuel of the RA reactor into the RF, its reprocessing and final disposal of radioactive waste material in the RF.

2.2 Technical approach of repackaging and removing

Within the frame of the tripartite contract, in May 2007, R&D Co. Sosny completed a report titled 'Design of SNF removal to FSUE PA «MAYAK»' detailing the conceptual approach to RA SNF repackaging [2]. The conceptual approach outlined the development of related documentation, tools, facility modifications, regulatory approvals, international agreements and licenses.

The concerned SNF inventory is comprised of 8030 TVR-S fuel elements (Fig. 1). Approximately 17% of those contained highly enriched uranium. The fuel elements (by the time of repackaging) were stored in 30 aluminium barrels in the storage pool containing 4929 fuel elements; and in 297 stainless steel channels in RA reactor vessel containing 3101 fuel elements. Given the large number of SNF elements (8030) and the anticipated poor condition of the fuel [3], it was already decided in the contractual phase that it would be necessary to repackage all of the fuel to ensure safe handling and transport.

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[2] This agreement paved the way so that in 2002, the Republic of Serbia could be the first IAEA Member State to repatriate fresh HEU fuel to the RF [1].
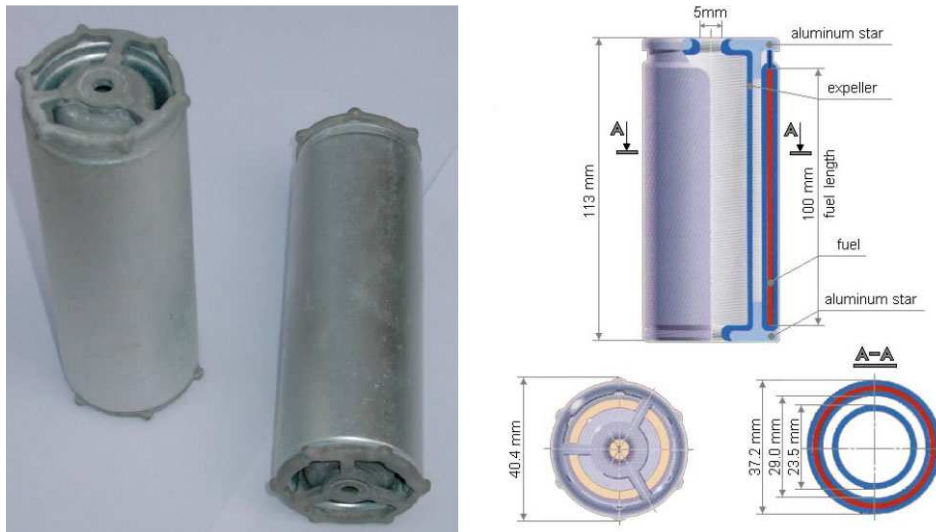
Fig 1.   TVR-S fuel element

The repackaging technology was defined by two factors: (1) the SNF type and its storage solution; (2) the transport package(s) that would be used for transportation. The fuel and its storage mode were given and the possibility to select transport packages was also determined (license issue, container availability). Hence, in view of all circumstances (number of fuel to be shipped and containers available) 16 Skoda VPVR/M and 16 TUK-19 casks were selected. Two types of canisters (TC1 for TUK-19 and TC2 for WPVR/M) were developed according to the selected transport casks. The 3D-drawings of the canisters, as well as the new basket together with their placement in the containers are presented in Fig. 2.
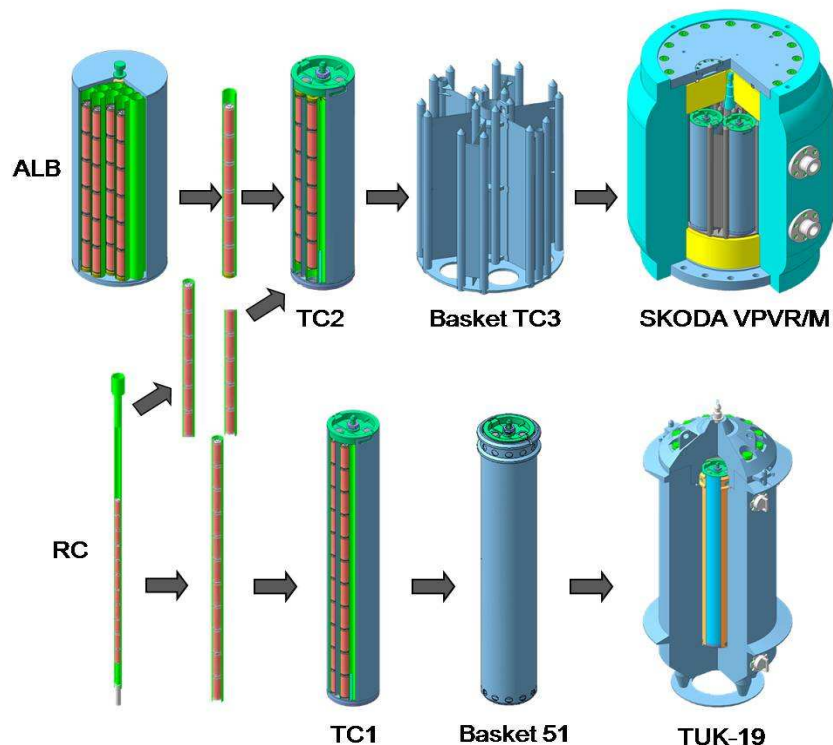

Fig 2.   3D-drawings of the canisters and basket for VPVR/M[3]

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[3] Source: A.V. Smirnov's presentation given in the 3rd annual Regional Workshop on RRRFR Programme Lessons Learned May 25-28, 2010, Poiana-Brasov, Romania.

2.3   Main milestones passed during the project implementation

The obligatory documents, manufacture of custom fuel repackaging tools and equipment, then the site preparation were made according to the conceptual design and technical specification defined by Sosny. The main milestones passed during the project implementation were:

- Development the Conceptual and Technical Designs as well as work plan (2007-'08);
- More than 200 custom fuel repackaging tools and equipment designed and fabricated by SOSNY were delivered to PC NFS throughout 2009.
- FSAR review conducted by IAEA was successfully completed in Jul'09. This milestone led to FSAR approval and licensing by the Serbian Regulatory Authority (SRA) in Nov'09 for SNF repackaging and transport.[4]
- All equipment was assembled, tested and its operation demonstrated and approved, as well as operating personnel trained as of Nov'09.
- A custom water chemistry control system (WCCS) was installed and fully operational as of Nov'09. The WCCS minimizes radiation dose rates in working areas of SNF storage pool primarily by controlling specific activity of Cs-137.[5]
- The SNF repackaging from December 2009 to May 2010.
- Final decision on shipment route in January 2010;
- Contract 2006-1312 amendment in June 2010 that was necessary mostly due to the shipment route and mode changes;
- Follow up safety and readiness assessment in July 2010 prior to start of container loading;
- Skoda SKODA VPVR/M cask loading was done in two steps (accordingly as the empty containers became available): in August-September 2010 twelve containers were loaded and then a further four in November;
- All trans-boundary and import permissions to the RF were obtained by the beginning of November 2010;
- The 16 TUK-19 casks were loaded during October-November 2010;
- The ISO transport containers (14 with casks and 1 with handling tools) were in road-ready status by November 12;
- SNF Convoy departed Vinča site on November 19, 2010;
- The shipment arrived to Mayak on December 22, 2010.

2.4   IAEA's support in general

Within the framework of IAEA's Technical Cooperation projects RER/3/006 and SRB/4/002, different activities were provided to support the safe removal of Vinča RA Research Reactor's spent fuel. The backing activity of the Agency, on the basis of support initiation can be divided into three groups:

(1) Support provided upon demand of the Serbian Regulatory Commission (SRC).
(2) Support provided upon demand of the operation organisation and/or contractual parties; and
(3) Traditional support which is typically an integral part of any TC project launched by the Agency (outlining a project, call for bids, contracting, appraisal of deliverables, etc.).

The support activates belonging to the second and third group are jointly discussed in this paper.

3   IAEA's Technical Advisory and Expert Support

The Public Company Nuclear Facilities of Serbia (PC NFS) was created by the Serbian Government in 2009. Since its inception, PC NFS has assumed control and responsibility for all

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[4] Special thanks to the Slovenian Nuclear Safety Administration for their in-kind contribution of expert support to the SRA, which contributed significantly to making this achievement possible.
[5] Special thanks to an in-kind contribution by the USDOE-NNSA for the custom design and technical assistance.

of Serbia's nuclear assets, including the safe removal of SNF from the Vinča RA Research Reactor. As a new company faced immediately with a big challenge the PC NFS asked and greatly welcomed any effective and authentic backing that helped to fulfil its first job safely and in a timely manner. Hence the Agency, who even encouraged the Serbian Government to create a legal entity exclusively responsible for the waste management in Serbia to aid the VIND programme, provided technical advisory and expert support. This support was not only provided at the demand of the operation organization but frequently exceeded the "standard duties" and traditional supporting role within the frame of a TC-project.

The technical advisory and expert support of the Agency encompassed mainly the following five activity fields: (1) traditional technical support, (2) documentation preparatory support, (3) trans-boundary licensing support, (4) on-site technical reviews and (5) advisory support to implement equipment.

The (2) document preparatory support and (3) trans-boundary licensing support have some overlapping with the support of safety and readiness reviews, thus they are discussed in a separate paper [4].

3.1   Traditional technical support

As it was pointed out, not only the quantity of the usual technical TC project support was increased due to the complex and long lasting project but the kind of the support crossed also frequently the usual framework. Considerable support activates resulting mostly turning point in the project history were:

- **Technology selection.** The Agency played an initiative and decisive role in 2005 to survey the site conditions, as well as infrastructure available in order that to define the technology approach (first repackaging then transfer) for SNF repatriation and to draft the technical requirements of an international bid that resulted in the tripartite contract for developing repackaging and shipment technology and shipment of the SNF to RF being signed in September, 2006.
- **Legal entity for the VIND Programme.** During the project termination it became obvious that the VIND project objectives didn't fit the original organisation structure of the Institute as the VIND Project represented a second organisation sitting on top of the traditional structure. These dual structures generated personal conflicts and jeopardised the project transparency. Recognising these risk factors, in 2009, following the acceptance of "Law on Protection Against Ionizing Radiation" by the Parliament, the Serbian Government decided to provide a legal entity to the VIND Project. The Agency encouraged this approach.
- **Traditional technical support.** This covers biddings, issuing purchase orders, appraisal of deliverables, participation in panel reviews and factory and site acceptance tests. These technical issues were covered by four technical officers. To illustrate the project complexity, in the last one and half years 14 purchase orders were issued and 117 deliverables were technically evaluated that represented almost 90% of the project budget.

3.2   Documentation preparatory support

Upon demand of the PC NFS and sometimes contractors as well, the Agency ensured an overall consultative advisory and review support to develop the documentation. The support concerned PC NFS's Mandatory Documents that were required to be elaborated and kept up-to-date by the SRA and/or PC NFS's requirements are discussed in the separate paper referred as [4].

Regarding the technical documents accompanying the shipment the Agency provided overall assistance. Using the documentation of the earliest made shipments a complete set of templates were given to the PC NFS to elaborate fuel and container passports, radiation survey protocols, leakage test certificates, etc. Specimens of different protocols, emergency response

plans, third party liability insurance policy, etc. were also handed over to support the preparation of technical and accompanying documentation. The final set of documentation was also revised by Agency experts.

3.3 Trans-boundary licensing support

While the onsite work meant a unique technical challenge for the operators, the obtainment of all needed authorisation and permits provided a puzzle for the legal entity intending to ship the SNF by the means of different transport modes from a non EU country through the EU to the RF. Although the transit countries were EU members, the procedures were only partly harmonised, and, of course, the initial country's and target country's regulations and procedural rules were completely different. Therefore, on this subject, the Agency had to provide long lasting complex advisory and coordination support, and sometimes had to act as a mediator between authority representatives and applicants. This support issue is also discussed in the separate paper referred as [4].

3.4 On-site technical reviews

The on-site technical review activity of the Agency encompassed the entire on-site activities. It took the form of a periodic review and advisory support to review the on-going activities, which covered staff training, equipment commissioning, test operation of newly manufactured or already used accessories, devices and tools, as well as implementation of work procedures (e.g. package maintenance before loading, cask loading and drying, waste management, etc.). Within a facility walk down the acting Agency's technical officer(s) revised the equipment conformity, work performed in terms with the requirement and the written procedures. Repeatedly revised and emphasised issues were:

- work performance to strengthen planned, supervised and documented work culture;
- safety provisions including visible management commitment to safety, understanding the concepts of safety issues, trust between management and front-line staff;
- facility walk downs served also to obtain commonly with the operators lessons learned on the "overview" of a technical malfunctions that occurred (if any) between two visiting periods. The goal in each case was to clearly understand the causal relations, evaluate how the problem was fixed (corrected the work procedure) and the repaired equipment was tested, and documented the event.

The observed and revised issues were discussed in each case with the PC NFS's management in closing meetings following the facility walk downs.

3.5 Advisory support to implement equipment

These were very specific subject-oriented interactions, usually to mitigate risk forecasted in advance or to manage critical path scenarios that came up unexpectedly. Two interactions were made during the loading period:

- **TUK-19 Transfer Flask.** This device was designed by Sosny, manufactured in Romania and operated for the first time by IFIN-HH in the Romanian HEU SNF repatriation project. This equipment was used also for TUK-19 cask loading at the Vinča facility. Although the majority of the "teething problems" of the device were resolved during the IFIN-HH operation, some deficiencies were reported on the Lessons Learned meeting that took place in Poina-Brasov in May 2010. Being the designer and the main contractor is the same company (Sosony), the PC NFS was strongly supported by the Agency and advised that the technical requirements be clearly specified in the leasing agreement of the transfer flask and also prescribed that the designer (Sosny) should provide written instructions on the safe usage of the transfer flask, and demonstrate the operability of the flask (commissioning test), as well as train the user's operators for the proper operation of the cask. Due to this prudent agreement and strong Agency review the designer made the needed corrective

actions and the upgraded transfer flask operated safely and reliably without the already identified "typical" failure during the 16 TUK-19 casks loading.[6]

- **TUK-19 cask hot air- and vacuum drying.** A delay was announced in September 2010 for shipping the empty TUK-19 casks to the Vinča facility. The challenge put before the stakeholders of the Vinča shipment was to determine how to absorb this delay in the project schedule so as to maintain the original shipping schedule from Vinča. The originally scheduled 20 days for TUK-19 loading no longer had been kept. The IAEA developed and proposed an "in-parallel cask loading, drying and vacuuming plan" for TUK-19 transport casks, which reduced the previous 20-day in-series plan to 5 days, and initiated a technical meeting for optimizing the loading of fuel transport casks with the participation of all Vinča stakeholders. The proposed in-parallel approach was studied by all technical stakeholders for one week in advance of the meeting and was discussed in extensive detail during the meeting. It was agreed by all participants that the proposed in-parallel approach ensured the optimum and fastest safe procedure for loading and drying/vacuuming the transport casks. It was also estimated that the theoretical net time demand (5 days) to load and dry the TUK-19 casks allowed an additional 3 days in the schedule to absorb minor delays, unexpected technical events, and transfer of loaded casks to the final ISO shipping containers. The agreed in-parallel TUK-19 cask hot air- and vacuum drying approach was implemented, and the TUK-19 casks were loaded and dried by the implemented in-parallel approach within 10 days working in one shift/day only, as the time frame was not so tight at the end due to further changes.

3.6 Means of support

The SNF repatriation from the Vinča facility was the largest and most complex project in the Agency history. To comply with the facility demands, as well as ensure safe and timely project termination the Agency facilitated several means of support. In terms of technical advisory support following means of support were the determinatives:

- **Managerial and coordination assistance.** Contributed Agency's organisation units were TC and NSNS. A Special Programme Manager (SPM) was hired based on an extensive technical and project management background. The SPM managed the in-house coordination within the Agency, served as the main technical coordinator between the TC and technical departments; maintained meetings with donors, international organisations, project stakeholders; initiated actions for implementation, including the recruitment of experts and procurement of equipment, in consultation with technical divisions and counterparts.

- **Technical backing.** Contributed Agency's organisation units were RRS, MTPS, OLA, and SG. This activity included contracting, procurement support, participation on panel reviews, and technical appraisals of deliverables. The SNF repatriation project had four TOs, plus two technical experts assigned to the Project Management Unit (PMU) at the Vinča site.

- **Meeting, workshops organised by IAEA.** In 2010 for example, eleven meetings and workshops were organised to facilitate the project termination safely and in a timely manner. The meeting days were used to organise goal oriented bilateral, multilateral ad-hoc meetings as well.

- **Missions.** Contributed Agency's organisation units were NSNI and RRS plus involvement of external experts. Three missions were conducted in the last two years, the subject of which were mostly safety and readiness assessment, but these missions also supported the backing activity of the Agency and gave the opportunity to have an on-site walk down.

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[6] Some problem happened with its grapple: it got stuck once while getting out of the cask, due to the small gap between the opening of the cask and grapple pincers. This can serve a lesson learned to the designer (terminal fittings with assuming harsh operation circumstances).

- **Document preparation and reviews.** Contributed Agency's organisation units were RRS, NSNI and NSNS. This included in-house meetings, expert discussions to prepare in advance or review documents, as well as outline templates and draw up requirements.
- **Stand-by support via e-mails and phone.** Contributed Agency's organisation units were RRS and NSNI. This form included ad-hoc discussions clarifying requirement, prompt advisory support in response to daily questions, involving competent authorities and experts (if needed), and intermediation between stakeholders.
- **On site review visits.** Contributed Agency's organisation units were RRS, NSNI and NSNS. A few subject-oriented trips to the facility were organised, but mostly the site walking downs were linked to the meetings announced in advance.

4   Conclusion

The technical advisory and expert support provided by the Agency to the operating organisation and stakeholders formed a wide-ranging professional backing, which was provided almost on a standby basis upon the stakeholders' demand. The continued advisory and expert support significantly contributed to the smooth project termination, and helped ensure that all jobs were done safely and in a timely manner, and that in the end the often called "never ending project" was successfully accomplished. In summarizing the experiences gained from the technical advisory and expert support four important conclusions can be made:

- The technical advisory and expert support strengthened the policy of the safety awareness among the front line operators: all activates were carried out with trained staff, in planned, controlled and documented way that resulted that no safety, security or radiation relevant issues as well as incidents, accidents or any other remarkable event were reported.
- While the technical support strengthened the nuclear safety culture norms among the staff members of the operating organisation the IAEA's backing fostered the self-confidence mainly in the PC NFS's management and operators.
- The technical support was often coupled with managerial and coordination provision which resulted that the Agency embodied a team-maker role (gathered the stakeholders into a real working team).
- While the IAEA fulfilled its advisory and coordination assistance, the Agency unintentionally played the role of a mediator amongst the stakeholders, which consequentially yielded additional Agency support: a kind of "charity mission" to resolve conflicts between core actors and strengthen trust issues amongst the contractual parties.

Finally, as lesson to remember – a truth of an old pearl has proven again: well-organised teamwork brings success.

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REPROCESSING OF LEU U-MO DISPERSION AND MONOLITHIC FUELS

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ABSTRACT

For conversion of high-performance research reactors from high-enriched uranium (HEU) to low-enriched uranium (LEU) fuel, a fuel material with a higher density than uranium aluminide is required. Development studies are underway to develop U-Mo dispersion and monolithic fuels for conversion of several high-performance reactors. For dispersion fuels, development is narrowing down to a composition of U-7Mo dispersed in an aluminium matrix containing ~5% silicon. For monolithic fuels to be used in high performance research reactors in the United States, a zirconium-bonded U-10Mo foil appears to be the fuel of choice. For conversion to be realized a back-end disposition path is required for both fuels; one disposition pathway is reprocessing.

Argonne National Laboratory is developing a pyroprocess for reprocessing spent monolithic fuel. Pyroprocessing was chosen over conventional aqueous solvent extraction due to the necessity of adding fluoride to the fuel-dissolution solution in order to dissolve the zirconium bonding layer on the U-Mo fuel. The proposed flowsheet and development activities will be described.

A literature survey points to the ability to reprocess U-Mo dispersion fuels by an aqueous process, but due to several special characteristics of the fuel, the solvent-extraction flowsheets will be a departure from that normally used for the reprocessing of power reactor fuel. Special concerns that must be addressed in reprocessing these fuels are, for example, the low solubilities of uranyl molybdate, molybdic acid, and silicic acid in nitric acid solutions. This paper will address these concerns and development activities required to overcome them.

1. Introduction

The Fuel Development program of the National Nuclear Security Agency (NNSA) Global Threat Reduction Initiative (GTRI) is currently engaged in the development and testing of a novel nuclear fuel that would enable conversion of U.S. high-performance research reactors (HPRRs) to low enriched uranium (LEU) fuel.[1] These reactors include the High Flux Isotope Reactor (HFIR) at the Oak Ridge National Laboratory (ORNL), the Advanced Test Reactor (ATR) at the Idaho National Laboratory (INL), the National Bureau of Standards Reactor (NBSR) at the National Institute of Standards and Technology (NIST), the Missouri University Research Reactor (MURR) at the University of Missouri–Columbia, and the MIT Reactor-II (MITR-II) at the Massachusetts Institute of Technology (MIT). The most promising fuel type currently under investigation is a layered, monolithic design consisting of a uranium-molybdenum alloy foil with 10 weight percent molybdenum (this fuel is referred to as U-10Mo), enclosed within a 6061 aluminium cladding.[2] The uranium-molybdenum foil is separated from the aluminium cladding by a diffusion/bonding interlayer of zirconium. This design provides a fuel-meat density of approximately 15.3 grams of uranium per cubic centimetre, making it particularly promising as a LEU alternative for HEU in research and test reactors.

Studies are also going on throughout the world to develop U-Mo dispersion fuels for (1) converting current HPRRs that are fuelled by high-enriched uranium (HEU) to low-enriched uranium (LEU), (2) converting current LEU-silicide-fuelled reactors to a fuel that promises to be more easily reprocessed, and (3) designing of new LEU-fuelled HPRRs.

Once the fuel is developed and qualified, the key to using a fuel in a reactor is having a disposition path for the spent fuel. This paper discusses potential reprocessing paths for both the monolithic and Al-dispersion U-Mo fuels.

2.   Disposition of U-Mo Monolithic Fuels

The baseline monolithic fuel-plate design consists of U-10Mo core with a thickness ranging from 0.01 – 0.05 cm. The fuel meat is clad in Al-6061 alloy with a thickness ranging from 0.04 – 0.24 cm (sum of both sides). The cladding and the U-10Mo foil are separated by a zirconium layer that is approximately 25 µm thick.

Annually, 1.8 metric tons of LEU will be used to run the five U.S. HPRRs.[3] The projected number of element used in the five reactors and mass of LEU per element are shown in Table 1. Currently, there are three disposition options for spent U-10Mo fuel: (1) long-term storage awaiting a US high-level waste (HLW) repository, (2) aqueous reprocessing at the Savannah River Plant or elsewhere in the world using a tributyl-phosphate based solvent extraction process, and (3) pyroprocessing of the spent fuel either at the Idaho National Laboratory's Fuel Conditioning Facility, another DOE facility, or in a brand-new facility.

Reactor	Elements/yr	g-LEU/element	Spent LEU kg/yr
MITR	7	4,257	30
MURR	19	7,287	138
NBSR	28	1,924	54
ATR	82	6,942	569
HFIR	8	127,500	1020
Total			**1811**

Tab 1: Projected Number of elements used per year and LEU per element for US HPRRs[3]

2.1   Long-Term Storage with Eventual HLW-Repository Disposal

With the loss of a Yucca Mountain Repository, the amount of time required for long-term storage and the availability of a storage location is unclear. Currently, many state governments have forbidden storage of radioactive waste without a disposal option.

2.2   Aqueous Processing Option

The presence of a Zr-bonding layer for the monolithic U-Mo fuel puts a difficulty on aqueous processing of this fuel. The interface between the U-Mo fuel and the Zr layer is likely to contain U-Zr compounds that can react explosively when dissolved in nitric acid alone.[4-7] It is, therefore, common practice to dissolve fuels that contain U-Zr intermetallic compounds in nitric acid containing significant quantities of hydrofluoric acid. Process flowsheets for dissolving and processing this fuel using fluoride have been developed, but not demonstrated.

The Savannah River Site (SRS) has the capability to reprocess this fuel if the free fluoride (HF + F$^-$) concentration is kept low enough to make corrosion of the dissolver and other stainless-steel equipment manageable. At the free-fluoride concentration required to dissolve this fuel, the corrosion rate is estimated to be in the range of 0.78 mm/year. Determination of an "acceptable" rate of vessel corrosion is an empirical task that depends upon the construction of the vessel, the expected time of contact, and the desired service life of the system, among other factors. Such a determination requires an engineering assessment, although rates below 1.3 mm/ year are considered "low," "mild," or "adequate" in similar contexts.[8,9].

Even if the fuel could be processed at SRS, the current plan is to shutdown H Canyon by 2019, making reprocessing of the fuel at that site unlikely. Therefore, it does not appear that aqueous reprocessing of this fuel at SRS should be considered a viable option.

2.3   Pyrochemical Reprocessing Option.

As part of the GTRI Fuel Fabrication Capability scrap-recycle effort, Argonne has developed a pyrochemical option. It was chosen as the best option for scrap recycle because of a number of factors when compared to aqueous processing:

- Limited chemical reagents
- Less process waste
- Reduced process footprint

Pyrochemical processing is also the preferred option for the treatment of spent HPRR fuel. Pyroprocessing uses molten salt electrochemical methods to separate and recycle uranium from the fission products contained in used fuel. The technology was developed for treating fuel discharged from advanced liquid-metal-cooled reactors, and its feasibility was demonstrated by conditioning fuel discharged from the Experimental Breeder Reactor II (EBR II), which is located in Idaho. The process used in Idaho was designed to treat U – 10 wt% zirconium (U-10Zr) fuel, but it shares many of the same key separation operations that would be required to recover uranium metal from HPRR fuel. Feasibility lab- and engineering-scale experiments conducted at Argonne National Laboratory have demonstrated recovery of uranium from U-10Mo fuel scrap to establish recovery efficiencies and optimize operating parameters for the electrorefining process. In fact, a higher degree of uranium recovery is expected from the U-10Mo fuel because of the difference in electrochemical properties between molybdenum and zirconium.

The flowsheet describing the pyrochemical process to treat used HPRR fuel is shown in Figure 1. The process is composed of seven principal operations that begin with a chemical decladding step using aqueous sodium hydroxide to remove the aluminium cladding. Once the fuel is declad, the material is mechanically chopped and placed into fuel baskets. The fuel baskets are transferred to an electrorefiner that electrolytically recovers uranium metal from the spent fuel. Active-metal fission products and transuranic elements would partition to the molten salt, and the noble-metal fission products and the zirconium and molybdenum components of the fuel would remain in the fuel baskets. The uranium-metal product with residual salt is transferred from the electrorefiner to the cathode processor to consolidate the metal and remove any adhering salt. The resulting pure uranium-metal ingot can be melted with HEU to reach 19.8% ^{235}U enrichment for recycle in fuel fabrication. Salt distillation is also performed on the noble-metal fission products, such as molybdenum and zirconium that remain within the fuel baskets, to create a stable metal waste form and allow residual salt to be recovered. The molten salt collected from the uranium consolidation and noble-metal fission product processing operations is recycled to the electrorefiner after it undergoes treatment to remove the salt-soluble fission products and a replenishment process to generate uranium-trichloride oxidant, which is required by the electrorefining process due to the presence of active metal fission products in the fuel. The waste generated from the process is composed of aqueous waste, a noble-metal waste, and an active-metal fission product waste.

The hot-cell facility envisioned to process the spent HPRR fuel would be sized to treat approximately 2 tons of fuel annually. The current facility used for the treatment of EBR II fuel at Idaho is of sufficient size to treat spent HPRR fuel and has established the ability to work with the complexities of spent fuel in a remote environment. Appropriate nuclear facility operations protocols and staff are in place to support the fuel treatment operation.   An advantage of converting the Idaho facility to process used HPRR fuel is that fuel chopping, electrorefining, cathode processing, and uranium product casting operations have been demonstrated at this facility. The electrorefining and uranium consolidation processes have been developed to an engineering-scale for a similar fuel-type (i.e., U-10Zr), but these operations would require equipment upgrades to the technology (e.g., planar electrorefiner) currently being demonstrated at Argonne for HPRR fuel treatment. In addition, new casting furnaces would need to be installed into the facility.   The system, however, has the disadvantage of not being engineered or optimized for the proposed HPRR flowsheet. For example, chemical decladding of the fuel would need to be completed outside the facility and the unclad HPRR fuel would need to be dried and transferred into the inert atmosphere

facility. Additionally, fission product drawdown, fission product waste production and oxidant production systems would also need to be incorporated into the existing facility. Substantial process engineering including removal of old equipment and system integration would be required to retrofit the facility with the equipment needed for HPPR fuel treatment. Given that many of the processing locations are fixed within the current hot cell, material handling, which is extensive for pyrochemical processes, may prove to be the largest hurdle to facility optimization. Another option would be to use hot cells at PNNL or SRNL. These are air hot cells, but pyrochemical operations could be placed in inert modules and used in these facilities. A third option is to build a new facility.
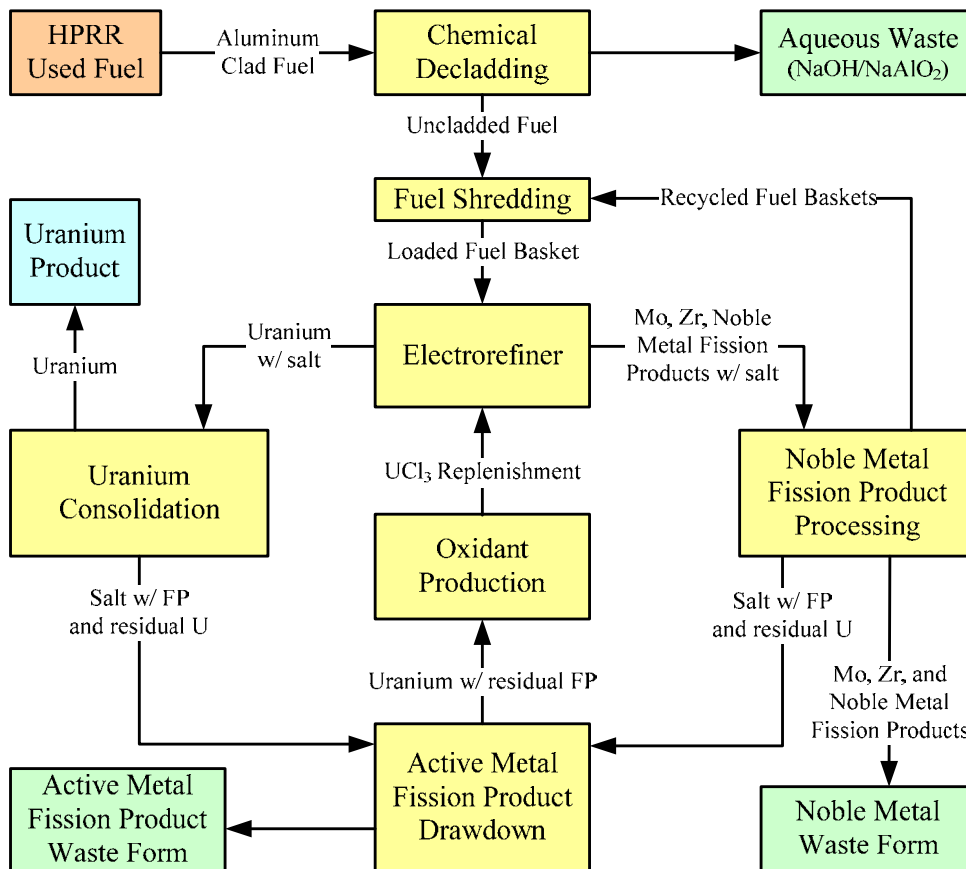

Fig 1: Conceptual Pyrochemical Process to Treat Used HPRR Fuel

3. Reprocessing of U-Mo Dispersion Fuels

In a 2005 publication, Herlet et al. described experimental work on the dissolution of both irradiated and unirradiated U-Mo dispersion fuel. They also provided a scheme to reprocess the fuel at La Hague by adding the dissolved fuel to the dissolved power-reactor feed to the PUREX process. The spent LEU U-Mo fuel solution would be dissolved to ~15 g-U/L and fed into the conventional steam at a ratio of 1 part per 12 parts ~200g-U/L of dissolved spent power reactor fuel.[10]

At the time of the Herlet et al. study, the requirement for adding Si to the aluminium matrix was not known; generally, ≥ 6 wt% Silicon is required to provide stable reactor performance in U-Mo dispersion fuels.[11,12] Therefore, in this paper, we have looked at the effects of the addition of 6% Si to aluminium fuel matrix on the reprocessing scheme provided by Herlet et al.

For this study, we have chosen the U-7Mo LEU fuel suggested for conversion of the Laue Langevin Institute (ILL) High Flux Reactor (RHF).[13] This feasibility study considered the use of 3% Si; the fuel composition in this report was modified to allow 6% Si. Table 2 provides

the composition of a 15g-U/L solution of the dissolved fuel, which includes fuel and Al/Fe/Ni cladding but ignores fission and activation products formed during reactor operation.

| Element | Concentration | |
	g/L	M
H		4
Al	1.19E+01	4.42E-01
U	1.50E+01	6.31E-02
Mo	1.13E+00	1.18E-02
Si	1.81E-01	6.45E-03
Mg	8.81E-02	3.62E-03
Ni	9.77E-02	1.66E-03
Fe	9.09E-02	1.63E-03
Mn	2.87E-02	5.23E-04
Cr	2.68E-02	5.16E-04
C	2.62E-03	2.18E-04
Ti	4.79E-03	9.99E-05
Zr	7.66E-03	8.40E-05

Tab 2: Calculated solution composition for unirradiated proposed ANL LEU U-Mo dispersion fuel at 15g-U/L (components with concentrations above 10^{-5} M.)

Results from several studies have shown that Mo is soluble under the conditions specified by Herlet et al.[10,-14-23] However, the Si added to the fuel meat is likely to hinder the proposed scheme. The major problem with silica precipitation is its chemical form; silica gel, is a non-crystalline precipitate that cannot be readily filtered and can only be effectively removed by use of a centrifuge. La Hague does not have a centrifuge, and formation of silica gel was their primary concern with reprocessing of uranium-silicide research reactor fuel.
The saturation index (SI) for silica gel was calculated at three temperatures using the thermodynamic code Geochemist's Workbench and equilibrium constants from Helgeson et al.[24] The results presented below are from a calculation that was done for a 4 M nitric acid solution containing 6.45E-3 molar dissolved silica. Calculations were also run for nitric acid concentrations ranging from 2 to 8 molar (for the same dissolved silica concentration). The results showed that the solubility of the silica gel does not vary with acidity.
The saturation index (SI) is used to estimate whether a solid phase will be present in a given mixture. If SI < 0, the solution is undersaturated and therefore stable (no solids); if SI > 0, the solution is supersaturated and the solid would be expected to precipitate.

The saturation index for the reaction $SiO_2(am) \leftrightarrow SiO_2(aq)$ is defined as follows:

$$SI = Log_{10}\left(\frac{[SiO_2(aq)]}{K_{sp,SiO_2(am)}} \right)$$

Where $[SiO_2(aq)]$ is the instantaneous ion activity product for the reaction and $K_{sp,SiO2(am)}$ is the solubility constant for the solid phase $SiO_2(am)$. Figure 2. shows the Saturation Index at three temperatures. The calculations suggest that the solution of interest is supersaturated with respect to silica gel at 25°C and 75°C, but is slightly undersaturated at 110°C. Figure 3. shows the amount of amorphous silica that is expected to precipitate as the solution is cooled from 110°C to 25°C as well as the corresponding dissolved concentration of silica.

4.   Summary and Conclusions
A pyrochemical process was selected as the most appropriate means to reprocess zirconium-bonded monolithic U-Mo fuel used in conversion of U.S. HPRRs to LEU fuel. For

reprocessing of U-Mo dispersion fuels, the silicon that must be added to the fuel meat for stability during reactor operation adds a new concern to its reprocessing by aqueous means.
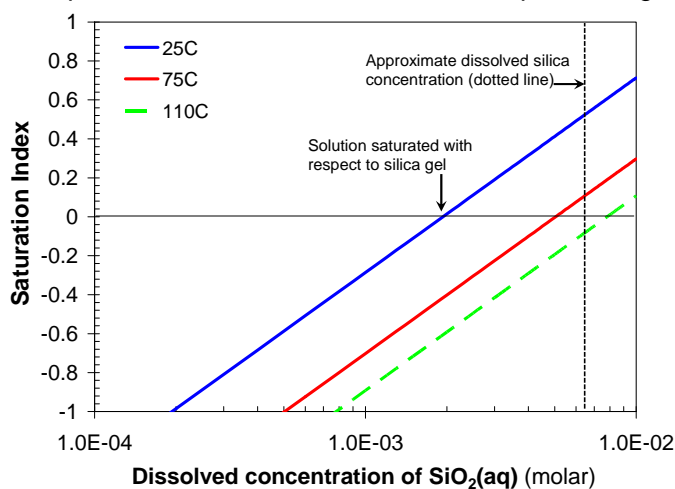

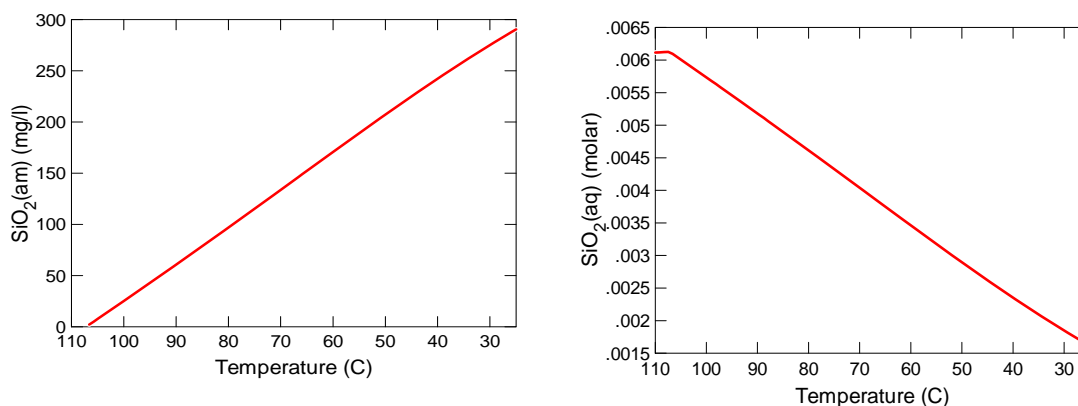Fig 2. Saturation index vs. dissolved concentration for silica gel in 4 molar nitric acid.

Fig 3. The diagram on the left shows the amount of silica gel (in miligrams per liter of solution) that is expected to precipitate as the solution is cooled from 110°C to 25°C. The diagram on the right shows how the dissolved concentration of silica varies as the solution is cooled.

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IAEA'S SAFETY REVIEW OF VINČA'S
SPENT FUEL REPACKAGING OPERATION

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1.   Introduction

The RA research reactor, at the "Vinca" Institute of Nuclear Sciences approximately 15 km South-East of downtown Belgrade, has been in extended shut down since 1984. The Serbian government decided in 2002 to a permanent shut down of the reactor and to participate in the repatriation programme to ship the spent nuclear fuel back to the country of origin, Russia.

A new Serbian Law [1] was issued on 12 May 2009 by which a new Agency for Ionizing Radiation and Nuclear Safety of the Republic of Serbia (Serbian Radiation Protection and Nuclear safety Authority: SRPNA) was established. The SRNPA replaced the Serbian Regulatory Commission (SRC) which was created in April 2005. The law required also that all nuclear liabilities and assets have to be transferred, from the Vinča Institute of Nuclear Sciences, to a new Public Company Nuclear Facilities of Serbia (PC NFS), including the responsibility for the research reactors, spent fuel and radioactive waste management at the Vinča Institute.

The repackaging of the degraded fuel elements, which was necessary before loading into the transport containers to ensure safe transportation, started early December 2009 and was completed the end of May 2010. The loading of the transport containers started in August 2010 and was completed mid November 2010. The spent fuel transport left the premises on 19 November 2010 and reached safely the final destination on December 22, 2010. IAEA's Safety Review activities from 2005 – 2009 of Vinča's Spent Nuclear Fuel Repackaging Operation were discussed during the RRFM 2010 meeting [2]. This paper discusses the safety review activities conducted in 2010.

2.   IAEA's support to ensure safe repackaging and transport

In the frame work of IAEA's Technical Cooperation projects RER/3/006 and SRB/4/002 a number of different activities were organised to advise the Vinca Institute of Nuclear Science, the former operator, PC NFS and SRPNA on all safety aspects and documentation to obtain the required licenses and to ensure safe and timely shipment of the spent nuclear fuel. During 2010 technical assistance was provided on all operational aspects and management system procedures to identify further improvement opportunities to ensure a safe and efficient repackaging and loading operation, for a further reduction of the dose to the staff and for a further reduction of the contamination risks.

3.   Nuclear safety and operational radiation protection

3.1   Licensing and shipment documentation

For the shipment of the Spent Nuclear Fuel many documents had to be prepared and several licences had to be applied for in different countries. These documents had to be prepared by the different departments of PC NFS, SOSNY (the main contractor) and TRANSING the transporter. The required

licences for the transport needed to be issued by Serbia, the trans-boundary countries (Hungary, Slovenia) and Russia the country of the final destination. Also Mayak the "end user" required documentation regarding the characterization of the spent fuel to ensure safe reception and temporary storage of the fuel. But PC NFS as formal consignor was formally responsible for both the licence applications as well as for the transport documentation. In order to be able to take this responsibility and ownership of the documentation, it was advised to appoint a Document Control Manager. To facilitates the tasks of the document control manager the IAEA introduced a Document Control Sheet (DCS) and assisted in the implementation of a dedicated document control procedure. The DCS was based on earlier experiences gained with the preparation and transport of spent fuel from Hungary to Russia [3]. The DCS facilitated the identification of all documents which were required for the transportation and the documentation to ensure safe loading and transportation in an efficient way. The DCS provided also information regarding the responsible organization and person for the preparation, the deadlines, the requesting organization, the status and revision dates of the documents.

The documents were categorized based on their objective (licensing, authorization, technical, operational procedures, etc.) and were divided into three main groups:
- Licensing and Authorisation Documents
- Mandatory Documents;
- Operational Documents

The 1st group of documents consisted of the formal licensing documents, including the applications and supporting documents to obtain all the required licenses. The second group of documents were required either by the end-user or by general shipment regulations (e.g.: invoices, packing lists) and the 3rd group of documents were the documents required by the license, by (inter)national regulations and the management system programme to ensure safe operations.

Regarding the group of Mandatory Documents the Agency ensured a consultative advisory and review support to their developments. During the meetings it was consequently emphasized that the main objective of the operational documentation was to ensure that all activities should be carried out with trained staff in a planned, controlled and documented way. The most important documents that were developed by PC NFS in this stage and advised on by the IAEA were the working procedures, templates and check lists, radiation and environmental protection plan, emergency preparedness, and a training plan covering all activities to ensure that all activities could be performed safely and in an efficient way.

A one day meeting was organised and shared by the IAEA, with all principle stakeholders (14) responsible for the preparation of the documents necessary to obtain a license and authorization for the transport, to obtain the required certificates for the containers, the documents required by the end users and the operational documents. During this meeting the available documents, the documents under preparation and the missing documents were identified. The stack holders, including the individuals responsible for the preparation and the required deadlines were agreed upon. In addition the bottle necks were identified and discussed and the required actions and the responsible for those actions were also identified. In addition the procedure to supply and distribute the documents and to update and distribute the DCS, in order that all the stakeholders were permanently updated on the status of all required documentation, was agreed

For the majority of the documents at least two and sometimes four hard copies in English, two or four in Russian and two electronic copies were required to accompany the shipment. In addition one set of all the documents was prepared for own use and one set was prepared to be available for inspection by SRNPA. The document control procedure was reviewed and advises were given regarding a more systematic way of identification of the documents, to prepare a table of content which could also be used as a checklist for the shipping documentation, and to archive the set of documents (both the hard copies as well as the electronic copies) at several physical separated locations to ensure that the required numbers of copies were always available even in case of incidents.

3.2   Trans-boundary licensing support

While the onsite work meant a unique technical challenge for the operators, the obtainment of all needed authorisations and permits provided a puzzle for the legal entity intending to ship the SNF by the means of different transport modes from a non EU country through the EU to the Russian Federation. Article 14 of the Council Directive 2006/117/Euratom of 20 November 2006 (hereinafter Shipment Directive) describes the procedure for obtaining authorisation for transit of spent fuel or radioactive waste through the Community as well as the required post authorisation activities. On the basis of the Shipment Directive, the Commission Decision of 5 March 2008 established the standard documents for the supervision and control of shipments of radioactive waste and spent fuel as referred to in the Council Directive 2006/117/Euratom. The commission decision consists of the sections B-1 (application for authorisation of shipment of spent fuel) until B-6 (acknowledgement of receipt of the spent fuel). The natural or legal person who has the responsibility for managing the shipment within the first Member State of transit has to respect and act according to the provisions from both above mentioned documents. Although the transit countries were EU members, the procedures were only partly harmonised, and, of course, the initial country's and target country's regulations and procedural rules were completely different. Therefore, on this subject, the Agency and his experts provided long lasting complex advisory and coordination support.

3.3   Emergency Preparedness

The implementation of the local emergency response plans and the drill plans which were developed during some missions in 2009 were reviewed and advices were given for further improvement. Especially the additional requirements for local exercises, which were not all performed conform the planning due to the time pressure for the repackaging and loading activities, were discussed. Also the draft external response plans were discussed and advices were given regarding the use of emergency cards, training of all involved staff members and practical exercises. In order to increase the emergency preparedness a training week on emergency preparedness was organized in July 2010, with all internal and external response forces. This training week was the final preparation of all the response teams for the National Emergency exercise which took place in the 3rd quarter of 2010.
The emergency plans and the exercises were discussed and agreed upon in a Governmental Working Group, in which the ministries and civil agencies supporting the project participated.

3.4   Unusual events

During the repackaging and loading operation 4 unusual contamination events were reported. Two workers were slightly contaminated in their face with some drops of contaminated water and in two occasions contaminated hands were reported. The workers did not wear eye protections since the available equipment did not have sufficient visibility under the heavy duty circumstances. The contaminated staff members were examined by the medical department and a whole body counting was performed. Although not contaminated, one worker was sent for medical examination since he was concerned about possible contamination. The four external contaminations were very modest not resulting in additional effective dose consequences and no internal contamination was observed. The events were all discussed in the combined Nuclear, Health Physic Safety and Security Committee and new spectacles with better visibility in heavy duty circumstances were ordered. Advices were given to improve and formalize the registration of unusual events. A practical example for such a procedure was provided. No internal contamination was detected by the workers.

3.5   Readiness review

From 27 – 30 July 2010 a safety and documentation readiness review was organized and conducted by an Agency lead expert team [4], to review the safety aspects and documentation readiness in order to identify the status of the implemented safety measures and the status of the required documentation in

order to recommend further actions to ensure safe and timely shipment of the Spent Nuclear Fuel. One team reviewed the status of the licensing procedures required in the country of origin (Serbia), the transit countries (Hungary and Slovenia) and in the country of destination (Russia). A second team was devoted to review the technical documentation, which has to be prepared for the shipment, and the third team reviewed the safety provisions and operational procedures to ensure safe loading and shipment of the SNF.

The team recognized the technical competences, safety awareness, and the substantial progress which was made in the preparation of the required documentation for the shipment of the SNF. The documents reviewed, the discussions held, the training programme being conducted, and the equipment shown established a high degree of confidence in a safe loading and shipment operation.

The team made recommendations and suggestions which should be implemented to ensure timely finalization of all required documentation and timely implementation of the documents and emergency plans to ensure a safe and smooth shipment.

These recommendations are mainly associated with the need to:

- Finalize the radiological and security emergency plans in consultancy with all stakeholders from the local and national authorities and define and implement an action plan in consultation with the involved authorities;
- Formalize the changes in the processes as defined in the FTSR and in the Technical Specifications of Mayak;
- Inform the Agency for Ionizing Radiation Protection and Nuclear Safety of Serbia on the process changes and on the latest revision of the procedures to be used during the loading and transportation;
- Further improve the radiation protection plan and its implementation;
- Revise the Document Control Sheet in a more systematic manner;
- Keep clear records of all applicable documents, including copies of applications and their related application documentation;
- Keep the required number of paper copies and an electronic copy in a well-defined location and keep an additional hard copy and an electronic copy in a separate location;
- Include the documents as specified in the post authorization requirements of the Council Directive 2006/117/EURATOM in the Document Control Sheet;
- Finalize the preparation of the templates which need to be filled in during loading and for the preparation of the shipment; and
- Prepare the necessary checklists for the required notifications, for smooth loading operations and for the readiness check for transportation.

3.6   Radiation protection

As reported in [2] a Water Chemistry Control System (WCCS) was installed to reduce the caesium from the basin water. A balance between the reduction of the activity in the basin water and the collection of radioactive material in the filters, resulting in possible exchange of the filters and higher dose levels for the staff, was applied. The WCCS was operated to maintain basin activity below a level of about 300 Bq/ml being a factor 3 lower than the activity level of the basin water at the start of the project.

The last year introduced staff surveillance programme, by which the estimated anticipated dose per shift and the actual received dose were registered, was used as a management tool to optimise the balance between the operation of the WCCS and dose received by the staff.

During several missions the required radiation protection provisions for the transport were reviewed and discussed in detail. The procedures and templates for the radiation and contamination measurements of the casks and ISO transport containers were discussed and practical examples for

efficient contamination controls were provided. In addition advices were provided to ensure that all the instruments which would be used for the preparation of the transport were calibrated.

Due to the higher radiation levels in the working area above the storage pool and the high activity levels from the stored components a detailed training and dose surveillance programme for the safeguards inspectors was discussed and agreed in order to ensure that the received doses by the safeguard inspectors would be as low as reasonable achievable. In addition advises were given to relocate all high active components to reduce the dose levels around the working area to the extent possible.
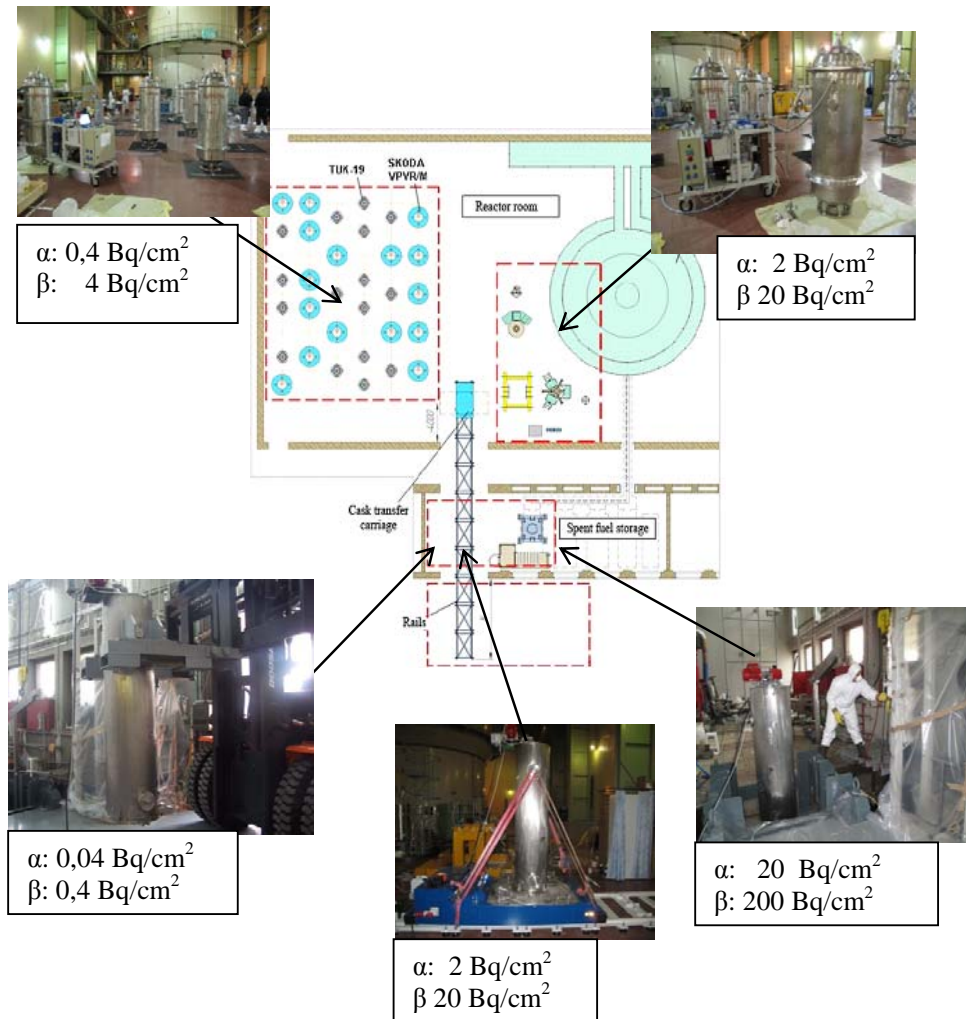

Figure 1: Layout of the spent fuel basin and reactor hall with the allowed contamination levels

Several operating organizations in different countries executed a fuel repatriation programme in 2010 and consequently the TUK19 and Skoda transport containers had to be shared amongst them, resulting in a limited availability for the loading of the SNF especially in the TUK-19 containers by PC NFS.

To avoid unnecessary decontamination activities, which could result in a delay of the loading process, three different controlled radiation protection zones were introduced and the allowable contamination levels with associated protective measures were defined and agreed upon. In addition the ventilation regimes, which have to be applied to avoid contamination of the environment, were defined. It was agreed to carry out the contamination measurements before every shift in order to allow for decontamination before the start of the shift by which the delay in the loading process was reduced to the minimum. Also advises were provided to use professional V-shape floor wipers for the contamination control of large floor areas resulting in more efficient surveys.

In order to increase the visibility of the basin water, which was decreased due to the formation of algae, a commercially available mobile water cleaning system similar to the ones being used for cleaning of small swimming pools, was ordered. For this commercial design, the filter housing had to be installed outside the water, which could result in an unshielded radiation source. The Radiation Protection Supervisor was not involved in the purchase of the cleaning system, and no radiation protection measures were defined. Advices were given to improve the appropriate radiation protection measures to ensure operation of the cleaning system without undue risks for the workers or without undue risks for spreading of contamination. Also the requirement to obtain the advice of the Radiation Protection Supervisor for every purchase where radiation protections aspects might be involved, was added to the purchasing procedure.

The cumulative dose received by the staff during the whole repackaging and loading operation was 14 man-mSv, The average dose amounted to 0,25 mSv and the maximum dose received by a staff member was 0,72 mSv. These values are 1/7 of the estimated doses as presented in the Safety Analyses report.

None of the staff members were internally contaminated due to the repackaging and loading operations. Taking the heavily contaminated basin water into account it could be concluded that all implemented radiation protection measures were effective.

4.   Conclusions

The IAEA implemented through the Technical Cooperation programme extensive assistance to the operating organisation for the preparation and repatriation of the Vinča's Spent Nuclear Fuel to the country of origin. The assistance provided by the IAEA, the efficient implementation of the recommendations and the extended training programme together with the technical competence of the counterpart and the contractor, ensured safe and efficient repatriation, with improved radiological conditions for the planned operations.

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PROTECTIVE COATINGS FOR WET STORAGE OF ALUMINIUM-CLAD SPENT FUEL

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ABSTRACT

Corrosion protection of spent RR fuel for long term wet storage was considered important, primarily from the safety standpoint and the use of conversion coatings was proposed in 2008. This paper presents the results of: (a) on-going field tests in which un-coated and lanthanide-based conversion coated Al alloy coupons were exposed to the IEA-R1 reactor spent fuel basin for durations of up to a year; (b) preparation of cerium modified hydrotalcite coatings and cerium sealed boehmite coatings on AA 6061 alloy; (c) corrosion resistance of coated specimens in NaCl solutions. The field studies indicated that the oxidized and cerium dioxide coated coupons were the most corrosion resistant. The cerium modified hydrotalcite and cerium sealed boehmite coated specimens showed marked increase in pitting corrosion resistance.

1. Introduction

It is well known that over 62,000 Al-clad research reactor spent fuel assemblies are stored in wet facilities around the world. [1] Most of the storage facilities have water quality management programmes, to prevent and/or reduce degradation of the fuel cladding. Pitting corrosion is the main form of degradation which could breach the cladding and release fissile material, contaminating thereby storage facilities as well as other stored fuels. It has been shown that maintenance of water parameters within specified limits does not prevent pitting corrosion of the fuel cladding, due to synergism between many basin water parameters that affect corrosion of aluminium and its alloys. [2-4] Therefore it is important to consider some form of corrosion protection for stored spent RR fuel. Use of conversion coatings is a well established corrosion control technique in many industries.

Formation of cerium hydroxide films on Al alloys immersed in solutions containing cerium compounds as inhibitors has lead to the development of rare earth based conversion coatings on Al alloys, mainly to substitute hexavalent chromium, a known human carcinogen. [5-9] Use of conversion coatings, to protect spent research reactor fuel assemblies, was proposed in 2007 and laboratory as well as field investigations carried out at IPEN in Brazil. Preliminary results revealed that the pitting corrosion resistance of aluminium alloys AA 1100 and AA 6061, used as cladding of RR fuel plates or elements, increased when coated with lanthanide-based compounds. [4, 10] These investigations were subsequently extended to include cerium modified boehmite and hydrotalcite coatings on Al alloy surfaces. Inclusion of cerium impregnated with boehmite was motivated by the fact that a thick layer of boehmite forms on spent fuel surfaces. Hydrotalcite (HTC) is lithium aluminium-nitrate-hydroxide hydrate (a form of talc) and it forms on Al alloys, when the alloy is immersed in an appropriate alkaline Li salt solutions. [11-13]. Immersion of Al alloys in such solutions result

in formation of a polycrystalline barrier film composed mainly of HTC like compounds. The main advantage of this coating process is its simplicity.

This paper presents the results of: (a) on-going field tests in which uncoated and lanthanide-based conversion coated Al alloy coupons were exposed to the IEA-R1 reactor spent fuel basin for durations of up to a year; (b) the formation of and the corrosion resistance of cerium modified hydrotalcite coatings and cerium sealed boehmite coatings on AA 6061.

2. Methods and materials

Aluminium alloys AA 1100 and AA 6061 (Table 1) were used in the first set of laboratory and exposure tests. However, only AA 6061 was used to study boehmite and hydrotalcite (HTC) coatings. Details of specimen and coupon preparation, surface treatment, application of lanthanide coatings and laboratory corrosion testing in the first set of experiments can be found elsewhere. [4]

Table 1. Chemical composition of aluminium alloys (wt%)

Alloy	Cu	Mg	Mn	Si	Fe	Ti	Zn	Cr
AA 1100	0.16	<0.1	0.05	0.16	0.48	0.005	0.03	0.005
AA 6061	0.25	0.94	0.12	0.65	0.24	0.04	0.03	0.04

The field tests consisted of preparing Al alloy coupons, stacking of these coupons in racks, immersion of the racks in the spent fuel section of the IEA-R1 research reactor in IPEN, Brazil, for 2 and 12 months, removal of the racks and examination of the coupons. This procedure was used in the IAEA coordinated CRP on "Corrosion of Research Reactor Aluminium Clad Spent Fuel in Water". [2] Circular coupons 10 cm in diameter and 3 mm thick of the two alloys were coated with oxides of cerium, lanthanum, praseodymium and an oxide concentrate of cerium. [10] The coupon stacking sequence in the racks from top to bottom consisted of as-received; oxidized; cerium oxide coated; cerium oxide concentrate coated; praseodymium oxide coated; lanthanum oxide coated. Some coupons were oxidized at 300° C in air for 4 h to form a surface oxide layer to simulate spent fuel plate surfaces. The surface features of AA 1050 and AA 6061 coupons exposed for 2 and 12 months were examined, both visually and with an optical microscope.

Table 2. Aqueous solutions, their composition and experimental conditions used to prepare and to coat AA 6061 specimens.

Solution	Purpose	Composition of solution and conditions
1	Degrease	25 g/L $Na_2 SiO_3$; 25 g/L $Na_2 CO_3$; 65 °C; 2 minutes.
2	Deoxidize	10% HNO_3; 3% $NaBrO_3$; 55 °C; 3 minutes.
3	Form boehmite	Deionized water; 97-100° C; 5 minutes.
4	Incorporate Ce in boehmite	0.1% $CeCl_3$; 97 °C; pH 4; 5 minutes.
5	Incorporate Ce, Li and Al in boehmite	0.1% $CeCl_3$; 1% $LiNO_3$; 1% $AlNO_3$; 97 °C; pH-4; 5 minutes.
6	Form hydrotalcite	6.9g/L $LiNO_3$; 28.3 g/L KNO_3; 2.4 g/L LiOH; 0.06 g/L $NaAlO_2$; 98 °C; pH 12; 10 minutes.
7	Incorporate Ce in hydrotalcite	10 g/L Ce $(NO_3)_3$; 30% H_2O_2; room temperature.

To prepare boehmite and HTC coatings, AA 6061 specimens (2 x 2 x 0.2 cm) were first degreased in solution 1 and deoxidized in solution 2, under conditions shown in Table 2. The specimens were then boehmite or HTC coated by immersion in solutions 3 or 6 respectively.

HTC formed on a specimen surface is shown in figure 1. Some of the boehmite coated specimens were further treated in solution 4 or 5 to incorporate Ce or Ce+Li+Al respectively. Similarly, some HTC coated specimens were treated in solution 7 to modify it with Ce. The electrochemical behaviour of uncoated and coated specimens was determined from anodic potentiodynamic polarization measurements carried out with a standard 3-electrode arrangement in 0.1 M NaCl, using a saturated calomel electrode (SCE) as the reference electrode.
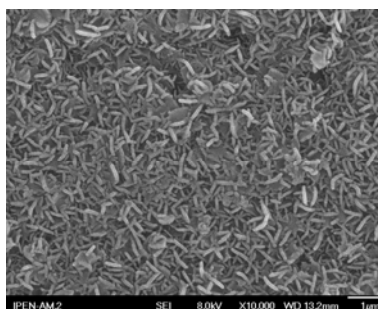

Fig 1. Hydrotalcite layer on the surface of AA 6062 specimen.

3. Results and discussion

3.1. Coupons exposed to IEA-R1 reactor spent fuel section.

The coupons from the different racks were dismounted, rinsed, decontaminated, dried and examined visually and with an optical microscope. The top surfaces of all coupons revealed more pits compared to the bottom facing surface of the same coupon, indicating the influence of settled solids on the top surfaces. Table 3 summarizes the main features observed on the top surfaces of the different coupons. Overall, the coupons exposed for 12 months were significantly darker, either brown or black, indicating formation of a thicker surface oxide, compared to corresponding coupons exposed for 2 months.

Table 3. Surface features of coated and lanthanide coated AA 1050 and AA 6061 coupons exposed for 2 and 12 months to the IEA-R1 reactor spent fuel section.

Alloy	Treatment	Surface features after exposure to IEA-R1 spent fuel basin for	
		2 months	12 months
AA 1050	None	Dull, no pits	Dark, stained, some pits
	+ CeO$_2$	**Yellow bright, no pits**	**Mostly bright, no pits**
	+ La$_2$O$_3$	Bright, many pits	Dark, many pits
	Oxidized	Dull, no pits	Dark, few pits
	Oxidized + CeO$_2$	**Yellow bright, no pits**	**Dark, few pits**
	Oxidized +CeO$_2$ conc	Bright, few pits	Dark, many pits
	Oxidized + La$_2$O$_3$	Bright, many pits	Brown, many pits
	Oxidized + Pr$_2$O$_3$	Bright, many pits	Dark, many pits
AA 6061	None	Dull, no pits	Dark brown, many pits
	+ CeO$_2$	**Yellow, no pits**	**Very dark, very few pits**
	+ CeO$_2$ conc	Light brown, few pits	Dark surface, many pits
	+ La$_2$O$_3$	Dull, many pits	Dark surface, many pits
	Oxidized	Almost bright, no pits	Dark, some pits
	Oxidized + CeO$_2$	**Semi bright, no pits**	**Dark, few pits**
	Oxidized + La$_2$O$_3$	Semi-bright, many pits	Dark, many pits
	Oxidized + Pr$_2$O$_3$	Semi-bright, many pits	Dark, many more pits

A study of the surface features of the different coupons revealed that in general after 12 months of exposure to the spent fuel section of the IEA-R1 reactor, the CeO_2 coated coupons were the most resistant to pitting corrosion. The coupons coated with La_2O_3 and Pr_2O_3 were heavily pitted after just 2 months of exposure. Even though preliminary laboratory tests indicated improved corrosion resistance of La_2O_3 and Pr_2O_3 coated specimens, to the same extent as CeO_2 coated specimens, long term field tests have shown otherwise. [4] This could be attributed to formation of only soluble Pr and La hydroxides on the coupon surface, and inability therefore to provide 'active corrosion protection', like that provided by cerium coated coupons.

3.2.   Corrosion behavior of boehmite and HTC coated specimens.

The electrochemical behaviour alloy AA 6061, with or without the coatings revealed differences in the anodic as well as the cathodic polarization curves. The free corrosion potential (E_{corr}) and the pitting potential (E_{pit}) were recorded and the corrosion current densities (i_{corr}) determined by the Tafel extrapolation method. [10]

Table 4. Corrosion current (i_{corr}), corrosion potential (E_{corr}) and pitting potential (E_{pit}) of alloy AA 6061 in 0.1M NaCl

Surface condition	i_{corr} ($mA.cm^{-2}$)	E_{corr} (mV vs SCE)	E_{pit} (mV vs SCE)
None	1.5×10^{-6}	- 760	- 750
Boehmite	2.0×10^{-7}	- 711	- 650
Boehmite + Ce	3.5×10^{-7}	- 754	- 600
Boehmite + Ce + Li + Al	7.1×10^{-7}	- 741	- 565
Hydrotalcite	3.5×10^{-7}	- 718	- 580
Hydrotalcite + Ce	4.0×10^{-7}	- 764	- 420

Table 4 summarizes the corrosion current, the corrosion potential and the pitting potential of the AA 6061 specimens with the different coatings. The E_{pit} of the untreated specimen was very close to its E_{corr} signifying active corrosion. With boehmite on the surface, the E_{pit} increased to -650 mV. Modification of the boehmite with cerium hydroxide increased E_{pit} even more to -600 mV. This indicates an increase in the pitting corrosion resistance upon introduction of Ce into the pores of the boehmite coating. Impregnation of the boehmite with Ce, Li and Al resulted in a further increase in the pitting corrosion resistance by increasing the E_{pit} to -565 mV. A HTC layer on the alloy surface resulted in an even more marked increase in the pitting resistance by increasing the E_{pit} from -750 to -580 mV. This increase in pitting resistance of the alloy with a HTC coat was further enhanced by modifying HTC with Ce. The E_{pit} increased significantly to -420 mV. The cathodic current density of the AA 6061 specimen in 0.1 M NaCl decreased by an order of magnitude upon coating it with either boehmite or HTC. The difference in the i_{corr} between the boehmite coated and the HTC coated specimens was slight with no marked change even with the introduction of Ce into the coating. The open circuit or corrosion potential E_{corr} of AA 6061 increased to almost the same extent with formation of boehmite or HTC. Modification of either coating with Ce increased E_{corr} to again the same extent.

The mechanism by which HTC imparts pitting corrosion protection is refered to as 'passive corrosion protection', in other words, the HTC acts as a physical barrier between the solution and the surface. On the other hand the mechanism by which the cerium modified HTC imparts protection is considered to be 'active corrosion protection', analogous to that in chromium coatings. This involves release of Ce ions from the coating, transport through the solution and action at defect sites to stifle corrosion. It has been speculated that if a Ce^{4+} bearing inorganic coating contacts a solution, soluble Ce^{4+} is released into the solution.

When these ions encounter reducing conditions, like those associated with exposed bare metal at coating defects, it reduces to Ce^{3+}, which forms an insoluble hydroxide and precipitates. The precipitated Cerium hydroxide at the defect then stifles further corrosion.

4. Conclusions

1. The CeO_2 coated coupons were the most resistant to pitting corrosion after 12 months of exposure to the spent fuel section of the IEA-R1 reactor.
2. The electrochemical measurements revealed that the pitting corrosion resistance of AA 6061 alloy increased when it was coated with boehmite. Impregnation of the boehmite with Ce increased further the pitting corrosion resistance.
3. The pitting corrosion resistance of HTC coated AA 6061 specimens was higher than those coated with boehmite. Cerium modification of the HTC coating increased pitting corrosion resistance further.
4. A coating process involving simple immersion in several solutions to form cerium containing conversion coats on Al alloys, and thereby increasing the pitting corrosion resistance of the alloy, has the potential for scale-up to increase the corrosion resistance of Al-clad spent RR fuel assemblies during long term wet storage.

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WATER CHEMISTRY CONTROL SYSTEM FOR RECOVERY OF DAMAGED AND DEGRADED SPENT FUEL

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ABSTRACT

The International Atomic Energy Agency (IAEA) and the government of Serbia have led the project co-sponsored by the U.S, Russia, European Commission, and others to repackage and repatriate approximately 8000 spent fuel elements from the RA reactor fuel storage basins at the VINČA Institute of Nuclear Sciences to Russia for reprocessing. The repackaging and transportation activities were implemented by a Russian consortium which includes the Sosny Company, Tekhsnabeksport (TENEX) and Mayak Production Association. High activity of the water of the fuel storage basin posed serious risk and challenges to the fuel removal from storage containers and repackaging for transportation. The risk centered on personnel exposure, even above the basin water, due to the high water activity levels caused by Cs-137 leached from fuel elements with failed cladding. A team of engineers from the U.S. DOE-NNSA's Global Threat Reduction Initiative, the Vinca Institute, and the IAEA performed the design, development, and deployment of a compact underwater water chemistry control system (WCCS) to remove the Cs-137 from the basin water and enable personnel safety above the basin water for repackaging operations. Key elements of the WCCS system included filters, multiple columns containing an inorganic sorbent, submersible pumps and flow meters. All system components were designed to be remotely serviceable and replaceable. The system was assembled and successfully deployed at the Vinca basin to support the fuel removal and repackaging activities. Following the successful operations, the Cs-137 is now safely contained and consolidated on the zeolite sorbent used in the columns of the WCCS, and the fuel has been removed from the basins. This paper reviews the functional requirements, design, and deployment of the WCCS.

1. Introduction

Isolation canisters to store fuel assemblies with breached cladding in basins may have significant amounts of cesium dissolved in the local water trapped inside them. Because cesium quickly diffuses into the general basin when a canister is opened, radioactivity dose rates can increase to high levels for workers in the area. One such case involved a set of isolation canisters containing pieces of fuel that leaked very large amounts of activity (up to an estimated $6x10^6$ Bq/ml) into the local water of the isolation canisters that were themselves in underwater storage in the Receiving Basin for Offsite Fuel (RBOF) at the Savannah River Site (SRS). The plan for RBOF deinventory necessitated the recovery of the fuel for transportation to the L-basin at the site. A special design of deionizer was needed to provide local deionization of the canister water without requiring large quantities of the RBOF basin water to be processed.

A special underwater deionizer (UD) was designed, fabricated, and successfully deployed to process the high activity water and enable fuel retrieval from the SRS isolation canisters [1, 2]. The UD was operated as a single pass system, flushing water (for 10 hours total) from 6 large isolation canisters (416 L internal volume in each) containing fuel pieces through a column containing cation exchange media. This took less time and had less operational impact than opening the canisters and deionizing the entire basin volume (9500 m^3). This approach to fuel recovery resulted in deinventory of the basin well ahead of schedule.

A similar fuel recovery and repackaging campaign was planned at the Vinca Institute to recover aluminium-clad fuel stored in sealed aluminium barrels and stainless steel channel holders. The Vinca Institute, IAEA, U.S. NNSA/DOE, Sosny Company, Tekhsnabeksport (TENEX) and Mayak Production Association planned the overall campaign to repackage and transfer the fuel from Vinca to the Mayak facility in Russia for reprocessing. An important component of the overall operations was the need for clean up of the cesium from the Vinca basin water before and during the repackaging operations.

A Water Chemistry Control System (WCCS) based on the UD used at SRS, was designed by Savannah River National Laboratory (SRNL) personnel under U.S. Department of Energy (DOE) sponsorship. It was designed to remove cesium from the bulk basin water and trap the cesium on underwater columns containing sorbents while meeting the safety requirements at the Vinca site and also the space limitations, repackaging cycle time, and overall project schedule needs.

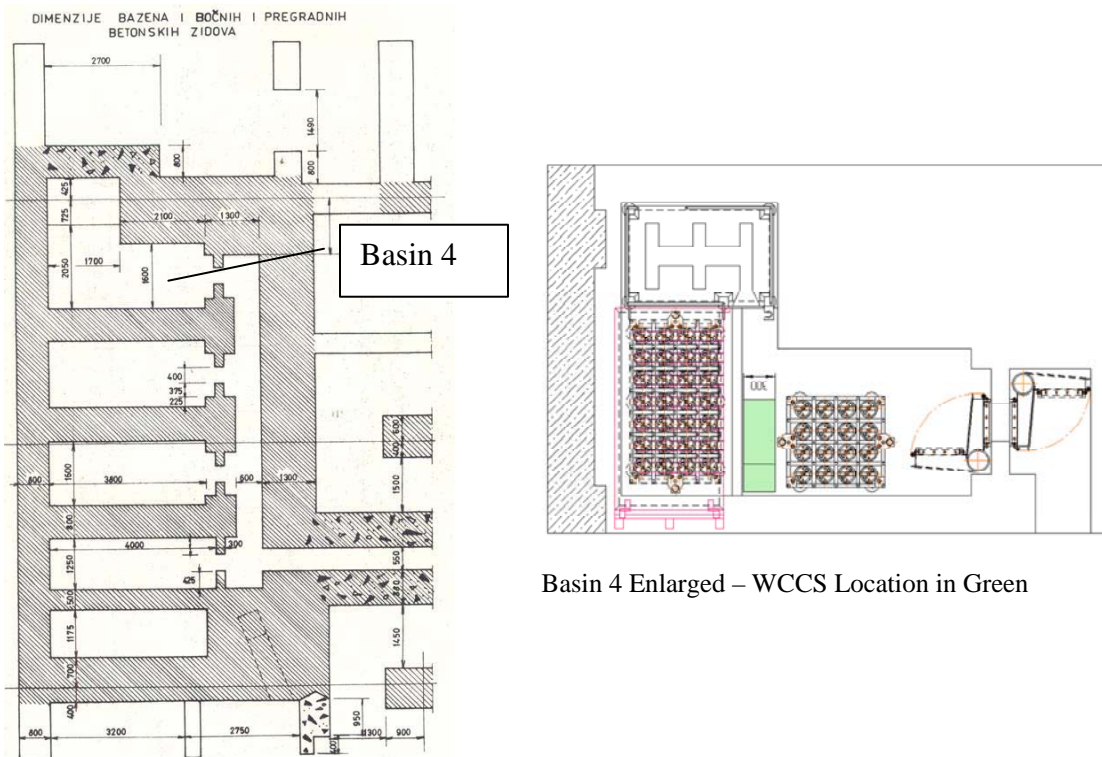The Vinca fuel storage basins are arranged as shown in Figure 1 below.

Basin 4 Enlarged – WCCS Location in Green

Figure 1. Vinca Basins 1-4 and Basin 4 (enlarged) with Repackaging Equipment Locations

2. Water Chemistry Control System (WCCS) for Vinca Fuel Reprocessing

2.1 Functional Requirements

Technical requirements for the WCCS were prepared by the repackaging contractor for the SRNL design team to identify the functional and safety requirements of the WCCS. The requirements included:

- Fit in allotted underwater space (green zone)
- Reduce activity from 10^7 to $\leq 10^5$ Bq/l in a 40 m^3 volume in 24 hrs
- Filter particulates
- Upstream sample port
- Monitor fluid flow

- Repairable/replaceable parts
- Remote component replacement
- Remotely controlled
- Tools to operate/maintain WCCS
- Support safety assessment
- Avoid interference with other equipment

The repackaging equipment dimensions left limited space for the WCCS (footprint 1 m x 0.3 m x 1.5 m high) in Vinca's basin 4 (see Figure 1). Using available space and sorbent properties, SRNL maximized WCCS capacity and flow rates. Design iterations incorporated input from all parties. The selected sorbent media was ResinTech® SIR-600, an inorganic-based ion exchanger, highly selective for cesium removal and radiation-resistant. Figure 2 shows the WCCS as it was configured at Vinca installation.
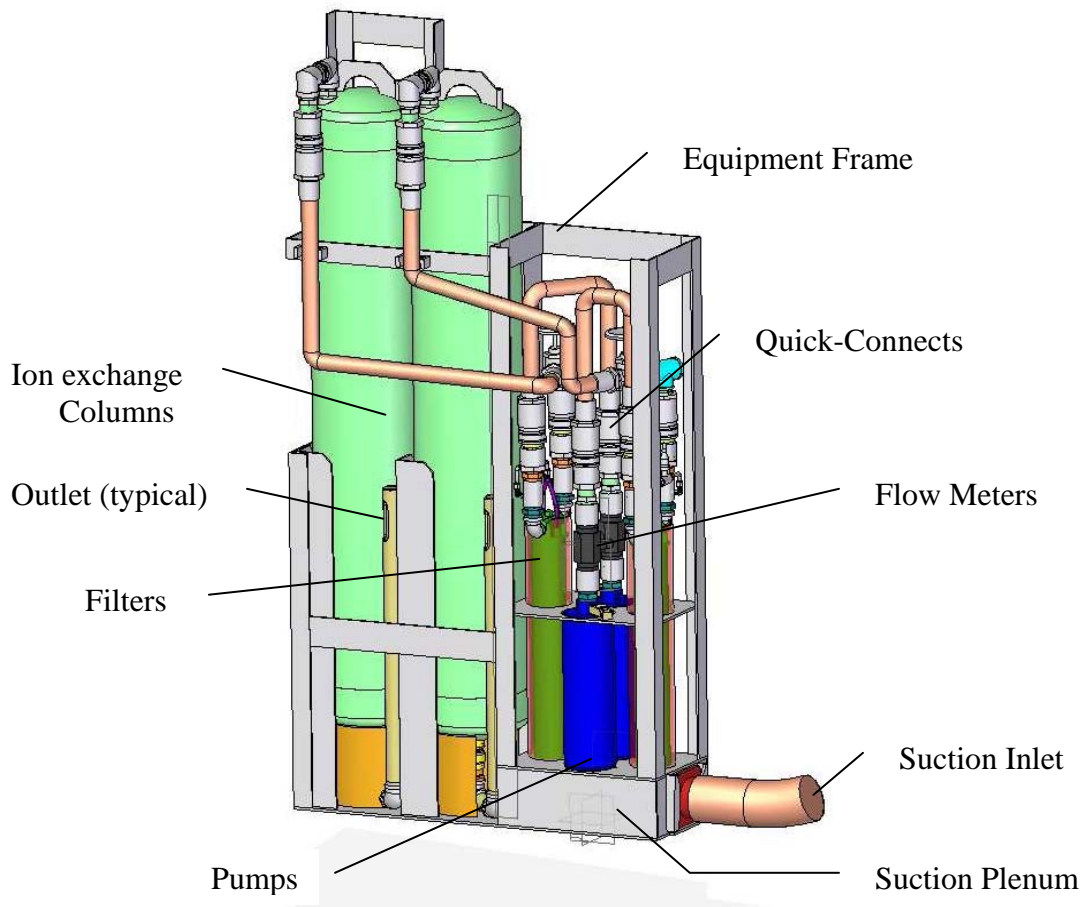

Figure 2: Water Chemistry Control System for Vinca Institute Fuel Repackaging

The unloading process was expected to release a large quantity of dissolved cesium into basin 4. Cumulative activity was expected to be 10^{12} Bq. Vinca fuel storage containers were not designed to allow purging before opening. Therefore any cesium dissolved inside the container would be mixed with basin water when the storage can was opened. Activity removal from basin water was estimated using the formula

$$C(t) = R/(Q\,\epsilon) + [C_0 - R/(Q\,\epsilon)]e^{-(Q*\epsilon/V)t}$$

Where,

C(t) is the concentration of activity in Bq/mL in a volume of water at time t;
R is the total activity released into a volume of water in Bq/s (e.g. from all leaking fuel);
Q is the volumetric flow rate in L/s into and out of a closed loop deionization system;
V is the volume of water in the basin; and
ϵ is the efficiency of the deionizer system.

Figure 3 below shows an example of results for the WCCS operation to reduce the activity in the total volume of basin water (200 m^3) from the interconnected basins in comparison to the predicted (theory) activity removal results. The difference is primarily attributed to a finite release rate R that was assumed to be zero in the predicted results.
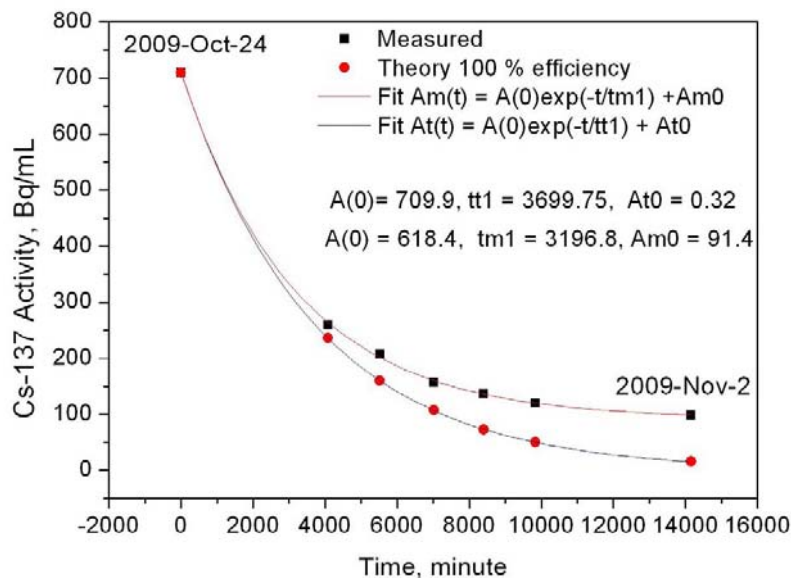

Figure 3. Activity Removal Results from the Total Water Volume (200 m^3)

2.2   WCCS Design

The WCCS is arranged as two independent systems, each with a pump, filter, and sorbent column. Components were designed to be remotely replaceable, connected with flexible reinforced hose. Connectors are valved, push-to-connect, quick-disconnect couplings. The multi-stage centrifugal type pumps are submersible, electrically powered, and meet system head requirements and flow characteristics (up to 75 litres/min each). The pumps fit into gasketed openings in a suction plenum at the bottom of the frame, drawing water into the plenum through a single suction hose (see Figure 2).

Drilling into the "aluminium barrels" containing the fuel was necessary to remove the fuel and the activity contained within the water of the barrels. Any debris emanating from the drilling was intended to be released into basin 4 (40 m^3 volume) only. Suction nozzle placement and low fluid velocity in the suction hose keep cuttings from entering the pumps. The suction plenum slows fluid further and strainers on the pump suction keep objects from entering the pumps. A turbine-type flow meter mounted at the outlet of each pump provides reasonable flow indication although not optimally positioned due to constraints in the WCCS design to meet the requirements.

Hoses connect the flow meters to filters downstream of the pumps. Filters have stainless housings containing woven stainless cartridges with mesh openings of 100 microns. Filter housings sit in a tubular socket mounted atop the suction plenum, kept from rotating by slots accepting the filter inlet and outlet piping. Filter housings are not fixed in any other way, allowing them to be replaced as necessary by disconnecting their hoses and lifting them out of the socket using the lifting eyes attached to the top of the housings. A differential pressure transducer across the filter provides information to the control stand, allowing operators to anticipate filter replacement as the pressure differential increases.

Filtered water flows to the top of the resin column, through an internal diffuser and into the resin. Each column has a capacity of 56 litres of resin, allowing a small amount of space for resin swelling, if necessary. Non-swelling sorbent media, such as zeolite, may occupy the entire space between the column's internal baffles. The sorbent ultimately selected for the operation was a zeolite sorbent that preferentially captured cesium compared to other water impurity species that were present (e.g. Ca, Mg). The system provided for the maximum capacity in flow rate and capture volume for the space it occupied. Water exiting the column flowed to an outlet about 1 m above the base of the WCCS frame (see Figure 2).

2.3   WCCS Deployment

Fabrication of the WCCS design was contracted by the IAEA to a fabricator (VUJE) in Slovakia. SRNL designers consulted with the fabricator on questions of design and allowable part substitutions. Upon completion, team members met at the fabricator's facility to inspect the WCCS hardware, review the documentation, and observe a functional test of the WCCS.

The WCCS was shipped to Belgrade for installation in Vinca basin 4 in June 2009. Vinca operations staff modified the WCCS to meet their particular operating and maintenance needs. Following these actions, the device was installed in basin 4 (see Figure 4).

The initial flow rate through each of the two lines in was approximately 70 litres/minute, and the pressure differential across the filters was approximately 0.25 bar.

Figure 4. WCCS Installation in Vinca Fuel Storage Basin 4

2.4 WCCS Operations Summary

The activity in the entire basin water volume was reduced prior to repackaging operations from the initial value of approximately 700 Bq/ml to approximately 100 Bq/ml.

The first aluminium barrel was opened in December 2009, and the WCCS was turned on to operate intermittently, as needed. The water activity increased up to an estimated level of 3000 Bq/ml when some of the barrels were opened, but was quickly reduced by continued WCCS operation to at or below approximately 300 Bq/ml prior to resuming operations. This significantly reduced the levels of exposure to the operators, located on platforms directly over the basin water, who were performing the repackaging.

Large volumes of debris also emanated from the barrels as the fuel was retrieved, eventually causing plugging of the filters, and reduced flow rates in each of the two lines of the WCCS that were operated simultaneously. The filters were replaced when the pressure differential reached approximately 1 bar. The system was successfully returned to operation with the flow rates and pressure differential parameters in both lines at the initial, pre-operation conditions of 70 litres per minute and 0.25 bar, respectively. All the fuel from 30 aluminium barrels and 297 channel holders in the Vinca basins was safely recovered by August 2010.

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IMPACT OF BERYLLIUM REFLECTOR AGEING ON SAFARI-1 REACTOR CORE PARAMETERS

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ABSTRACT

Build-up of ^{6}Li and ^{3}He, that is, the strong thermal neutron absorbers or the so called poisons, in beryllium (Be) due to fast neutron irradiation is an important topic in the management of all Be reflected reactors. The presence of these isotopes in the Be reflector changes the physical characteristics of the reactor, such as reactivity, neutron spectra, flux level, power distribution, etc., and therefore must be accounted for in reactor core calculations. This paper presents detailed calculations to quantify the Be reflector poisoning phenomenon as well as the influence of impurities that were originally present in the fresh Be reflector elements of the SAFARI-1 research reactor. The calculational methodology uses MCNP to compute the spectrum over the Be reflector elements, FISPACT activation code to calculate isotopic number densities of ^{6}Li and ^{3}He as well as that of other secondary impurities, as a function of the reactor power history, and finally the OSCAR-3 code system to quantify the impact of Be poisoning on core parameters.

1.   Introduction

SAFARI-1 is a tank-in-pool MTR type research reactor licensed to operate at 20MW. The reactor has been operated for about 45 years since its commissioning in 1965. It has an 8 × 9 core lattice, housing 26 fuel elements, 5 control rods, 1 regulating rod, a number of solid lead shield elements, solid and hollow aluminium filler elements as well as solid and hollow Be reflector elements. The Be reflectors are used on three sides of the core and most of them, originally installed in 1965 are still in use. The reactor has been operated on fully LEU core since 2009. Currently the reactor is operated for about 30 days, interrupted by a 4 - 5 day refuelling shut down period and one extended 12 days refuelling and maintenance shut down per annum [1]. The OSCAR code system [2] has now been in use for more than a decade as a primary tool for fuel management calculations such as reloads and core follow. However, in the OSCAR-3 code Be reflector burn-up is not accounted for; OSCAR models Be as a non-burnable material and as 100% pure Be. The individual neutron irradiation history of each Be reflector element, as well as the impact of Be poisoning on reactor parameters, were never well known nor investigated before for SAFARI-1 reactor. In recent years bending of some Be reflector element has been observed, suggesting ageing of the reflector elements.

This paper presents detailed calculations to quantify the Be reflector poisoning phenomenon and its impact on SAFARI-1 reactor core parameters. The influence of impurities that were originally present in the fresh Be reflector elements (1965), on the past and present thermal neutron absorption cross section of the reflector, was also investigated.

1.1   Be reflector placement in SAFARI-1 research reactor

The Be reflectors, of which the hollow elements are used for sample irradiation purposes, are placed at the periphery on the three sides of the core. The placement of the Be reflectors in the SAFARI-1 reactor core is shown in Fig 1. There are 19 Be reflector elements in total with the south and eastern side of the core completely occupied by the Be reflector elements (i.e. column 2 and row A in Fig 1). The western side is composed of the Be reflectors as well as aluminum filler elements. H8 is the only isolated reflector element.
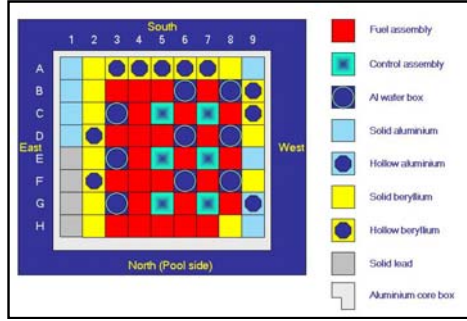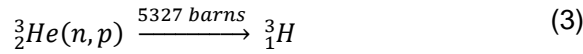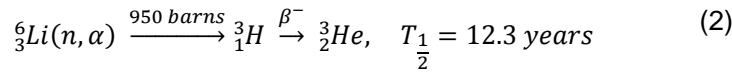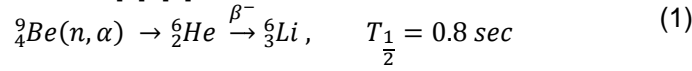
Fig 1. Placement of Be reflector in SAFARI-1 reactor

2.   Isotopic transmutation in a neutron irradiated beryllium reflector

In this study the two isotopes of interest are [6]Li and [3]He, the so-called "poisons" due to their large absorption cross section for thermal neutrons. The main chain reactions describing formation and build-up of these isotopes in Be due to fast neutron (i.e., 0.7 MeV – 10 MeV) irradiation are as follows [3], [4]:

$$^{9}_{4}Be(n,\alpha) \rightarrow {}^{6}_{2}He \xrightarrow{\beta^-} {}^{6}_{3}Li , \qquad T_{\frac{1}{2}} = 0.8 \; sec \tag{1}$$

$$^{6}_{3}Li(n,\alpha) \xrightarrow{950 \; barns} {}^{3}_{1}H \xrightarrow{\beta^-} {}^{3}_{2}He, \quad T_{\frac{1}{2}} = 12.3 \; years \tag{2}$$

$$^{3}_{2}He(n,p) \xrightarrow{5327 \; barns} {}^{3}_{1}H \tag{3}$$

In reaction (1) [6]He is transformed to [6]Li almost immediately (i.e., in 0.8 sec). [6]Li is burnt by thermal neutrons according to reaction (2) resulting in the formation of ^{3}H, which decays with a half-life of 12.3 years forming [3]He. The concentration of [3]He increases significantly during reactor shutdown periods and is converted back into ^{3}H according to reaction (3) when the reactor is back at full power. However, the concentration of [3]He and ^{3}H in Be increases linearly in time. In FISPACT [5] activation code, the number densities of poison isotopes are computed by solution of a set of differential equations describing the above-mentioned reactions, given by equations 4:

$$\frac{dN_i}{dt} = -N_i(\lambda_i + \sigma_i\phi) + \sum_{j\neq i} N_j(\lambda_{ij} + \sigma_{ij}\phi) \tag{4}$$

where N_i is the number density of nuclide i at time t; λ_i is the decay constant of nuclide i (s^{-1}); λ_{ij} is the decay constant of nuclide j producing i (s^{-1}); σ_i is the total cross section for reactions on i (cm^2); σ_{ij} is the reaction cross section for reactions on j producing i (cm^2) and ϕ is the neutron flux $(n.cm^{-2}.s^{-1})$.

3.   Computational methodology

The calculational methodology was as follows: the MCNP [6] SAFARI-1 reactor model was generated using OSMINT [7] program. The 172-group energy spectra for FISPACT runs is then calculated in each Be reflector element using MCNP. Next, the complete Be reflector material specifications, including impurities, was used in FISPACT to calculate the isotopic number densities of [6]Li and [3]He, as well as that of other secondary impurities, as a function of the detailed reactor power history (45 years). The computed number densities of the neutron poisons were then used in the OSCAR-3 cross section generation module to prepare the cross sections for the burnt Be reflector. Finally, the OSCAR-3 code was used to perform full core reactor calculations to quantify the impact of Be poisoning on core parameters. Effects on the general core parameters are evaluated against the reference, i.e., non-burnt, pure Be reflector. The results are also compared to experimental low power flux measurements.

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4.   Results and Discussion

OSCAR-3 calculations were conducted based on a representative fully HEU core and results presented in subsequent paragraphs.

4.1   Impact of impurities on Be reflector effectiveness and on core parameter

The total absorption in the Be reflector due to transmutation products, presented in terms of equivalent boron content (EBC), is increased from 2 *ppm* up to about 25 *ppm* after 45 years of irradiation (1965-2010), with poisoning contribution as follows: impurities: 6%; ^{6}Li: 48% and ^{3}He: 46% of the total neutron absorption. Initial impurities and the rest of the inventory, except for ^{6}Li and ^{3}He isotopes, contribute insignificantly to the total absorption in Be reflector and thus have negligible effects on core parameters.

4.2 ^{6}Li and ^{3}He buildup in Be reflector and impact on core parameters

The poison buildup in Be reflector up to the end of 2007 is presented below. Note that due to lack of cycle details as from 1965 until end of 2002, the spectrum was only updated as from 2003 onwards to improve the accuracy in poisons number density. Poison buildup based on the average spectrum as well as individual Be reflector elements spectrum is shown in figures below. Note that only the heavily (H8) and least poisoned (A2) Be reflector elements are shown.
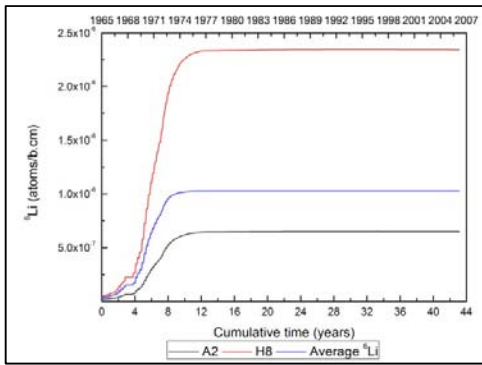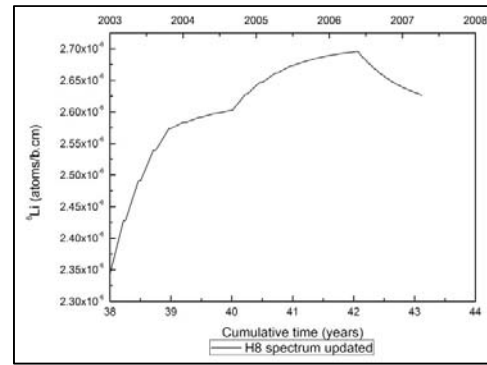

Fig 2. ^{6}Li build-up in Be reflector

Fig 4. Effects of spectrum updating on ^{6}Li
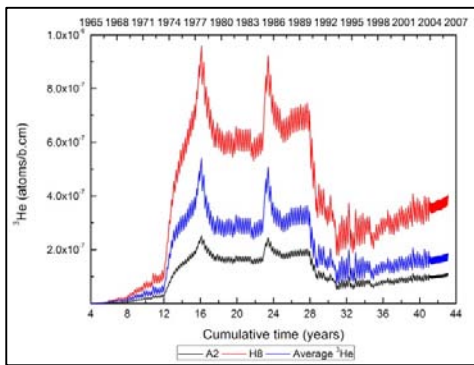

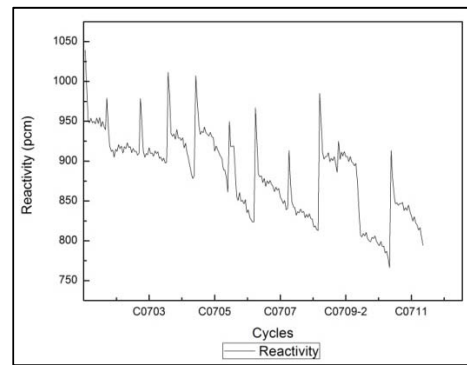Fig 3. ^{3}He build-up in Be reflector

Fig 5. Effects of poisons on reactivity

In Fig. 2 and 3 H8 is the highest curve (heavily poisoned), A2 is the lowest curve (least poisoned) element; the average lies between the two curves. The average poison representation in Be underestimate H8 and overestimate element A2 by a factor of approximately 2, but it represents all the other elements within a 30% band.

Furthermore, buildup of ^{6}Li in position H8 increases sharply due to its fast spectrum environment, i.e., situated between the control rod and two typically fresh fuel elements (see Fig. 1). A2 is in a more thermalised and a low flux level region (see Fig. 1), as a result the transmutation of poisons in this position is slower. In Fig. 3 it is clear that ^{3}He is very

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sensitive to reactor operation schedule, especially shutdown periods. The two peaks in this figure represent the reduction in power and a 6 month shutdown in 1981 and 1988 respectively. In Fig. 4 it is seen that with variation in spectrum (i.e., during the last 7 years) ^{6}Li number density does not reach equilibrium. Spectral changes lead to rather weak increase in ^{3}He. Fig. 4 shows the negative reactivity effects introduced on core by Be reflector poisoning over 11 cycles of 2007. The effects vary from 1000 *pcm* to about 750 *pcm* between the beginning and end of 2007.

Detailed poison distribution in each element is applied to a single cycle OSCAR-3 calculations and impact on core parameters evaluated against the reference, i.e., non-burnt, 100% pure Be reflector at BOC. The thermal fluxes redistribution is shown as percentage differences in the core map of Tab. 1 below.

	1	2	3	4	5	6	7	8	9
A	-22.7	-28.1	-21.1	-14.7	-16.8	-9.4	-7.6	-8.6	-8.1
B	-19.8	-22.7	-16.6	-13.4	-8.1	-4.6	-2.4	-3.2	-8.4
C	-15.3	-14.4	-9.0	-5.8	-2.4	-0.7	0.5	-0.8	-7.5
D	-12.5	-9.8	-3.7	-1.6	0.1	1.2	1.3	0.5	-5.9
E	-11.0	-10.4	-1.9	1.2	2.4	3.0	3.8	2.3	-2.0
F	-9.7	-9.8	-0.5	3.2	4.1	4.4	4.1	3.0	-2.1
G	-8.2	-8.0	1.2	4.1	5.8	5.4	4.9	3.2	0.3
H	-7.1	-7.5	2.2	5.1	5.5	5.8	3.9	-4.3	-1.7

Tab 1. Thermal flux redistribution due to the Be reflector poisoning at BOC $(\frac{\varphi_{poison}}{\varphi_{pure}} - 1)\%$

In Tab 1 it is seen that Be reflector poisoning impact largely on thermal fluxes of the reflector and fuel elements on the south-eastern side of the core. Note that both the southern and eastern sides are fully reflected by Be (see Fig 1). Element A2, i.e., the least poisoned element show a large reduction in thermal fluxes while H8, the heavily poisoned element, show a small reduction in thermal flux. This is attributed to the surrounding environment of these elements as explained before (see Fig 1). Fuel elements adjacent to the Be reflectors on the south-eastern side of the core experience a flux depression of up to -16.6% (i.e., B3), while that towards the northern side increase by up to 5.8%. The observations above are consistent with the fast flux and power redistribution in the core. In a poisoned core less power is delivered by the fuel elements as compared to the pure Be reflected core. In general, the tilt in flux and power redistribution due to Be reflector poisoning is towards the south-eastern side of the core, accompanied by peaking towards the centre and the northern side of the core. These observations tend to correct the historical tilt seen in OSCAR-3 to match that of the experimentally measured SAFARI-1 reactor. The results are quantified by comparison to experimental lower power flux measurement (LPM).

4.3   Comparison to experimental plant data
Low power flux measurements in all fuel elements are performed prior to the reactor startup of each cycle with the reactor conditions modeled as accurate as possible. Comparison to LPM based on poisoned and 100% pure Be reflected cores was conducted using the 1st cycle of June 2007. The comparison is shown in the core map below.

In Tab 2 the calculated flux mainly peaks in fuel elements adjacent to the pure Be reflectors and decrease towards the centre and the northern side (poolside) of the core. This is mainly due to modeling Be reflector as a pure, non-burnable material in OSCAR-3. The results of the poisoned Be reflector model are shown in the core map of Tab 3 below.

	1	2	3	4	5	6	7	8	9
A									
B			10.78	16.31	-2.74		8.13		
C				-0.04		4.34		7.15	
D			4.77	-10.24	-18.21		-3.05		
E				-15.66		-1.37		14.29	
F			3.75	-6.03	-13.57		-13.74		
G				-5.08		6.17		-0.60	
H			4.11	-8.62	-6.48	-6.20	-7.61		

Tab 2 LPM comparison for pure Be reflected core $\left(\frac{Calculated}{Experiment} - 1\right)\%$

	1	2	3	4	5	6	7	8	9
A									
B			-4.25	2.80	-9.48		5.69		
C				-5.10		3.09		6.21	
D			1.28	-12.06	-18.27		-2.95		
E				-15.26		0.76		16.05	
F			1.43	-4.05	-10.88		-11.82		
G				-2.25		10.83		0.44	
H			5.72	-4.96	-2.44	-1.54	-4.85		

Tab 3 LPM comparison for poisoned Be reflected core $\left(\frac{Calculated}{Experiment} - 1\right)\%$

It is seen that Be reflector poisoning decrease fluxes on the south-eastern side of the core adjacent to Be reflector elements accompanied by an increase towards the centre and northern side of the core. These observations are opposite to that of Tab 2 above. Comparisons on the eastern side of the core show improvements (see Tab 3). More work on code-to-experimental data comparisons needs to be performed to account for and to better understand the differences observed above.

Reload calculations, performed prior to BOC of each planned cycle using OSCAR-3, serves to evaluate the safety parameters of the planned cycle. The results of the calculation performed with and without poisons are summarized in Tab 4 below. The calculations were performed at cold, clean, Xe free core condition except for cycle length predictions.

Safety Parameters	Pure Be reflector	Poisoned Be reflector	SAFARI-1 reactor parameters
Planned cycle length (days)	30	30	-
Calculated Cycle length (days)	35	31	-
Excess reactivity ($)	9.4	8.6	-
Control rod worth ($)	32.9	34.6	≥ 20$
Shutdown margin ($)	20.6	23.2	>1/2 (rod worth)
Estimated start-up bank (cm) without Mo99 target plates	48.9	51.1	51.04 (at start-up)
Estimated start-up bank (cm) with Mo99 targets	46.1	47.2	> 39 cm

Tab 4 Predicted Safety parameters before start-up

It is seen in Tab 4 that Be poisoning influence the safety characteristics of core. The negative reactivity introduced by the poisons reduces the predicted cycle length by about 4 days which is compatible with real life scheduling. The rod worth, shutdown margins and the start-up bank reactivity is increased. There is good agreement between the poisoned core start-up banks (without targets) and measured bank position (i.e., plant bank position measured during the LPM experiment). This suggested that the Be poisoning impact positively on the general reactor safety parameter prediction. In general, modeling Be reflector burn-up shows promising results.

5.   Conclusions

It is concluded that the initial impurities present in the Be reflector (1965) have a negligible effect on core parameters as well as on the total thermal neutron absorption of the reflector. These trace impurities and their transmutation products remain in the reflector throughout its service lifetime. Conversely, the formation of ^{6}Li and ^{3}He in the Be during reactor operation showed important effects on the general core parameters as well as the effectiveness of the reflector. To improve the accuracy and reliability of the predictive OSCAR code calculations, Be burn-up should be incorporated in the next releases of OSCAR. Based on this study, the inclusion of Be burn-up chains is planned for implementation in the currently tested OSCAR-4 code system. Temporarily, details poison distribution in the OSCAR-3 code system will be accounted for using the calculational approach discussed herein. Furthermore, it is proposed that poison concentration be updated at BOC of each cycle for practical purposes. More work on comparison to experimental data as well as studies on swelling and embrittlement of the reflectors by ^{4}He is ongoing.

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INVESTIGATION OF FUEL DEFECTS IN THE NRU REACTOR

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ABSTRACT

The National Research Universal (NRU) reactor is a heavy-water-cooled and -moderated, tank-type, multipurpose reactor that operates at up to 135 MW. Originally designed for operation with natural uranium, NRU was converted to high-enriched uranium fuel in 1964. In 1991, AECL converted NRU to low-enriched uranium (LEU) fuel containing Al-61 wt% U_3Si, with U enriched to 19.75 wt% ^{235}U. Since the reactor was converted, the LEU silicide dispersion fuel performance has been excellent, with no reported fuel defects. However, in late 2008, monitoring showed an increasing trend in fission product activity in the D_2O coolant/moderator. Despite steps taken to increase monitoring and analysis, locating the source of the problem proved to be difficult. Fission product analysis and isotopic analysis of U extracted from the coolant indicated that the source was a small defect in the LEU driver fuel. This paper summarizes the investigation of the NRU fuel defect. Results are presented from post-irradiation examinations to determine the root cause of the failure mechanism.

1. Introduction

The NRU reactor was built for three purposes: to be a supplier of industrial and medical radioisotopes used for the diagnosis and treatment of life-threatening diseases; to be a major Canadian facility for neutron physics research; and to provide engineering research and development support for the Canadian nuclear industry. The core is contained in an aluminum cylindrical tank approximately 3.7 m high x 3.5 m diameter. It has 227 vertical lattice sites arranged in a hexagonal array with a pitch of 19.7 cm. Control rods and low enriched uranium fuel rods occupy about half of the lattice sites; the remaining sites are for low temperature/low-pressure experiments and irradiations. Two high-pressure/high-temperature experimental loops and six beam tube facilities are also available.

The reactor is re-fuelled at power. NRU fuel rods are 12-element assemblies containing Al-61 wt% U_3Si dispersion fuel sheathed with finned Al cladding welded to Al end plugs. The approximately 3 m long fuel elements are separated by six spacer grid plates, and are welded at the bottom to a support (hanger) plate that is in turn pinned and welded to an aluminum flow tube. A bulge end and plug end are welded to the flow tube to complete the NRU rod. The fuel rod maximum power is 2.1 MW.

The coolant flows upwards through the flow tubes, over the fuel and exits through an orifice section above the elements into the vessel. The coolant temperature rise and flow rate in each fuel rod is monitored. The D_2O inventory (about 65000 kg) is circulated at a nominal rate of 1950 kg/s with all pumps at full speed. Helium is used as an inert cover gas to minimize isotopic degradation of the coolant/moderator system. The cover gas is monitored for gaseous fission products and the D_2O is monitored for soluble fission products and activated corrosion products.

2. Investigation

In 2008 November, routine monitoring showed an increasing trend in fission product activity in the NRU coolant/moderator. A team of subject matter experts was assembled to investigate the cause. Detailed analysis of the radioactive fission products and U circulating in the D_2O indicated that there was a defect in one or more of the LEU driver fuel rods in the core (the results ruled out a target or fast neutron rod being the primary source). However,

locating the exact fuel rod(s) among the approximately 90 rods in core proved to be difficult. The large D_2O inventory and high recirculation rate causes any released uranium or fission products from a defect to be quickly dispersed in the coolant system where it remains until removed by IX purification, by the gibbsite evaporator, by fission or decay, or by deposition on in-core surfaces such as flow tubes and fuel rods that are in turn removed. Surveys of neutron and gamma fields at the top of reactor did not identify specific hot sites. The adjuster rods were used to tilt the power distribution in different sectors of the core in the hope that a power transient in a failed rod might provoke a detectable release, but results from this manoeuvre proved inconclusive. Subsequently, several rods were identified as "suspect" based on in-reactor residence time compared to the period that moderator activity was increasing. Factors such as high power operation late in life, high burnup, and manufacturing history were also considered. The selected rods were then scheduled for removal from the reactor on a priority basis.
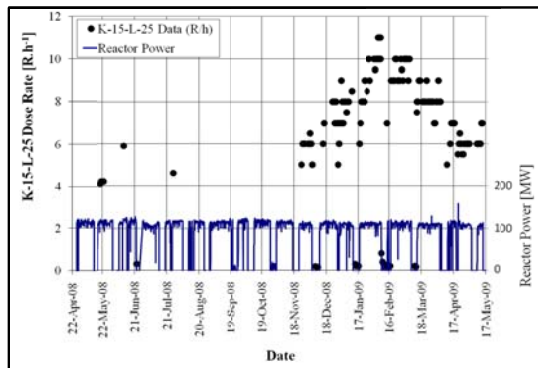

$_2O.$

By the end of February/early March after the prime suspects were removed, the moderator activity began to decline. The gross fission product activity trend during the period 2008 April to 2009 May is shown in Figure 1. The activity continued to decline until the reactor was shut down in May, indicating that the source was removed. However, when the rods were removed and stored there was insufficient evidence to conclusively identify which specific rod had failed, so further examinations were required to locate the defect.

3. Post-Irradiation Examination (PIE)

Initially, 10 of the discharged rods were selected for detailed underwater inspection. The flow tubes were cut at the hanger-plate weld location and the element assemblies were removed for inspection. After preliminary inspection of the visible surfaces of the outer fuel elements, the 12-element clusters were disassembled by cutting apart the grid plates, and the surface of each individual fuel element was inspected underwater. Of the 120 elements inspected, three were found with visible breaches in the aluminum cladding. Two defected elements were found in rod FL-1528 and one defected element was found in rod FL-1540. All three of the visible through-wall defects were located in the aluminum cladding over the end plug region near the closure weld where the cladding had bulged and split open to reveal the aluminum end plug beneath. Figure 2 shows a photograph of a typical defect.

Fuel sections ~ 150 mm long were cut out from the defected end region as well as from the middle and intact opposite end of each of the three elements. The PIE examinations in the hot cells included: visual examinations and photography; dimensional measurements; gamma scanning; optical metallographic examination; Scanning Electron Microscope (SEM) examination with Energy Dispersive X-ray (EDX) analysis and Wavelength Dispersive X-ray (WDX) analysis; and, U isotopic analysis for burnup measurements. The SEM exams were augmented with Secondary Ion Mass Spectrometry (SIMS) and X-ray Photoelectron Spectroscopy (XPS) examinations in an effort
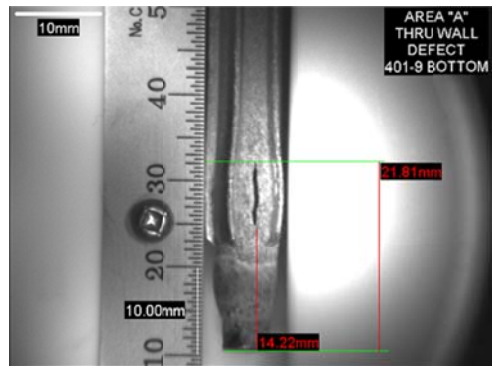
401-9.

to identify the chemical species in the samples.

4. Results

The visual examinations showed that except for the bulge defects, the external cladding surfaces appeared to be in good condition with no evidence of abnormal conditions such as overheating or excessive crud build-up. Dimensional measurements showed that cladding changes remained small over the fuel remote from the defect, while at the bulge/defect locations the residual cladding strain measured 12 - 20%. Gamma scanning of the stacked fuel core pieces showed that the fission product activity profile was consistent with the cosine flux shape in NRU, i.e., higher activity in the samples from the middle compared to the ends; however, the gamma scans showed no discernable evidence of missing fuel.

Metallographic examination of the primary defect locations show features that are consistent with internal pressurization of the cladding in the end plug region, deformation (bulging) and ductile rupture of the cladding. Figure 3 shows a cross section through the aluminum end plug and cladding at the bulge defect. The end plug is nominally ~90 mm long overall and the defect was over the plug near the weld, about 40-50 mm away from the fuel. The necked region between the fins is consistent with ductile failure under tension from hoop stresses in the cladding due to internal gas pressure. This common mode of failure was observed in all three elements and indicates a common root cause of the problem. SEM surveys revealed oxide/hydroxide layers of approximately the same thickness (~ 20 μm, see Figure 4) on the external surface of the cladding, on the internal surface of the cladding and on the end plug surface in the cavity. The nominally equal oxide thickness on both the internal and external surfaces suggests a similar duration of exposure to the coolant environment and indicates that the defect occurred early in the service life of the fuel. The surface of the split face was also corroded, consistent with a failure early rather than later in life.
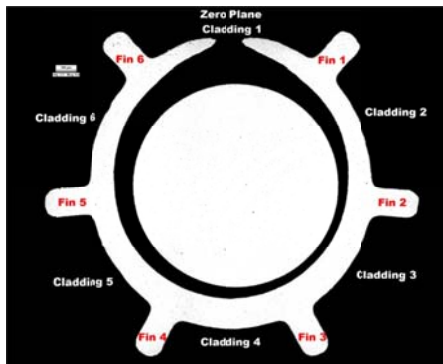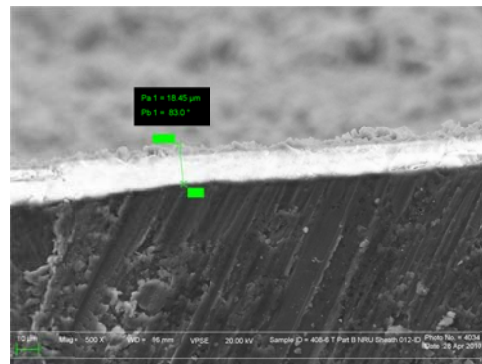

Fig 3. Cross Section Through Defect.

Fig 4. Surface Oxide Thickness from SEM Image.

The defect was located in the weld heat affected zone (HAZ) about 10 mm from the weld. Micro-hardness measurements showed that the weld HAZ has lower strength than the adjacent as-extruded cladding or weld parent material, which explains why the cladding bulged and split in this location. Metallographic examinations showed that the end plug welds were intact and not the source of a problem such as coolant ingress that could flash to steam.

SIMS and XPS examinations were performed in an effort to identify chemical species on the internal surfaces that may point to the source of the internal gas. Carbon and other impurities were detected but the results proved inconclusive.

Detailed optical metallographic examinations of the fuel revealed a common feature near each of the defects, namely, internal cracks/voids in the centre of the fuel core, with connected porosity providing a path leading to the end plug region. Figure 5 show the typical chevron-type cracks (arrowed) observed in the centre of the fuel core near the drilled hole that accommodates the end plug spigot. The end plug has a reduced diameter spigot that is

inserted in the hole, and is attached using a rolled joint to compress the fuel over the V groves in the spigot. The adjacent transverse section (Fig 5 (c)) shows the end plug spigot centered in the core; note, there is little evidence of coolant ingress or corrosion of the fuel annulus despite its proximity to the bulge defect. Two of the samples showed similar features with little evidence of corrosion but the third showed evidence of coolant reaching the fuel and causing a secondary defect due to oxidation and swelling of the fuel at the end plug joint. The cladding split where the fuel oxidized and swelled. It was estimated that about 0.6 g U was lost to the coolant from the three defects (based on metallography).
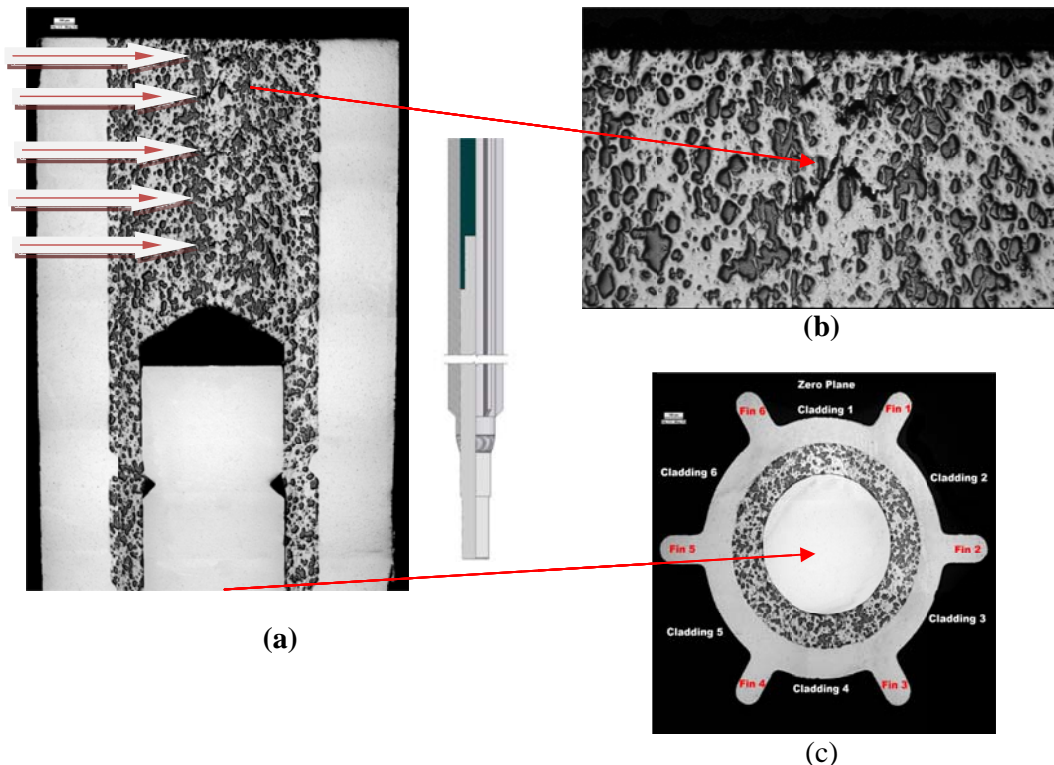

(a)   (b)   (c)

Fig 5. Micrographs of (a) fuel and end-plug spigot region at inlet (bottom) of element 401-9, (b) higher magnification image of chevron cracks in core, and (c) cross section through end plug spigot and fuel. Arrows point to chevron-type cracks in the extruded core.

The intact opposite ends of the three fuel elements were also examined. Two of the intact ends showed no centre cracks or connected porosity in the core and no evidence of internal gas pressurization over the end plug. The third sample showed less extensive centre cracks and porosity at the end of the fuel, and a discernable gap between the cladding and end plug consistent with internal pressurization. These results suggest there was sufficient pressure to separate the cladding from the end plug and to initiate sheath lift-off, but insufficient pressure to fully bulge and rupture the cladding at that end during the course of the irradiation. SEM exams with EDX and WDX analysis were used to check whether or not any residue of the species responsible for internal pressurization of the fuel could be unequivocally identified. Carbon was detected with all of the surface analytical methods used (as well as O, U, Si and trace impurities), but it could not be differentiated from the ever-present carbon found when analysing surfaces, and the results were inconclusive. Extended operation in the post-defect condition most likely obscured evidence of the environment at the time of the defect.

Fuel samples from the intact midplane regions showed the normal, stable NRU fuel microstructure with no evidence of anomalies or unusual features such as chevron cracks or porosity. The typical fuel interaction with the aluminum matrix was observed at the surface of the particles, and fission gas bubbles decorated the kernels. Measured reaction layer thickness ranged from ~15 μm at the centre of the core to ~7 μm at the periphery. The fuel

core diameter measurements showed, as expected, that swelling was higher in the mid-plane samples compared to the ends, as these regions operate at higher power and achieve higher burnup due to the cosine flux profile in the reactor core. The maximum mid-plane swelling, ~3%, is well within the previous experience base for NRU fuel [1] and shows that the fuel material was behaving satisfactorily in regions remote from the defect.

Burnup analysis was performed using measured $^{235}U/^{238}U$ and $^{236}U/^{235}U$ atom ratios. Conversion of the measured atom ratios to burnup were performed using the WIMS-AECL V3.10 code. The middle of the fuel rods achieved high burnup in the range 86-90 at% ^{235}U. The inlet ends achieved 59-64 at% burnup, and at the outlet ends 52-63 at% burnup. Results from U in the coolant were within the measured $^{235}U/^{238}U$ atom ratios.

5. Discussion

The literature [2] indicates that in forward extrusions, centre cracks and porosity are commonly observed where contamination and impurities are entrained in the material. The foreign material could not be conclusively identified during PIE, but is likely an excessive quantity of one of the organic compounds used in the manufacturing process. Likely contaminants include hydrocarbon materials used as parting/release agents to prevent the fuel material from sticking to the ram during core extrusion, rust inhibitors/oils used on the manufacturing equipment and cleaning solvents. A hydrocarbon contaminant would decompose during irradiation, producing gaseous species, including hydrogen gas, which would pressurize the interior of the fuel element.

Laboratory tests were performed on mini-elements that were fabricated with intentional additions of potential contaminants from the manufacturing process. Samples developed similar bulges with splits in the cladding over the end plugs after furnace heating for one hour at 500°C. Here a temperature in excess of that observed in NRU was applied as a rough approximation of radiolytic breakdown of the organics, which cannot be easily duplicated in the lab. Experiments were also performed to determine the pressure required to rupture the cladding. Samples were cut from the ends of archive full-length elements and internally pressurized in a test rig, with the cladding heated to 100°C to simulate the in-service temperature. The samples developed bulges in the weld HAZ that appeared almost identical to the NRU fuel defects, and ruptured at a pressure of ~12.4 MPa.

The amount of hydrocarbon material required to create a failure could be estimated from the size of the bulge defect and knowledge of the rupture pressure. Metallographic measurements of the three defects indicate the volume under the bulges ranged from ~89 to 106 mm^3. The maximum internal pressure, P, is given by

$$PV = nRT \qquad (1)$$

where V is the volume, n is the number of moles of gas, R is the gas constant and T is the temperature at the defect. Assuming the NRU defect occurred under conditions similar to those in the lab tests, i.e., P = 12.4 MPa, V= 0.106 cm^3, T = 373 K and R = 8.314, then

$$n = PV/RT = 12.4 \times 0.106/(8.314 \times 373) = 0.0004 \text{ moles of gas.}$$

If the pressurizing gas is H_2, then 0.0004 mol x 2 g/mol = 0.8 mg H_2 is required. For example, if the source is methanol, then 12.8 mg CH_3OH is required. It is more difficult to calculate the quantity of materials such as release agents that are mixtures of high molecular weight compounds.

The connected porosity provides a leak path from the centre cracks to the end of the fuel so that gaseous species released from the radiolytic breakdown of organic contaminants during irradiation could migrate and pressurize the end-plug region of the fuel element. If sufficient internal pressure is generated, the cladding will rupture at its weakest point, near the weld heat affected zone, and lead to element failure. Below this pressure threshold, the cladding may bulge but not fail.

Since NRU was converted from HEU to LEU fuel in the early 1990's, there have been no reported driver fuel defects. During the LEU fuel development program, mini-elements with holes drilled in the cladding were irradiated at linear power up to 87 kW/m in NRU to assess the fuel defect behaviour [3]. PIE results showed small cavities developed beneath the holes but very little fuel was lost to the coolant (from 1 mg ^{235}U after 29 days, to between 9 and 48 mg ^{235}U after 98 days of irradiation), confirming the good corrosion resistance of material. Subsequent experiments also showed that prior irradiation further enhanced the resistance to corrosion/erosion.

There appears to be little additional information in the literature on the defect behaviour of LEU silicide dispersion fuel under research reactor operating conditions. During initial development and irradiation testing in the ORR reactor, plate-type fuels containing U_3Si and U_3Si_2 with cladding defects suffered extensive corrosion during irradiation [4, 5]. In the case of the Al-U_3Si fuel plates irradiated in ORR, no evidence of the failure was detected, indicating the amount of fission product release was small. It was concluded that hydrolysis of water during irradiation liberated oxygen, which reacted with the fuel. The results show the LEU silicide fuel is corrosion resistant and that cladding breaches are not likely to result in catastrophic failure.

6. Summary and Conclusions

Post irradiation examinations were performed on three defected fuel elements from the NRU reactor. The observed defects were bulges with axial splits in the cladding in the end-plug weld heat affected zone. Chevron-type flaws were observed in the centre of Al-U_3Si fuel cores near the defected ends of the elements. These features are observed when impurities are present during extrusion. The impurity was not conclusively identified during PIE but is likely foreign material (organic contamination) that was introduced inadvertently in the manufacturing process. During irradiation, gas released from the radiolytic breakdown of organic contaminants would pressurize the interior of the fuel element, and the cladding will rupture at its weakest point in the weld HAZ if sufficient pressure is generated.

The oxides/hydroxides on the internal surfaces of the defected elements indicate that the fuel failed early in its operational life. Despite a considerable period of post-defect operation, only a small quantity of fuel material was lost to the coolant. This confirms the good corrosion resistance of the Al-U_3Si dispersion fuel, and the robustness of the NRU fuel design.

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MANAGING PREVENTIVE MAINTENANCE, IN-SERVICE INSPECTION, AGEING MANAGEMENT WITHIN A PRODUCTION ENVIROMENT

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ABSTRACT

SAFARI-1, the Nuclear Research Reactor located at Pelindaba, South Africa, has to meet a multitude of requirements relating to corporate policies and strategies, preventative maintenance, in-service inspection, plant ageing management, quality management, conventional and radiological safety, regulatory compliance; and security, commercial and financial goals which need to be incorporated into an integrated management system. The control of these various requirements and activities within the nuclear research reactor organisation becomes quite complex if the processes and systems are to be integrated maintained and at the same time managed so as to achieve the objectives and targets within the resources available. Plant maintenance is one aspect which became a challenge over the last decade to implement and optimize. This has been evidenced by the shorter reactor shut downs which are carefully planned to maintain operational schedules required for production and other interested parties.

Although SAFARI-1 maintains ISO 9001 and ISO 14001 certification, its present commercial and operational schedules need a very well coordinated management system to achieve all stakeholder requirements and still function within the design and safety requirements. In order for SAFARI-1 to maintain an exceptionally high operational level of more than 300 full power days per year, the planning, implementation and coordination of routine maintenance, in-service inspection, and projects related to upgrades and ageing management need to be well structured and controlled. The effective and efficient implementation of such a management system will ultimately ensure the success of the reactor in the competitive environment. This becomes particularly relevant where the development and the responsibility of implementation of systems and processes lie not only within the SAFARI-1 organization but also within various other departments in the parent South African Nuclear Energy Corporation (Necsa) organization.

The complexities of resources and processes implemented to achieve success with the various projects, processes and systems will be discussed and examples will be given of experiences during the last 10 years

1.   INTRODUCTION

The SAFARI-1 Research Reactor was designed, constructed, commissioned and licensed and has been operational since March 1965. Ongoing programmes of development, improvement through preventative maintenance, upgrading projects and ageing management have characterized this past operation and are expected to continue for the remaining life time of the facility. The aim of this all is to ensure the safe operation of the facility, reliability and life time extension to 2030.

Safety is considered by management to be the most important issue with respect to employees, public, operation and utilization of the reactor. The requirements for conformity are set out in an integrated Quality Health Safety Environmental management system (QHSE) and therefore are binding on all personnel and levels of management within SAFARI-1. Amongst others, effective maintenance, ageing management and in-service inspection are some of the most important activities to achieve safety and maintain the intent of the design objectives during the operation of the facility.

2.   QUALITY HEALTH SAFETY MANAGEMENT SYSTEM

The SAFARI-1 QHSE management system together with Necsa's corporative SHEQ management system defines the organizational structure, responsibilities of management and personnel, , quality, health, safety and environmental management, project management, reactor operations, reactor engineering, training and authorization as well as the utilization of SAFARI-1 Research Reactor. The SAFARI-1 management, now reports NTP PTY (Ltd) a wholly owned subsidiary of Necsa, has the primary responsibility for the safe operation of the facility and providing the finances for operation, utilization, maintenance and ageing management projects.. To discharge this responsibility, senior management has implemented a new organizational structure supported by an integrated QHSE management system which:

- Ensures that safety matters are given the highest priority;
- Ensures that established safety policies of Necsa and the regulator are adhered to;
- Provides a clear definition of responsibilities and accountabilities with corresponding lines of authority, accountability and communication;
- Ensures that sufficient staff are employed in engineering, nuclear safety and project management that are experienced and competent to carry out safety-important work according to their appropriate education, training and skills;
- Controls sound project management processes,, good design and engineering support, good quality engineering and work documentation based on good safety practices to ensure that projects related to plant upgrades/modifications are strictly adhered to also ensure success of implementation;
- Reviews budgets to provide for financial support of all projects according to priorities allocated; and
- Reviews priorities and cash flow quarterly;
- All safety matters are monitored and regularly verified through inspections and audits.   Any audit findings and non-compliances are formally controlled through action plans giving corrective measures which are followed up until closed out.

3.   SAFARI-1 OPERATIONAL HISTORY

The operating history (see also **Figure 1** below) of SAFARI-1 is summarized below. By March 2011 the reactor had been operational for 46 years and had produced a total of more than 3000 GWh of thermal energy. SAFARI-1 produced the 1st 1000 GWh within 30 years, the 2nd 1000 GWh within 8 years and the 3rd 1000 GWh within 7 years. The last 15 years since 1996 required a high utilization for commercial purposes. The main features are briefly discussed below:

- 1965: First criticality in March 1965 - after which the reactor operated at 6.67 MW thermal power using HEU fuel of US origin with an enrichment of 93%.
- 1968: The reactor was shut down for approximately 9 months to upgrade the heat removal systems for 20 MW operation.
- 1977: Due to political pressure the US fuel supply to SAFARI-1 was stopped, prompting a reduction in reactor operation to 5 MW and during weekdays only.
- 1981: Locally manufactured Medium Enriched Uranium (45% Enriched) fuel was supplied for the first time. The reactor continued to be operated at 5 MW on a weekday schedule for the next 12 years.

- 1988: The reactor was shut down for an extended period of 6 months to effect repairs to the pool liners.
- 1993: The start of a commercially oriented utilization programme led, for the first time in 16 years to a progressive increase in reactor power higher than 5 MW. Initially at 10 MW but between 1995 and 2000 the power was further increased to 18 MW average, and ultimately to 20 MW over the last 10 years.
- 1994: Enrichment of locally produced fuel elements was changed from MEU to HEU, with a ^{235}U content of 200g.
- 1995: One million MWh (1000GWh) total energy production since first start up.
- 2000: The ^{235}U content in locally manufactured HEU fuel elements was increased from 200g to 300g.
- 2003: Two million MWh (2000 GWh) total energy production since first start up.
- 2007: The first two LEU lead test fuel elements (340g of 19.75% enriched ^{235}U) were successfully irradiated in the core.
- 2008: Core conversion to LEU started.
- 2009: LEU conversion of the core was completed and regulatory approval for continuous loading of LEU fuel was obtained in August 2009.
- 2009: Three million MWh (3000 GWh) total energy production since first start up.
- 2010: SAFARI-1 staff and operation responsibilities were seconded to Nuclear Technology Products PTY (Ltd).
- 2010: Regulatory approval for commercial production of ^{99}Mo from LEU target plates was granted.
- 2010: Currently on 3.038 million MWh total energy production since first start up.

Table 1: SAFARI-1 AVAILABILITY

Year	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010
a) Available time [hrs]	8760	8760	8760	8784	8760	8760	8760	8784	8760	8760
b) Operational time [hrs] (a-c-d)	7608	7446	7541	7392	7501	6982	7167	7314	7222	7361
c) Scheduled downtime [hrs]	1089	1278	1176	1279	1214	1329	1321	1331	1499	1499
d) Unscheduled downtime [hrs] (e+f)	63	36	43	113	44	449	271	115	35	6
e) Plant unreliability [hrs]	33	11	29	25	2	349*	239*	36**	0.9	0.3
f) Beyond plant control [hrs]	29	25	14	87***	42	99	32	78**	34	6
g) Load factor [%] (b/a)	87	85	86	84	86	80	82	83	82	84
h) Loss in load factor due to plant unreliability [%] (e/a)	0.38	0.13	0.33	0.29	0.03	3.98	2.73	0.42	0.01	0.003

* Failure of deminerilizer anion column bottom sieve and resin released into primary system.
** Two unreliable channels of the six safety channels caused sporadically reactor scrams.
*** Failure of Primary Conical Strainer

From **Table 1**, since 1996 the reactor power levels were progressively increased from an average of 16 MW to 20 MW continuously and operation for the last nine years has averaged out at ~303 days a year at 20MW. Operating cycles are ~35 days (including a 5 day maintenance shut down), and one extended 12-day shutdown per annum. SAFARI-1 has a very good performance and reliable operational record. See **Table 2** below for data of the last four years.

Table 2

Operation	Year			
	2007	**2008**	**2009**	**2010**
Planned operation (days)	302.5	303.5	302.5	302.5
Actual operation (days)	306.1	303.2	301.0	306.6
Availability (%)	100.5	99.9	99.5	101.4

Figure 1
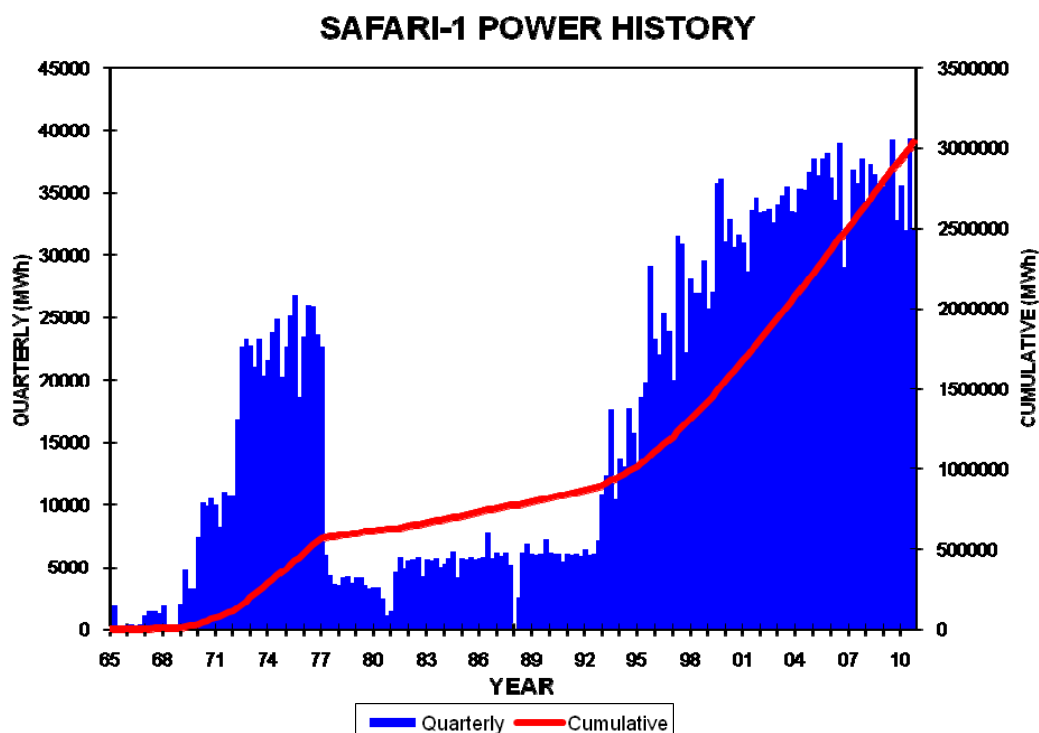
4.   MAINTENANCE SYSTEM

The SAFARI-1 research reactor maintenance programmes were adapted since 1996 (with more emphasis on maintenance and personnel competency) to ensure that systems, structures and components (SSC) continue to operate as required and to meet the design and safety objectives. The maintenance procedures form an integral part of the QHSE management system and prescribes the principles and controls established for periodic inspections and maintenance within the SAFARI-1 facility. This maintenance programme ensures that plant equipment and related nuclear safety and safety critical equipment or systems, are correctly inspected and maintained in accordance with ISO 9001 and IAEA safety guidelines (SS 35 G7, SS 50C/SG - Q13 and NS-G-4.2) and regulatory requirements.

4.1   MAINTENANCE PROGRAMMES

The SAFARI-1 Research Reactor has, since initial operation in 1965, applied a management system which was primarily focused on the technical design and safe operation of the plant. The informal system was never developed to comply with a specific code of practice. Since the early 1990's the challenge was to return to the international arena and to adjust the paradigm of employees from a Monday to Friday research culture into a more complex commercial culture, operating the plant 24 hours a day, 7 days a week, at a fixed operating schedule so as to meet customer and stakeholder

requirements. All maintenance procedures/activities are compliant with the Operating Technical Specification (OTS) and Safety Analysis Report requirements. Operating requirements of the OTS are vital and will not be compromised as a result of maintenance activities.

One of the areas was to develop, prepare and maintain maintenance programmes requiring maintenance at regular intervals in order to reduce the probability of failure and to suit the commercial programmes. Since 1997 the maintenance programmes for instrumentation, electrical, mechanical and reactor specific activities were optimised according to experience gained on SSC's:

- Routine maintenance (Preventative Maintenance).
- Periodic inspections (Preventative Maintenance).
- Ad-hoc maintenance – Request for Maintenance database (Corrective Maintenance).
- Functional inspection (Preventative Maintenance).
- Performance and functional tests (Predictive maintenance).
- Operational checks and maintenance (Reactor pools and hot cells).
- In-Service Inspection (ISI) procedure and ISI plan.
- Training and authorization of maintenance personnel.
- Control of equipment and spares.
- Project management and design control.
- Procurement control and release of items.
- Work permits (Radiological and conventional safety and Authorization of personnel).
- Record keeping of suppliers, SSC data and project file containing all information.

The maintenance programme lists responsibilities, frequencies, schedules (i.e. daily, weekly, monthly, yearly and more than one year), criteria and controls for the maintenance operations, identify systems that can be isolated while the reactor is in operation, applicable restrictions for on line maintenance and records to be kept.

4.2 MAINTENANCE SCHEDULES AND SHUTDOWNS

Only approved project and related work activities for the shutdown will be taken into account in planning maintenance activities, due cognizance is taken of the reactor operational programme, maintenance programmes, ISI plan as well as the nuclear safety aspects pertaining to the maintenance work. A pre-maintenance shutdown meeting is held, where after (13 times per year) a Maintenance Shutdown Plan is issued detailing all maintenance schedules and ad-hoc inspections or testing to be performed during the specific shutdown. After maintenance has been completed, reactor operations evaluates all work performed on SSC's, tests and/or inspections that were performed to ensure safe operation thereof before accepting the plant for operational purposes.

During the post shutdown meeting the Reactor Manager, Engineering Manager and relevant support managers/personnel, as applicable, jointly review all reports (including ISI reports) after each planned maintenance shutdown to verify that the plant is being correctly inspected and maintained in accordance with the OLC, OTS and maintenance schedules.

On a regular basis the regulatory body reviews and assesses documents/reports relating to maintenance and periodic inspections/testing to ensure adequate control of maintenance activities.

Maintenance Activity	No. Routine Work/Year on Equipment	No. Shutdown Work / Tasks	No. Ad-hoc Work /Shutdown	No. ISI Work
Mechanical	1082	152	242	22
Electrical	278	127	86	0
Instrumentation	559	293	106	0
Total	1919	572	434	22

5.   AGEING MANAGEMENT

5.1   Ageing Management Plan
The plan for the implementation of the Ageing Management Programme was drawn up, based on IAEA Guideline document **SSG-10** and further revisions thereof. The plan included the methodology for identifying, assessing, prioritising and addressing plant ageing issues.

5.2   Ageing Management Assessment
A series of IAEA Workshops on Ageing Management during 2009 and 2010 provided a valuable tool in the form of what is referred to as an Ageing Matrix. The matrix lists all SSCs typically found in Research Reactors in the vertical axis and provide a comprehensive list of ageing mechanisms (see section 5.7) in the horizontal axis, allowing the user to map out a plant-specific set of ageing issues in order to plan actions, budget and prioritize. The Ageing Matrix was used to draw up an Ageing Management Assessment of the SAFARI-1 Reactor that formed the basis for implementing an Ageing Management Programme (AMP) at the facility.

5.3   Ageing Management Projects
Any Ageing Management Projects that might arise from the above assessment are dealt with as follows:
- A prioritisation methodology has been developed, and included in the Ageing Management Plan, against which prioritisation of projects is done.
- Approval of Ageing Management projects are done based on processes prescribed in Necsa corporative SHEQ procedures, SAFARI-1 procedures and regulatory documents.
- All Ageing Management projects shall be managed as per a project management procedures and SHEQ processes prescribed on the categorisation for level of SSC's, safety classification and quality class level to establish the level of control and involvement of all parties.

5.4   Licensing
Where Ageing Management projects affect a safety critical or safety-important item, or otherwise affect the operating licence, the regulator forms part of the approval process for the project. In any event, the regulator is fully informed of all actions and a project implemented under the Ageing Management Programme, irrespective of whether they require regulatory approval or not, as the Ageing Management Projects enjoys a high profile status with all stakeholders and customers.

5.5   Ageing Evaluation Task Group
An ageing evaluation task group, headed by the Strategic Projects Manager was established to identify and evaluate SSCs affected by ageing, and implementing the resulting Ageing Management Projects. The task group consists of selected top management and section heads in the SAFARI-1 structure and may invite any other suitably qualified and experienced person/s to assist in the evaluations as required from support groups within Necsa licensing and SHEQ Departments

5.6   Grouping of Structures, Systems and Components (SSC)
Ageing Management Assessment specific to the SAFARI-1 facility is based on IAEA guidelines, which were adjusted to fit the type of reactor and its structures, systems and components and were grouped into:
- Reactor block, fuel and internals;
- Cooling systems;
- Confinement and containment;
- Instrumentation and controls;
- Power supply;
- Auxiliaries (e.g. fire protection, crane, hot cells and radioactive waste handling);
- Experimental facilities;
- Non SSC;

- ▪ Documentation (e.g. SAR, OTS, design and management system); and
- ▪ Staff training.

5.7   Clarification of Ageing Mechanisms

The following ageing mechanisms described were considered for evaluation, rating and allocating risk factors:

- ➢ Radiation- change of properties,
- ➢ Temperature - change of properties
- ➢ Creep due to stress/pressure
- ➢ Mechanical displacement/fatigue/wear from vibration, cyclic loads
- ➢ corrosion,
- ➢ Material deposition (e.g. crud)
- ➢ Flow induced erosion – (e.g. orifice concrete)
- ➢ Obsolescence -technology change damage due to power excursion and events
- ➢ Flooding - deposition; chemical contamination
- ➢ Fire - effects of heat, smoke, reactive gases
- ➢ Changes in requirements legislation or acceptable standards
- ➢ Other (time dependent phenomenon)

5.8   Remedial Action and Prioritisation

A simple mathematical model was developed to aid in the objective prioritisation of remedial actions and was applied throughout. The methodology followed was to allocate a score to each remedial action for each impact factor. The sum of these scores is then adjusted (multiplied) by a supplementary "Weighting Factor" on a scale of 1 to 10.

For each ageing mechanism applicable to a SSC identify a corrective action (e.g. replace, refurbish, redesign or maintenance action). The following number of projects was identified for further evaluation and or implementation:

- ❖ Ageing management projects - total 18 (e.g. reactor vessel life time, gamma channel, PLC systems, ventilation control systems, overhead crane and refurbish electrical equipment)
- ❖ Development and upgrade projects (e.g. non-AMP) - total 38 (e.g. Upgrade SAR, replacement of beryllium reflector elements, control room annunciator system, primary temperature and pressure instrumentation, strainer, pool structure, core grid plate, waste and storage effluent tanks, further development of data logging systems, QHSE system and safety culture)
- ❖ Upgrade and maintenance projects – total 24 (e.g. beam lines SANS, NDIFF, NRAD, foyer extension, workshop extension, cooling tower PLC, Mo rigs, silicon irradiation facility, demineralised systems, replacement electrical transformers, replacement of in-core components)

5.   CONCLUSION

An integrated QHSE management system complying with various requirements, codes and standards (corporate, regulatory and international) is needed to manage a nuclear facility safely and successfully. The integrated QHSE management system must be developed in conjunction with other management systems and should be effectively implemented to support the engineering, maintenance, and operational aspects to ensure that operational schedules are maintained. One of the most important objectives of a nuclear reactor is "Reactor Safety" and this can only be accomplished by having effectively implemented design control and project management processes, maintenance programmes, ageing management systems and training programmes for personnel to execute all activities.

References

This document is based on and includes concepts as described in:

[1] SSG-10:   Specific Safety Guide "Ageing Management for Research Reactors", IAEA, October 2010.

A HIGH OPERABILITY SUPERVISORY DIGITAL SYSTEM FOR TRIGA-TYPE RESEARCH REACTORS

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ABSTRACT

In this work, we propose an outline of a monitoring system to supervise variables coming from a fission nuclear reactor of TRIGA type (1-MW TRIGA reactor RC-1). The system can interface the control room instrumentation and can display the characteristic parameters (e.g. nuclear power, temperatures, flow rates, radiological parameters) in an intuitive, user-friendly way for plant operators. This aim is achieved using the Labview development environment. A front panel of a virtual instrument allows for a direct measure and a check that would not be possible by only reading the output data coming from the instruments of the control room, because of their standards and strict safety regulations. The acquisition system, for signals coming from the reactor, can process data and generate a detailed representation of the results. Statistics resulting from data analysis will be interpreted to optimize reactor management parameters. This system also includes a simulation tool to predict specific performances and investigate critical phenomena, or to optimize overall plant performances. In particular, it allows to have a feedback control and to perform predictive statistical surveys of all main process parameters.

1. Introduction

In a Nuclear Power Plant (NPP) the problem of supervising and control is crucial for enhancing all monitoring and operation tasks. In the last years, supervisory systems tend to become more and more totally digital, thus providing more advanced operational and maintenance functionalities, together with a greater cost-efficiency. As a matter of fact, a lot of working reactors are partially or totally supervised and controlled by analogue systems. In order to improve the Human-Computer Interface (HCI) for the so-called "old generation" plants, it could be worthwhile to introduce simple supervisory systems based on general-purpose technology but able to satisfy the reliability requirements demanded by NPP equipment. The simplest way to test such a system is in the context of a research reactor, due to its inherent capability of mimic a real nuclear power plant and to this purpose designed for use by scientific institutions and universities.The purpose of this paper is to present a tool for monitoring and supervising variables coming from a fission nuclear reactor. This supervisory system has been tested and validated on the basis of the control system for the 1-MW TRIGA reactor RC-1 at the ENEA, Casaccia Research Center. In Section 2 we present a description of the reactor, with a particular focus on the Control andInstrumentation aspects. In Section 3 we describe how the supervising tool has been designed and implemented. The potentiality of the simulation tool and some preliminary results of the validation process on TRIGA reactor control room are presented in Section 4. Finally, we discuss results and open problems in the conclusion section.

2. TRIGA reactor RC-1

TRIGA RC-1 is a pool thermal reactor having a core contained in an aluminium vessel and placed inside a cylindrical graphite reflector, bounded with lead shielding [1], [2]. The biological shield is provided by concrete having mean thickness of 2.2 m. Demineralized water, filling the vessel, ensures the functions of neutron moderator, cooling mean and first biological shield.

Reactor control is ensured by four rods: two shims, one safety fuel-follower rods and one regulation rod.

Produced thermal power is removed by natural water circulation through a suitable thermo-hydraulic loop including heat exchangers and cooling towers. In Tab 1 some irradiation facilities are listed.

Description	Neutron flux [n/cm^2/s]
Rotating rack (40 positions)	$2.0 \cdot 10^{12}$
Pneumatic transfer system	$1.25 \cdot 10^{13}$
Central channel	$2.68 \cdot 10^{13}$
Thermal column collimator	$\sim 1.0 \cdot 10^6$
Tangential piercing channel (w/o) collimator	$\sim 1.0 \cdot 10^8$

Tab 1: Irradiation facilities

The core and the reflector assemblies are located at the bottom of an aluminium tank (190.5 cm diameter). The overall height of the tank is about 7 m, therefore the core is shielded by about 6 m of water. The core, surrounded by the graphite reflector, consists of a lattice of fuel elements, graphite dummy elements, control and regulation rods. There are 127 channels divided in seven concentric rings (from 1 to 36 channels per ring). The channels are loaded with fuel rods, graphite dummies and regulation and control rods depending on the power level required. One channel houses the start-up Am-Be source, while two fixed channels (the central one and a peripheral) are available for irradiation or experiments. A pneumatic transfer system allows fast transfer from the peripheral irradiation channel and the radiochemistry end station.

The diameter of the core is about 56.5 cm while the height is 72 cm. Neutron reflection is provided by graphite contained in an aluminium container, surrounded by 5 cm of lead acting as a thermal shield. The fuel elements consist of a stainless steel clad (AISI-304, 0.05 cm thick, 7.5 g/cm^3 density) characterized by an external diameter of 3.73 cm and a total height of 72 cm end cap included. The fuel is a cylinder (38.1 cm high, 3.63 cm in diameter, 5.8 g/cm^3 of density) of a ternary alloy uranium-zirconium-hydrogen (H-to-Zr atom ratio is 1.7 to 1; the uranium, enriched to 20% in ^{235}U, makes up 8.5% of the mixture by weight: the total uranium content of a rod is 190.4 g, of which 37.7 g is fissile) with a metallic zirconium rod inside (38.1 cm high, 0.5 cm in diameter, 6.49 g/cm^3 of density). There are two graphite cylinders (8.7 cm high, 3.63 cm in diameter, 2.25 g/cm^3 of density) at the top and bottom of the fuel rod. Externally two end-fittings are present in order to allow the remote movements and the correct locking to the grid.The regulation rod has the same morphological aspect as the fuel rod: the only difference is that instead of the mixture of the ternary alloy Uranium-Zirconium-Hydrogen there is the absorber (graphite with powdered boron carbide). The control and safety rods are "fuel followed": the geometry is similar to that of the regulation rod but with fuel element at its bottom. The graphite dummies are similar to a fuel element but the cladding is filled with graphite. Fig 1 shows an horizontal section of the reactor.
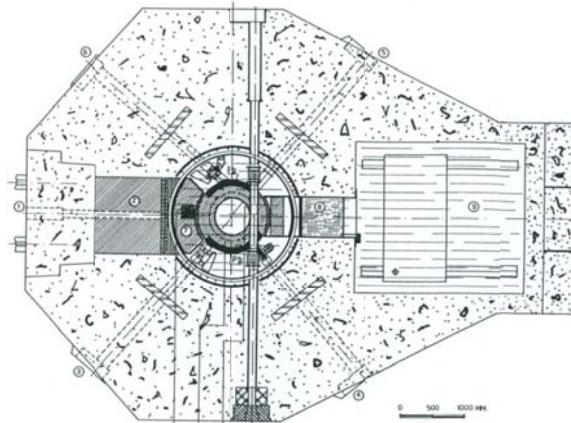
Fig 1: TRIGA RC-1 horizontal view

The parameters used in order to perform the reactor monitoring can be classified into three large groups: power monitoring, process monitoring and radiological monitoring.The reactor power is monitored by means of one starting channel (0.0 to 1 W), two wide range linear channels (0.5 - $5.0 \cdot 10^6$ W) and one safety channel (10 kW to 1.1 MW). The process monitoring includes temperatures (fuel elements, primary and secondary loops, cooling towers), flow rates (primary and secondary loops, water cleaning system, reactor hall air), levels (reactor pool, shielding tank), conductivities (primary loop, shielding tank loop).The radiological control is carried out by monitoring water activity (primary and secondary loops), air activity (reactor hall and experimental channels) and environmental radiation levels (reactor hall, control room and experimental channels).Only main plant parameters are mentioned herein, but a lot of secondary parameters can be easily monitored (such as control rods positions, switches status, alarms and so on) [3].

3. A flexible tool for supervising control room parameters

In order to design a full digital supervisor system, it is a good rule choosing a platform able to guarantee both modularity and flexibility, so as to provide the capability to interface all kind of instruments of the control room. Such goal is obtained using National Instruments Labview as software environment.
Labview is a graphic programming language that uses icons instead of text lines to create a specific application [4]. It is based on data flow, which determines the execution of a program. In the Labview environment it is possible to implement a user interface (Front Panel) containing different types of virtual elements. Code is added making use of the graphic representation of functions to control Front Panel objects, and it is contained into block diagrams. Labview is completely integrated to communicate with GPIB, VXI, PXI, RS-232, RS-485 hardware devices and DAQ plug-in.
Labview programs, called Virtual Instruments (VI), consists of three fundamental parts: the front panel, the block diagram and an icon with connectors box.
The front panel has controls (e.g. potentiometers, buttons and other devices), and indicators (graphics, LED, and so on) that are interactive input and output terminals.
Controls simulate output of instruments and provide data to the block diagram. Indicators simulate output devices and display data acquired or generated by the block diagram. The power of Labview is its hierarchical nature of VIs: once a VI has been created, it can be used elsewhere as a SubVI, thus creating a powerful network of interacting tools. In the prototype we propose here, the instrumentation is constituted by two National Instruments devices DAQPad 6020E with 16 analog channels and 8 digital ones and two signal adaptors SCB-68, one for each DAQ connected with the console. It receives signals from the selected devices. Every electrical connection ensures a high level of signal isolation. It is fundamental to notice that the new system doesn't interfere with the ordinary control room measurement and

control devices.Typically, each device measures a potential or a current and shows the value of the corresponding physical quantity: the temperature of the reactor pool water so as the γ–ray exposition in reactor room.

Our virtual system is able to have an interface with three possible implementations: reading from a log file of a variable (pool temperature, for example), reading form simulation signal or sampling from a physical channel. We can choose to switch between one of these mode with a control on the front panel of the application.The present work is based on the definition of a LabView object for each instrument of the console. The basic programming structure is typically composed of: an hardware interface, for signals acquisition; an applicative core, for signal elaboration; a user interface, for displaying the results. Such pattern of development is applied to virtual instrumentations, each one running independently from the others. The independence of virtual signals reproduces the independence of real signals in the control room.The VI's output can be compared with the analog output, which is always displayed on the console of control room. The system is made in such a way that independence of applicative core and front end from the implementation of the input level can be achieved. Other characteristics, such as channels interdependency, are implemented in the system and are available in simulation or real-time mode. The application is composed by five tabs: in the first one, it is possible to configure the system function modes, choosing between simulation, log file reading or real-time measurement; in the second one,synoptic scheme is shown containing the current value for control rods displacement together with a visual indicator of the current rod working; the third one contains all nuclear variables indicating the actual power reached by the plant (starting channel, two linear channels, logarithmic and safety channels); the fourth tab contains an indicator for conventional variables (temperature and flows); the last tab displays radiological data measured during reactor operations (Fig.2).
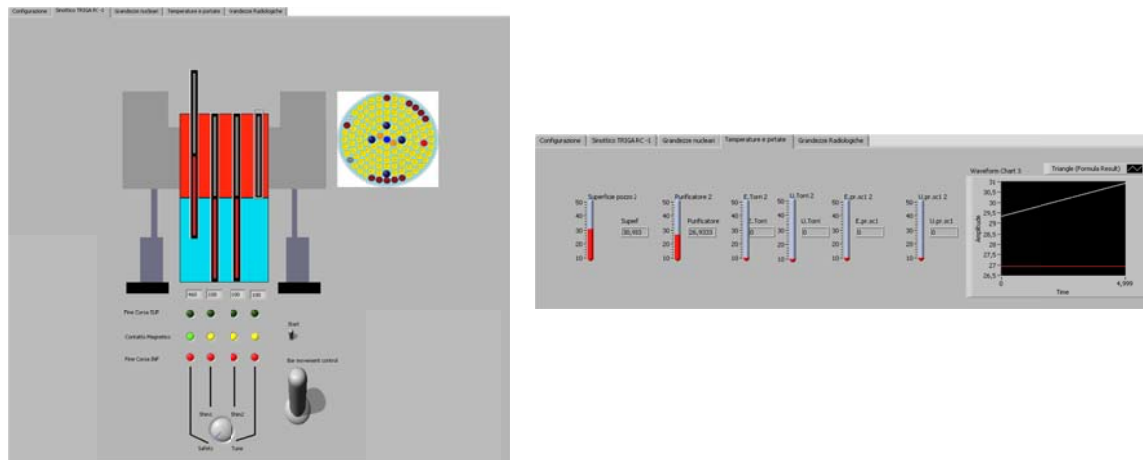

Fig. 2 Application front panels.Synoptic and temperature views.

Preliminary tests have been conducted on elementary measurement such as a thermocouple response to take into account the sensitivity and the correctness of our system modeled by LabView objects.

4. Conclusions

The system we have implemented with Labview objects has many features that allow a simple management of the devices present in a control room [5]. First, the system is scalable: any other device added to the actual control room can be easily modeled by using the SubVI LabView feature. Second, the system is available for a real time measurement that is totally independent from the control room instrumentation. This means, for instance, that a training operator can observe actions of a senior operator or supervisor directly from his/her PC without entering the control room, provided that a correct mode for data communication

protocol is preserved (e.g. by means of an internal dedicated LAN). Any variations of the measuredquantities will be displayed on the new digital supervisor system. It can be also used in a off-line mode. Indeed, when the reactor is shut down, it is possible to simulate the operation just using the simulating signal. Another way of use can be referred as a using-log-operation. This means that, during normal reactor operation, some quantities (or all), are logged into files. As a consequence, reading data with the proposed system, a specific operation state of the reactor can be reproduced. This can be useful for training students or operators to understand specific topics of reactor operations. In the next future,thanks to its flexibility and reusability, the system will be easily upgraded with the implementation of different instruments configuration. Control room instrumentation modeling is based on the creation of a basic virtual element, composed by two elementary steps: a simulation step and a processing step (for the simulated signals). Thus, the main effort is to define a signal able to reproduce real variables. Beyond initial validation through basic measurement on simple thermocouple signals, a further validation of this virtual instrument has been performed by comparing a logged variable, such as the temperature of the water reactor surface, with the corresponding simulated variable (Fig.3). In thus work, a linear interpolation of the function describing that variable behavior has been chosen. Of course, other interpolations choices are possible, e.g. using polynomial functions of an order higher than one. Generally, this procedure can always be applied to produce statistical comparison between simulated and real results.
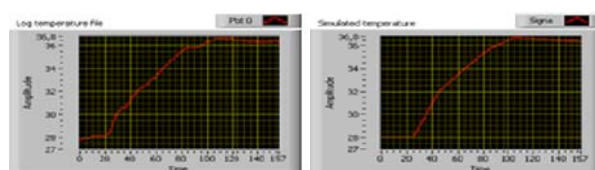

Fig.3Validation diagrams. Comparison between a logged temperature of the water reactor surface and the corresponding simulated variable

This work represents the first step of a high–operability supervisory tool design. Its future improvements in terms of functionality and integration cannot be carried out without the presence of a real console for a correct validation.

5. References

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