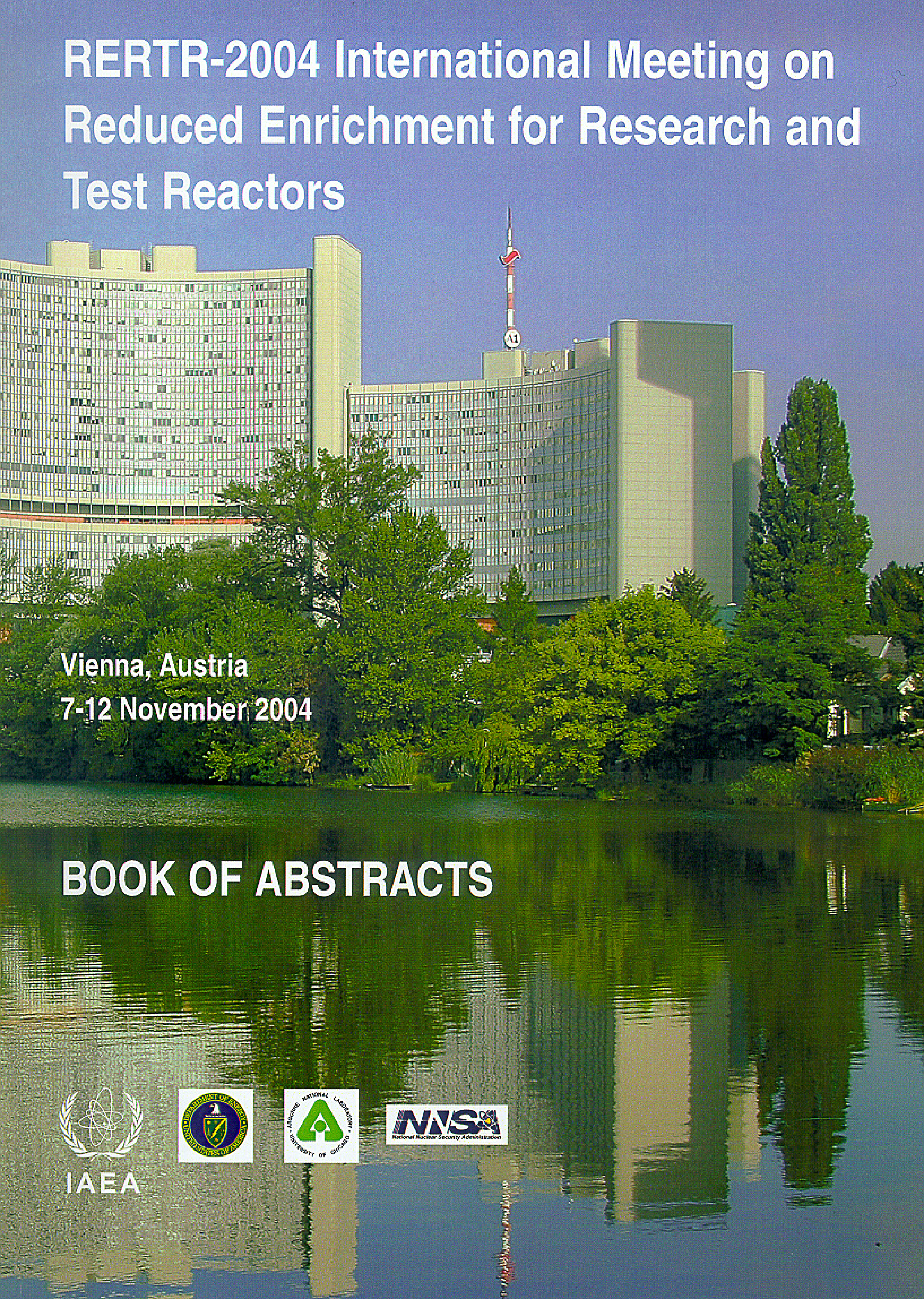
RERTR-2004 International Meeting on Reduced Enrichment for Research and Test Reactors



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BOOK OF ABSTRACTS

IAEA

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LIST OF PAPERS

STATUS AND PROGRESS OF THE RERTR PROGRAM IN THE YEAR 2004

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The overall status of the RERTR program at the time of the last RERTR meeting is reviewed, and the progress achieved since that meeting is described.

In the fuel area, unexpected failures of LEU U-Mo dispersion plates and tubes under irradiation testing have prompted a revision of the plans to qualify these fuels. While potential solutions to the difficulties with U-Mo dispersion fuels are being explored in collaboration with our international partners, greater emphasis has been placed on accelerating development of monolithic LEU U-Mo fuel. The feasibility of converting several Russian-designed research reactors to LEU fuels has been addressed, and progress has been made in the development of LEU-based 99Mo production processes. The Russian RERTR program has made significant advances.

A very important event of 2004 was the USDOE establishment of the Global Threat Reduction Initiative (GTRI). This new program accelerates and combines under the same USDOE management several programs, including RERTR, which aim to secure, remove, or dispose of, nuclear and other radioactive materials throughout the world that are vulnerable to theft by terrorists.

IAEA ACTIVITIES RELATED TO RESEARCH REACTOR FUEL CONVERSION AND SPENT FUEL RETURN PROGRAMS

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The IAEA has been involved for more than twenty years in supporting international nuclear non-proliferation efforts associated with reducing the amount of high-enriched uranium in international commerce. IAEA programs and projects have directly supported the Reduced Enrichment for Research and Test Reactors (RERTR) programme, as well as directly associated efforts to return spent research reactor fuel to the country of origin. IAEA efforts have included the development and maintenance of several data bases with highly useful information related to research reactors and research reactor spent fuel inventories that have been essential in planning and managing both RERTR and spent fuel return programs. Other IAEA regular budget programs have been highly useful in supporting research reactor fuel conversion from HEU to LEU, and in approaching issues common to many member states in dealing with RR spent fuel management issues and problems. The paper briefly describes IAEA involvement since the early 1980's in these areas, including regular budget and Technical Cooperation program activities, and focuses on efforts in the past five years to initiate or continue, assist and accelerate U.S. and Russian research reactor spent fuel return programs.

Key words: Research reactors, Research reactor fuel conversion, spent fuel management, research reactor spent fuel return programs

THE UNITED STATES FOREIGN RESEARCH REACTOR SPENT NUCLEAR FUEL ACCEPTANCE PROGRAM: PROGRAM EXTENSION

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The United States Department of Energy (DOE), in consultation with the Department of State (DOS), adopted the Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel in May 1996. The policy was slated to expire May 12, 2009. On October 15, 2003, a petition requesting a Program extension was delivered to the United States Secretary of Energy from a group of research reactor operators from foreign countries. The Secretary in April directed DOE undertake an analysis, as required by the National Environmental Policy Act, to consider potential extension of the Program. This paper discusses the findings from the NEPA analysis and the recommendations made, and potential changes in Program implementation that may result.

CURRENT STATUS AND FUTURE OF THE DEVELOPMENT OF NEW PIN-TYPE FUEL ELEMENTS AND ASSEMBLY FOR RESEARCH REACTORS

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VNIINM executes the development of LEU fuels on the basis U-? ? alloys and a new design of pin-type fuel elements under Russian RERTR Program. The development is carried out both for existing reactors, and for developing new advanced reactors.

These works is being performed in close cooperation with TVEL NPCC, NIIAR, PIYaF, RRC KI and ANL (USA) within international RERTR program.

At the present time the most important accomplishments can be summarized as follows.

- The irradiation of LEU U-Mo dispersion fuel (the uranium density equalled to 6,0 g/cm3) in two Russian research reactors: MIR (NIIAR, Dimitrovgrad) in pin-type fuel mini-elements and WWR-M (PIYaF, Gatchina) in full-scaled pin-type fuel assembly is going on.
- At the end of June, 2004 fuel in mini-elements had reached 50% burnup. The PIE results of the part of mini-elements, which had reached 25% burnup, have been received.
- The large set of calculational and experimental research on substantiation of IRT-type FA design with pin-type fuel elements has been completed. As result the optimum specification of dispersion pin-type FE and FA design has been chosen.
- The development of fuel elements on the basis of monolithic U-Mo fuel is in progress.

The summary of important results of performed works and their near-term future are presented in paper.

2004 STATUS OF RERTR ACTIVITIES IN CNEA-ARGENTINA

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During 2004, CNEA performed several RERTR activities, including :

LEU Very High Density Fuel development: two national technical meetings has taken place. During 2004 works are being focused on:

- U-Mo powder development and fabrication,
- research on U-Mo based ternary systems,
- UMo-Al diffusion couples out-of-pile experiments,
- 3-D thermal mechanical modelling of fuel behaviour under irradiation,
- atomistic modelling and experimental fuel particle and sheet coating,
- development of new matrix, new fuel particle and sheet, and new cladding
- development of advanced welding techniques to be applied on monolithic fuels

LEU high density fuel fabrication activities:

- status of fabrication tasks of silicide fuels for RRR reactor

LEU radioisotope production targets: two years of success

- target fabrication , irradiation and radioisotope production

Argentinean spent fuels: plans and projects

- intermediate storing
- final conditioning

Specific papers on topics will be presented during RERTR2004 meeting

STATUS OF REDUCED ENRICHMENT PROGRAM FOR RESEARCH REACTORS IN JAPAN

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The reduced enrichment programs for the JRR-3M, JRR-4 and JMTR of Japan Atomic Energy Research Institute (JAERI) had been completed by 1999. The KUR of Kyoto University Research Reactor Institute (KURRI) has been partially completed and is still in progress under the Joint Study Program with Argonne National Laboratory (ANL).

The JRR-3M using LEU silicide fuel elements had done a functional test by the Japanese Government in 2000, and the property of the reactor core was satisfied.

JAERI established an U-Mo fuel ad hoc committee for feasibility study concerning future LEU fuel instead of the silicide fuel in 2001, and an installation of the U-Mo fuel was estimated from 2012, but the U-Mo fuel development is facing very difficult situation to use an highly U-density of U-Mo fuel, so we will carefully study our U-Mo fuel installation plan. The Japanese Government approved a cancellation of the KUHFR Project in February 1991, and in April 1994 the U.S. Government gave an approval to utilize HEU fuel in the KUR instead of the KUHFR. Therefore, the KUR will be operated with HEU fuel until March 2006, then the full core conversion with LEU fuels will be done. All KUR spent fuel elements will be sent to the U.S. by March 2008.

PROGRESS WITH THE AUSTRALIAN REPLACEMENT RESEARCH REACTOR

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Construction of the new Australian Research Reactor, the replacement for the now 46 year old HIFAR research reactor, is now approximately 75% completed. Construction of the reactor facility began in April 2002 at ANSTO's Lucas Heights site near Sydney and commissioning is still on track for late 2005. Some details of the progress of construction and licensing and an outline of ANSTO research related to the use of Zircaloy-4 in the core of the reactor are given.

THE NEXT TEN YEARS OF RERTR FUEL DEVELOPMENT

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World events have placed new emphasis on the need for faster paced development and qualification of viable low-enrichment replacement fuels for use in research reactors that cannot operate on currently licensed and available fuels. The recently announced Global Threat Reduction Initiative (GTRI) sets definite goals for RERTR fuel development activities and resulting core conversions. To finally accomplish these long-standing goals will require a concerted and coordinated international effort as well as an increase in RERTR fuel development activity within the United States. This paper presents the U.S. fuel development technical strategy and plans for fuel development and qualification to meet the goals of GTRI over the nest decade.

POST-IRRADIATION ANALYSIS OF LOW ENRICHED U-Mo/Al DISPERSIONS FUEL MINIPLATE TESTS, RERTR 4 AND 5

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Interpretation of the post irradiation data of U-Mo/Al dispersion fuel mini plates irradiated in the Advanced Test Reactor to a maximum U-235 burn up of 80% are presented. The analyses addresses fuel swelling and porosity formation as these fuel performance issues relate to fuel fabrication and irradiation parameters. Specifically, mechanisms involved in the formation of porosity observed in the U-Mo/Al interaction phase are discussed and, means of mitigating or eliminating this irradiation phenomenon are offered.

FRENCH DEVELOPMENT AND QUALIFICATION PROGRAMS FOR THE JHR PROJECT FUEL ELEMENT

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Due to its high level flux performance, the Jules Horowitz Reactor (JHR), the European MTR in project, requires a compact core and a high density of fissile material.Consistent with the JHR design specifications, the reference fuel is the high density (8 g/cm3) UMo dispersion fuel with 0.61 mm meat thickness and 0.38 mm clad curved plates.World-size observations on UMo dispersion fuel have demonstrated good behavior for the UMo particles, but a weak behavior of the UMo/Al meat at high operating conditions due to the excessive formation of interaction compound between UMo particles and Al matrix.Different solutions are investigated in France as in other countries in order to improve the UMo dispersion fuel behavior. The monolithic option is also studied.Nevertheless, the necessary additional investigations risk modifying the international schedule for high density UMo qualification and licensing.So, to secure the JHR start, a backup U3Si2 fuel will be qualified with a 4.8 g/cm3 density and 0.61 mm meat thickness. To be consistent with the JHR objectives, this backup fuel will use ~35% 5U enriched fuel. This possible starting enrichment will be reduced at 20% when the international standard UMo fuel will be available.This paper presents both the updated French programs on the UMo fuel (dispersion and monolithic), and back up solution. Those programs are developed by CEA in close collaboration with AREVA Group (CERCA, TA and COGEMA) and within intensive international collaborations (USA, Germany, Korea, and Russia).The development schedule, including full size irradiation tests for qualification, matches the JHR planning, and keeps the first criticality compatible with the 2013 goal.

RECH-1 TEST FUEL IRRADIATION

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Since May 2003, one RECH-1 fuel element has been submitted to irradiation at the HFR-Petten, Holland. This paper presents the objectives and progress up to date of this fuel qualification under irradiation.

Besides, a brief description of CHI/4/021, IAEA's Technical Cooperation Project that has supported this irradiation test, is also presented here.

DEVELOPMENT, IRRADIATION TESTING AND PIE OF U-MO FUEL AT AECL

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This paper reviews recent U-Mo dispersion fuel development, irradiation testing and post-irradiation examination (PIE) activities at AECL. Low-enriched uranium fuel alloys and powders have been fabricated at Chalk River Labs, with compositions ranging from U-7 to -10 wt % Mo. The bulk alloys and powders were characterized using optical and scanning electron microscopy, chemical analysis, X-ray diffraction and neutron diffraction analysis. The analyses confirmed that the powders were of high quality, and in the desired gamma-phase. Subsequently, kilogram quantities of DU-Mo and LEU-Mo powder have been manufactured for commercial customers. Mini-elements have been fabricated with LEU-7Mo and LEU-10Mo dispersed in aluminium, at a nominal loading of 4.5 gU/cm3. These have been irradiated in the NRU reactor at linear powers up to 100 kW/m. The mini-elements achieved 60 atom% 235U burnup in 2004 March, and the irradiation is continuing to a planned discharge burnup of 80 atom% 235U. Interim PIE has been conducted on mini-elements that were removed after 20 atom% burnup. The PIE results are presented in this paper.

MTR FUEL PLATE QUALIFICATION IN OSIRIS REACTOR

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Qualification of new MTR fuel needs the irradiation in research reactors under representative neutronic, heat flux and thermohydraulic conditions. CEA is engaged since the 1990s in the reduction of enrichment of MTR fuel by studying the behaviour under irradiation of U3Si2 alloy-based fuel plates and more recently by starting a qualification program of UMo fuel.

The experiments are performed in the OSIRIS reactor by irradiating MTR full size fuel plates in the IRIS device. This device is derived from the device used previously in the SILOE reactor (Grenoble) which has been definitively shutdown at the end of 1997.

The IRIS device has the same external geometry as an OSIRIS standard fuel element and is located in the reactor core. It can be loaded with four removable plates, either inert or fuel plates, separated by inert plates.

To be successfully completed, the qualification experiment calls for a wide range of facilities and skills provided by the experimental platform. :

THE VALMONT EXPERIMENTAL PROGRAMME FOR THE NEUTRONICS QUALIFICATION OF THE UMO/AL FUEL FOR THE JULES-HOROWITZ-REACTOR

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The proper and safe estimation of the uncertainties associated with the results of neutronics calculations is crucial for the design studies of the future Jules Horowitz Reactor (JHR). However, the neutronics characteristics of this innovative core and, in particular, the use of the new UMo/Al fuel preclude the use of the existing qualification databases.

The need for specific experimental data for this fuel was met by the VALMONT programme, which consisted in the precise measurement of the reactivity effect of fuel samples in the MINERVE facility at Cadarache (France). These measurements, corroborated by γ-spectroscopic analysis on irradiated UMo/Al fuel, permit to qualify the HORUS3D/N neutronics calculation route for this fuel. Moreover, the application of the representativity method allowed reducing the uncertainty on the reactivity, due to the uncertainties of the basic nuclear data of the Uranium isotopes, by a factor of 3.

FULL SIZE PLATES IRRADIATION – FUTURE EXPERIMENT

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The French UMo group has performed four full sized experiments on UMo dispersion plates with different compositions (7-9% Mo) and different types of powder (ground and atomised).

The FUTURE experiment contained 2 full-sized plates with high LEU uranium loading, i.e. 8,5 $g.cm^{-3}$. The fuel meat consisted on 7 wt% molybdenum-uranium alloy atomised powder dispersed in pure aluminium matrix.

The plates were irradiated at BR2 – Belgium reactor in FUTURE device up to 350 $W.cm^{-2}$ and 130°C, respectively for the peak heat flux and clad surface temperature. The experiment was stopped prematurely after two cycles due to unexpected local pillowing at maximum flux plane of the plates.

After recalling the irradiation conditions, this paper presents briefly the most important results and analyses. MAIA code is used for mechanical interpretations.

IRRADIATION PERFORMANCE OF U3Si LEU FUELS IN HANARO

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To verify the irradiation performance of HANARO fuel at a high power and burnup, the in-pile irradiation tests were performed. Two types of test fuel, two un-instrumented fuels(Type-A) for higher burnup irradiation and one instrumented fuel(Type-B) for higher linear heat rate and precise measurement of irradiation conditions, had been designed and fabricated. All the test fuel assemblies were made of 6 fuel elements located in the outer ring of the hexagonal fuel assembly and 30 aluminum dummy elements. The test fuel assemblies were irradiated in the HANARO core. Type-A fuel assembly was discharged after 69.9 at% average and 85.5 at% peak burnup, respectively. Type-B fuel assembly was achieved a maximum power higher than 120 kW/m without losing its integrity and without showing any irregular behavior. Another test fuel assembly for the high power irradiation test was developed along the localization plan of HANARO fuel in KAERI. The high power test fuel assembly was composed of 3 pulverized and 3 atomized U3Si fuels. The test assembly was irradiated during 173.7 reactor operation days in CT hole with the highest neutron flux in HANARO core. The reactor physics calculations showed average discharge burnup of 63 at%U-235, maximum local burnup of 77 at%U-235, average linear power of 83 kW/m and maximum linear power of 121.6 kW/m.

Detailed non-destructive and destructive PIE (Post-Irradiation Examination), such as the measurement of burnup distribution, fuel swelling, clad corrosion, dimensional changes, fuel rod bending strength, micro-structure, etc., were performed in the IMEF(Irradiated Material Examination Facility) located in the inside of the HANARO boundary. The measured results were analyzed and compared with the predicted performance values and the design criteria. It was verified that HANARO fuel maintains proper in-pile performance and integrity even at the high power of 120 kW/m and up to the high burnup of 85 at%.

PIE OF THE 2ND IRRADIATION TEST (KOMO-2) FOR ATOMIZED U-MO DISPERSION ROD FUELS IN KAERI

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An alternative fabrication method for polycrystalline uranium foils has been investigated using a cooling-roll casting method in KAERI since 2001, in order to produce a medical isotope ^{99}Mo, the parent nuclide of 99^{m}Tc. The fabrication method of wide uranium foils produced by cooling-roll casting has been optimized to improve the quality of uranium foils and the economic efficiency of the foil fabrication with the modifications of the casting apparatus and the variations of the various process parameters. The injection control device of the uranium melt was applied to cooling-roll casting apparatus, in order to stabilize the fabrication process and to increase the yield of uranium foils through the prevention of the melt leakage. As the uranium has a low thermal conductivity, the collection apparatus was modified to fabricate the uranium foils without great defects soundly, leaded to improve the quality and the yield of the uranium foils. The dimension and the surface state of the uranium foils were also adjusted with the revolution speed of cooling roll, the ejection pressure of melt, the gap distance between nozzle slot and cooling roll, the superheat of the metal, and the atmosphere of melting and casting. Then, continuous polycrystalline uranium foils with a thickness range of 100 to 150? and a width of about 50 mm were fabricated with a better quality of uranium foils and a higher economic efficiency of the foil fabrication, through the modifications of the casting apparatus and the variations of the various process parameters.

COURSE OF PIN FUEL TEST IN WWR-M REACTOR CORE

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Pin type fuel element square form with twisted ribs was developed at VNIINM as an alternative for tube type.

Two variants full-scale fuel assemblies are under test in the core of WWR-M reactor at Gatchina.

One assembly contents fuel elements with dioxide LEU and other UMo LEU. Both types of fuel elements have an aluminium matrix.

The results of the first stages of the test are presented

.

AN INVESTIGATION ON THE IRRADIATION BEHAVIOR OF ATOMIZED U-MO/AL DISPERSION ROD FUELS

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The second irradiation fuel experiment, KOMO-2, for the qualification test of atomized U-Mo dispersion rod fuels with U-loadings of 4-4.5 gU/cc in KAERI was finished after irradiation until 70 at%U235 peak burn-up and is subjected to IMEF (Irradiation material Examination Facility) for post-irradiation analysis in order to understand fuel irradiation performance of U-Mo dispersion fuel. Current results on PIE of KOMO-2 revealed that the U-Mo/Al dispersion fuel rods exhibit sound performance without break-away swelling, but most of the fuel rods irradiated at high linear power show the extensive formation of the interaction phase between the U-Mo particle and Al matrix. In this paper, the analysis on the PIE results, focused on the diffusion related microstructures obtained from the optical and EPMA observations, will be presented in detail. And thermal modeling will be carried out to calculate the temperature of the fuel rod during irradiation.

Y-12 PRODUCT IMPROVEMENTS EXPECTED TO REDUCE METAL PRODUCTION COSTS AND DECREASE FABRICATION LOSSES

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The Y-12 National Security Complex supplies uranium metal and uranium oxide feed material that is then fabricated into fuel for research reactors around the world. Over the past two to three years, Y-12 has learned a great deal about its Low Enriched Uranium (LEU) product. The LEU is produced by taking U.S. surplus Highly Enriched Uranium (HEU) and blending it with depleted or natural uranium. The surplus HEU comes from dismantled U.S. weapons parts that have been declared as surplus. Those research reactors that use LEU from Y-12 are making important contributions to international nuclear non-proliferation by using LEU rather than HEU, and by helping to disposition former weapons material.

We clearly understand that our customers want to keep fuel costs as low as possible. We at Y-12 are making every effort to improve efficiencies in producing the uranium through standardizing the chemical specifications as well as the product mass and dimensional qualities. This paper will discuss the new standard specification that we have proposed to existing LEU metal customers and fuel fabricators. It will also cover Y-12's progress on a new mold-design that will result in a more uniform, higher quality product that is less expensive to produce. This new product is expected to decrease overall fabrication losses by 5-10%, depending on the fabricator's process. The paper will include planned activities and the schedule associated with implementation of the new specification and product form.

UPDATE ON FUEL FABRICATION DEVELOPMENT AT ARGONNE NATIONAL LABORATORY

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In its effort to develop research reactor fuel with a high fissile loading, Argonne National Laboratory has continued its advanced fuel development efforts. Monolithic fuel, where the fuel is in the form of a single fuel foil, is being developed as the ultimate in fuel loading capacity. Work has been done on different monolithic fabrication methods that have resulted in process refinements. Effort is also underway to develop a uranium-molybdenum dispersion fuel plate that will be resistant to the irradiation shortcomings noted in previous tests. Alloying additions to the aluminum matrix are being investigated. These fuels are being fabricated for use in an irradiation experiment scheduled for insertion in 2005.

IMPROVING THE PERFORMANCE OF U-Mo FUELS

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Recent developments showed that uranium-molybdenum nuclear fuel particles dispersed in an aluminum matrix had misbehavior when irradiated at high neutron fluxes. The appearance of a third phase, with the presence of great porosity in the interaction zone of the Al/U-Mo interface, conditions severely the performance of this fuel. At the light of the resolution of this limitation, UMo monolithic fuel achieves a greater importance, since there is some expectation that in this bulk geometry the problem will not be present.

From the simplest point of view, the addition of extra alloys to the aluminum matrix or to the nuclear fuel can be an alternative to reduce the interface growing kinetics and thereafter the appearance of the problematic third phase. The kinetics reduction would be a quantitative effect controlling chemical potentials (diffusion driving force) and barely will avoid the problem. Similar considerations can be attributed to the monolithic fuel if only quantitative solutions are proposed.

In this paper are presented two drastical alternatives, from the point of view of qualitative metallurgy, for increasing the performance of U-Mo fuels. The first one is related with the coverage of the fuel particles with compound diffusion barriers to avoid the transportation of uranium and aluminum threw them. The second alternative is a monolithic fuel with zircaloy cladding where the interaction is much smaller than with aluminum.

REACTION LAYER BETWEEN U-7WT%MO AND AL ALLOYS IN CHEMICAL DIFFUSION COUPLES

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Several failures in U-Mo dispersion fuel plates like pillowing and large porosities have been reported during irradiation experiments. These failures have been assigned to the formation of a large (U-Mo)/Al interaction product under high operating conditions. The modification of the matrix by alloying Al to change the interaction layer and improve its irradiation behavior, has been proposed. This paper reports diffusion experiments performed between U-7wt%Mo and various Al alloys containing Mg, Si and/or Zn. By the use of SEM and X-Ray diffraction, it was found that with 5.2wt% Si the interaction layer is constituted by (U,Mo)(Si,Al)3 and no (U,Mo)Al4 is detected.

As part of the studies of properties of the U-Mo alloys the start of the transformation of the g phase vs. temperature is being evaluated for the present U-7wt%Mo alloy. These results are being used to plan the future diffusion program that will include diffusion under irradiation at CNEA RA3 reactor

A MICROSTRUCTURALLY-BASED MODEL FOR THE EVOLUTION OF IRRADIATION-INDUCED RECRYSTALLIZATION IN U-MO MONOLITHIC AND AL-DISPERSION FUELS

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In a monolithic U-Mo fuel design, in the absence of substantial interaction product development fuel swelling will be the primary deformation mechanism. Irradiation-induced recrystallization appears to be a general phenomenon in that it has been observed to occur in a variety of nuclear fuel types, e.g. U-x Mo, UO2, and U3O8. The recrystallization process results in sub-micron sized grains that accelerate fission-gas swelling due to the combination of short diffusion distances, increased grain-boundary area per unit volume, and greater intergranular bubble growth rates as compared to that in the grain interior. An expression has been derived for the fission density at which irradiation-induced recrystallization is initiated that is athermal and weakly dependent on fission rate. The initiation of recrystallization is to be distinguished from the subsequent progression and eventual consumption of the original fuel grain. The formulation takes into account the observed microstructural evolution of the fuel, the role of precipitate pinning and fission gas bubbles, the triggering event for recrystallization, as well as the evolution of recrystallization as a function of burnup. The calculated dislocation density, fission gas bubble size distribution, fission density at which recrystallization first appears, and the subsequent progression as a function of burnup are compared to measured quantities. Estimates of fuel swelling for a monolithic U-Mo fuel design are provided.

THERMOMECHANICAL DART CODE IMPROVEMENTS FOR LEU VHD DISPERSION AND MONOLITHIC FUEL ELEMENT ANALYSIS

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A collaboration agreement between ANL/USDOE and CNEA Argentina in the area of Low Enriched Uranium Advanced Fuels has been in place since October 16, 1997 under the "Implementation Arrangement for Technical Exchange and Cooperation in the Area of Peaceful Uses of Nuclear Energy. An annex concerning DART code optimization has been operative since February 8, 1999.

Previously, as a part of this annex a visual thermal FASTDART version and also a DART TM thermomechanical version were presented during RERTR 2002 and RERTR 2003 Meetings. During this past year the following activities were completed:

- Optimization of DART THERMAL code Al diffusion parameters by testing predictions against reliable data from RERTR experiments.
- Improvements on the 3-D thermo-mechanical version of the code for modeling the irradiation behavior of LEU U-Mo monolithic fuel

Concerning the first point, by means of an optimization of parameters of the Al diffusion through the interaction product theoretical expression, a reasonable agreement between DART temperature calculation with reliable RERTR PIE data was reached.

The 3-D thermomechanical code complex is based upon a finite element thermal-elastic code named TERMELAS, and irradiation behavior provided by the DART code. An adequate and progressive process of coupling calculations of both codes at each time step was reached. The coupling of the various components of the calculation was benchmarked and validated against RERTR PIE data.

Various results will be shown during RERTR2004 meeting

ENHANCEMENTS TO THE PLATE FUEL PERFORMANCE CODE FOR ANALYZING FULL-SIZE AND MONOLITHIC FUEL PLATES

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The PLATE fuel performance code, under development for several years, was originally designed for the thermal analysis of miniature U-Mo or U3Si2 experimental dispersion fuel plates. Recent code enhancements have extended the analysis capability of PLATE to full-size fuel plates. Additionally, the capability to analyze fuel plates using monolithic U-Mo fuels has been added. Data from the postirradiation examination of the XP-2 and XP-5 monolithic fuel plates from the RERTR-4 irradiation test have been used to develop an interaction rate correlation for U-Mo fuel and Al-6061 cladding; high-burnup data from several RERTR-4 U-Mo fuel plates have been used to refine the fuel swelling correlations employed by PLATE. These code enhancements are described.

ANALYSES FOR INSERTING FRESH LEU FUEL ASSEMBLIES INSTEAD OF FRESH HEU FUEL ASSEMBLIES IN THE DALAT NUCLEAR RESEARCH REACTOR IN VIETNAM

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Analyses were performed by the RERTR Program to replace 36 burned HEU (36%) fuel assemblies in the Dalat Nuclear Research Reactor in Vietnam with either 36 fresh fuel assemblies currently on-hand at the reactor or with LEU fuel assemblies to be procured. The study concludes that the current HEU (36%) WWR-M2 fuel assemblies can be replaced with LEU WWR-M2 fuel assemblies that are fully-qualified and have been commercially available since 2001 from the Novosibirsk Chemical Concentrates Plant in Russia.

The current reactor configuration using re-shuffled HEU fuel began in June 2004 and is expected to allow normal operation until around August 2006. If 36 HEU assemblies each with 40.2 g 235U are inserted without fuel shuffling over the next five operating cycles, the core could operate for an additional 10 years until June 2016. Alternatively, inserting 36 LEU fuel assemblies each containing 49.7 g 235U without fuel shuffling over five operating cycles would allow normal operation for about 14 years from August 2006 until October 2020. The main reason for the longer service life of the LEU fuel is that its 235U content is higher than the 235U content needed simply to match the service life of the HEU fuel. Fast neutron fluxes in the experiment regions would be very nearly the same in both the HEU and LEU cores. Thermal neutron fluxes in the experiment regions would be lower by 1-5%, depending on the experiment type and location.

PROGRESS IN JOINT FEASIBILITY STUDY OF CONVERSION FROM HEU TO LEU FUEL AT IRT-200, SOFIA

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The new 200 kW IRT-Sofia research reactor of the Institute for Nuclear Research and Nuclear Energy (INRNE) of the Bulgarian Academy of Science, Sofia, Bulgaria is jointly studied with the RERTR Program at Argonne National Laboratory (ANL) to examine the feasibility of conversion from the use of fuel containing highly enriched uranium (HEU, 36% 235U) to use of fuel containing low enriched uranium (LEU, 19.75% 235U).

The reference design had a core configuration using 14 IRT-2M fuel assemblies (four 4-tubes and 10 3-tubes) with 36% HEU. This HEU fuel is no longer available since it was transported from INRNE to Russia in December 2003 as part of an agreement with the US DOE. An LEU core configuration using 14 IRT-4M fuel assemblies (four 8-tubes and ten 6-tubes) which yields a similar flux performance, when compared with the HEU design, was created.

Results of detailed calculations comparing the new LEU core with the reference HEU core design are presented. From these results it is concluded that the LEU core performance (both in term of fluxes for the experiments and in fuel consumption) is very similar to the HEU reference core.

FEASIBILITY STUDY OF THE WWR-K REACTOR CONVERSION TO LOW-ENRICHED FUEL

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Outcomes of the first stage of calculation studies, referring to the eight-tube fuel assembly (FA) and four-tube one (in which the CPS channel control rod is located), are presented for fuel elements on a base of the UO2+Al composition with the uranium density 3.0 g×cm-3. Parameters of the eight-tube FA are as follows: the fuel element and fuel core thickness values are 1.60 and 0.70 mm respectively; a gap between neighbor fuel elements – to provide heat agent flow - comprises 2.00 mm, the uranium-235 content is 270.6 g, the heat-exchange surface area is 13146 cm2.

Neutronic calculations are performed with 3-D codes, the thermalphysic calculations – with the code «ASTRA». Versions of the reactor core with both light-water and beryllium reflector are considered.

For the initial reactor core, loaded with 18 eight-tube FAs and 10 four-tube ones, with water reflector the following is obtained: Keff=1.0836±0.0002, the thermal/fast (?>1.15 MeV) flux density values are, respectively, (2.30±0.30)×1014 and (4.2±0.3)×1013 cm-2 c-1 . For the 120-day campaign, the uranium-235 burnup level in the highest fuel rating FA reaches 19.5%; level of the stationary poisoning by isotope Xe-135 comprises 3.4% Dk/k, and 64.5 g of Pu-239 is produced in the reactor core. At the heat agent flow velocity comprising 2.0 m/s and the temperature in the reactor core entrance comprising 35 °C the maximum temperature of the most hot fuel element is 64 °C. All data are given for the reactor 6-MW thermal power.

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FEASIBILITY STUDY FOR LEU CONVERSION OF THE WWR-K REACTOR AT THE INSTITUTE OF NUCLEAR PHYSICS IN KAZAKHSTAN USING A 5-TUBE FUEL ASSEMBLY

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A feasibility study by the RERTR program for possible LEU conversion of the 6 MW WWR-K reactor concludes that conversion is feasible using an LEU 5-tube Russian fuel assembly design. This 5-tube design is one of several LEU designs being studied. The assembly contains 200 g 235U with an enrichment of 19.7% in four cylindrical inner tubes and an outer hexagonal tube with the same external dimensions as the current HEU (36%) fuel assembly (112.5 g 235U). The fuel meat material, LEU UO2-Al dispersion fuel with ~2.5 g U/cm3, has been extensively irradiation tested in a number of reactors.

Since the 235U loading of the LEU assemblies is much larger than the HEU assemblies, a smaller LEU core with five fuel rows (instead of six fuel rows in the HEU core) would consume about half as much many fuel assemblies per year as the HEU core and provide thermal neutron fluxes in the inner irradiation channels that are ~17% larger than with the present HEU core. The current 21 day cycle length would be maintained and the average discharge burnup would be ~42%. Neutron fluxes in the five outer irradiation channels would be smaller in the LEU core unless these channels can be moved closer to the LEU fuel assemblies.

Results show that the smaller LEU core would meet the reactor's shutdown margin requirements and would have an adequate thermal-hydraulic safety margin to onset of nucleate boiling.

PRELIMINARY FEASIBILITY STUDY FOR LEU CONVERSION OF THE 10 MW IRT-1 REACTOR AT THETAJOURA NUCLEAR RESEARCH CENTER IN LIBYA

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A preliminary feasibility study by the RERTR program for LEU conversion of the 10 MW IRT-1 reactor at the Tajoura Nuclear Research Center in Libya concludes that the conversion is feasible using LEU IRT-4M fuel that is qualified in accord with ROSATOM requirements and is expected to be commercialized (licensed) for production by the Novosibirsk Chemical Concentrates Plant in Russia in 2005. The results show that the number of IRT-4M fuel assemblies used per year would be less than half the number with HEU (80%) fuel because of the higher 235U loading of the LEU fuel assemblies. Thermal neutron fluxes for isotope production in the irradiation positions in the beryllium reflector will be less than 2% lower than with current HEU fuel. All of the information for this paper was obtained from limited open-literature sources. A detailed feasibility study is in progress.

CALCULATED RESEARCH FOR CONVERSION OF RESEARCH REACTOR IN UZBEKISTAN TO LOW ENRICHED FUEL

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The paper comprises the results of calculations for conversion of nuclear reactor VVR-CM (located in Uzbekistan) from HEU to LEU by means of tube and rod type fuel elements with U-Mo alloy nuclear fuel.

CORE CONFIGURATION OF THE SYRIAN REDUCED ENRICHMENT FUEL MNSR

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The possibility of substituting the actual HEU by a LEU by a MEU in the Syrian MNSR is investigated through a pre-constructed 3-D detailed model of the reactor.

Core configuration does not change if a reduced enrichment fuel (20% u-235, with the same percentages of impurity and eliminating aluminium) is used. The required density for the reactor to be critical in this case would be 7.29 g/cm^2.

If a specific fuel is used (20 w/o U235, 40 w/o U), the reactor will have again the same actual initial excess reactivity if 2 standard fuel rods are added to each fuel circle.

WHAT THE DIFFERENCE TO USE LEU AND HEU FUELS SEPARATELY OR TOGETHER IN A RESEARCH REACTOR

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Concerning of nuclear material safety, most of the research reactors are advised to shift from HEU (high enriched-%93 U235) to LEU (low enriched-%20 U235) fuel. When LEU and HEU fuel elements are to be used together in a research reactor, some design and safety problems are encountered. According to use of the research reactor, some of them such as MTR type may not show any considerable difference for HEU or LEU fuel, but the efficiency of radio-isotop production generated from the thermal neutron reaction may decrease about twenty-thirty percent when LEU fuel is used. Fine mesh-sized 3-D neutronic analysis of TR-2 research reactor shows some problems when LEU and HEU fuel elements are used together in the reactor. In thermal-hydraulic analysis, LEU fuel element design gives better result for the edge fuel plates.

LVR-15 REACTOR PERFORMANCE AND TRANSFORMATION TO LOW ENRICHED FUEL

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Experimental research reactor LVR-15 situated in Nuclear Research Institute Rez, plc. has been utilized since 1956. The present reactor nominal power is 10 MW. Standard reactor cycle is 21 days and the reactor operates 8 – 10 cycles per year. The State Office for Nuclear Safety newly licensed the reactor till 2014.

The reactor is of polyvalent use. The basic research is carried out using horizontal neutron beams and one of them is used for development and application of the boron neutron captures therapy for brain tumors. Mostly material testing of PWR and BWR specimens is performed in high-pressure loops and irradiation rigs operated in the reactor. Several vertical channels serve for production of neutron transmutation doped silicon and isotopes production for medical purposes.

Reactor was originally designed with EK-10 fuel type with the 80% 235U enrichment. Later, the fuel enrichment was changed to 36 % with the use of the IRT-2M fuel type. The transformation took several reactor cycles using gradually different mixed core configurations.

The neutron-physical characteristics of the reactor core and fuel cycle are carried out using the WIMS-D4m, NODER and OMEGA programs. The codes were used for a preliminary evaluation of essential changes of main neutronic characteristics of the LVR-15 core with the prospective changeover to 20 % 235U enrichment. Three types of FA-s has been assessed: 1) currently used IRT-2M(36%), 2) IRT-3M(20%), 3) IRT-4M(20%) and results are presented in the article.

FLUX ENHANCEMENT OPTIONS FOR AN LEU-FUELED MIT REACTOR

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The Monte-Carlo transport code MCNP was used to evaluate possible arrangements of cores for the MIT Reactor using monolithic LEU fuel. Plate and moderator thicknesses were varied, and fixed absorbers and inner reflectors added in an effort to maximize available neutron fluxes at in-core and ex-core locations of experimental facilities. Addition of D2O in the H2O moderator was also evaluated. Comparisons of the fast, epithermal, and thermal fluxes were made at selected locations. Keff was also evaluated and critical blade heights compared with the existing HEU core. Results indicate that the LEU fluxes could approach HEU values with the use of a fueled in-core experimental facility, a fixed boron absorber spider and an inner beryllium reflector.

NUCLEAR FACILITIES OF THE OF THE NATIONAL ACADEMY OF SCIENCES OF BELARUS ON THE BASIS HIGH ENRICHED URANIUM

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The investigations in the field of nuclear physics, development of numerical calculation methods for nuclear reactors, neutron physics and etc. are carried out at the Joint Institute for Power and Nuclear Research - Sosny (JIPNR-Sosny) since the 60s after putting into operation the research reactor and the critical assemblies. A large range of different configuration (geometry,composition) of critical assemblies have been constructed at the NAS Belarus during 25 years of studying neutronic of the special (fast and thermal) reactors. The Chernobyl accident brought a massive public reaction to nuclear efforts and the reactor ceased operation in 1987 and was shut down in 1991 and at present all investigations in these fields are being carried out on the basis of the subcritical assemblies driven with high intensity neutron generator. The facilities with fast and thermal neutron spectra are fuelled with UO_2 enriched to 10% -90% in ^{235}U.

ACCELERATOR-DRIVEN SUBCRITICAL ASSEMBLY: CONCEPT DEVELOPMENT AND ANALYSES

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A conceptual design activity has been started to develop an accelerator-driven subcritical assembly using the electron accelerators of Kharkov Institute of Physics and Technology (KIPT) of Ukraine. The main functions of the subcritical assembly are the production of medical isotopes and the support of the Ukraine nuclear power industry. Reactor physics experiments and material performance characterization will be carried out utilizing the capabilities of the subcritical assembly. The neutron source of the subcritical assembly is generated from the interaction of 100 KW electron beam with a tungsten or uranium target. The electron beam has a uniform spatial distribution and the electron energy is in the range of 100 to 200 MeV. Also, a 23-MeV deuteron beam with beryllium target is under consideration. The subcritical assembly is planned to use high enriched uranium (HEU). This activity is evaluating the possibility of converting the assembly to use low enrichment uranium (LEU) without penalizing its performance. Different fuel forms are under consideration with water coolant including the LEU fuel of the Kiev research reactor. Beryllium or water is utilized for the reflector of the subcritical assembly. In the design process, the neutron flux is maximized to enhance the capabilities of the subcritical assembly. The conceptual design and the design analyses will be presented in this paper. The analyses include nuclear physics, neutronics, heat transfer, hydraulics, and material issues.

USE OF COMPUTATIONAL FLUID DYNAMICS (CFD) TOOLS FOR FUEL ASSEMBLY ANALYSIS

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Computational Fuel Dynamics (CFD) tools have been applied to analyze the thermal-hydraulics behavior of several research reactor fuel assemblies. The detailed three-dimensional fluid flow and temperature fields obtained from these analyses are used to obtain refined estimates of various safety parameters, such as the margin to onset of nucleate boiling.

As a particular example, the STAR-CD computer program has been applied to the pin-type and tube-type fuel assemblies proposed as low enriched uranium (LEU) replacements for the high enriched uranium (HEU) (36%) IRT-3M fuel assemblies currently used in the WWR-SM reactor in Uzbekistan. For fuel assemblies containing twisted, finned pin-type fuel, STAR-CD was first used to model the single pin having the highest power density along with its associated coolant as an isolated unit cell. Velocity, pressure, temperature, heat flux, etc. are calculated on a detailed spatial basis in the coolant, cladding, and fuel. The model can be expanded to include multiple fuel pins; the computed motion of coolant from unit cell to unit cell can reduce the peak temperatures below what one would compute using a single-pin model.

Similar analyses have been performed for assemblies where the fuel is in a set of concentric tubes which are separated from each other by coolant. When the entire assembly is modeled, the calculated results show, for example, the relative under- and over-cooling which occurs in various portions of the fuel assembly.

SAFETY ANALYSIS OF THE WWR-M REACTOR IN UKRAINE TO ALLOW OPERATION USING LEU FUEL

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The 10 MW, WWR-M research reactor of the Kiev Institute for Nuclear Research is jointly studied with the Argonne National Laboratory for conversion from HEU (36%) to LEU (19.75%) fuel. The reactor currently uses HEU (36%) WWR-M2 fuel assemblies (3 tubes, UO2-Al fuel meat with 1.1 gU/cm3 and 37.0 g 235U). Candidate LEU replacement fuel assemblies are: LEU WWR-M2 (3 tubes, UO2-Al fuel meat with 2.5 gU/cm3 and 41.7 g 235U). To qualify this LEU fuel for conversion of the WWR-M reactor in Ukraine, neutronic and thermal-hydraulic characteristics of LEU fuel equilibrium core and transition mixed core containing both LEU and HEU fuel are calculated. The following accidents are analyzed: spontaneous withdrawal of a control rod bank; incidental falling of a fuel assembly in a cell of the core; partial blockage of coolant flow across the core with accompanying partial melting of the core; full break of the first loop pipe with accompanying full melting of the core. The safety of fresh and depleted HEU and LEU fuel storage is analyzed. The models applied for calculations are validated against measured data, which include critical experiment results for fresh HEU (36%) WWR-M2 fuel assemblies and measured neutronic distributions in a real WWR-M reactor core.

PREFORMANCE OF PARR-1 WITH LEU FUEL

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The Pakistan Research Reactor (PARR-1) went critical in 1965 with HEU fuel. The reactor core was converted to LEU fuel with power upgradation from 5MW to 10 MW in 1992. The reactor has been operated with LEU fuel for about 10,000 hours and has produced about 66000 MWh energy up to now. The average burn up of the irradiated fuel is about 42%. The fuel performance during the last 12 years has been excellent. Post irradiation visual inspection of the fuel has revealed no abnormality. During operation there have been no signs of releases in the pool water establishing the full integrity of this fuel. The reactor has been mainly utilized for radioisotope production, beam tube experiments including neutron diffraction studies, neutron radiography etc. Studies have been completed to operate the reactor with a mixed core (HEU + LEU) to utilize the less burnt HEU fuel elements. A major project of production of fission moly using PARR-1 is in final stages.

MONOLITHIC FUEL AND HIGH-FLUX REACTOR CONVERSION

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Monolithic fuels are the most promising candidate for a next generation of high-density research reactor fuels. If successfully developed, the remaining HEU-fueled reactors in the world could presumably be converted to low-enriched fuel and the use of highly enriched uranium in the civilian nuclear fuel cycle eventually terminated.

The most challenging type of reactors to convert are single element reactors because their core geometry is generally the least flexible. This specific reactor type is therefore the primary focus of this article. Based on new computational tools and optimization methods, neutronics calculations are presented to assess the potential of monolithic fuels for conversion of high-flux reactors in general and of single element reactors in particular.

REDUCED ENRICHMENT PROGRAM FOR FRM-II

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In 2004 the high-flux research reactor FRM-II in Garching started operation using U_3Si_2 high-density fuel (up to 3 gU/cc) with an enrichment of 93%. As committed in the nuclear license for FRM-II its fuel has to be converted to MEU with a maximum enrichment of 50% until end of 2010. However, reactor performance and safety margin must not be significantly modified using a new qualified fuel. FRM-II has launched an important investigation program consisting of UMo fuel irradiations in the range of FRM-II reactor conditions. According to the last UMo irradiation results the international community has proposed in 2004 improvements in order to reduce the $(UMo)Al_x$ interaction product formation. Six full size plates with UMo fuel were manufactured by CERCA and for two of them silicon is added to the Al-matrix as one of the most promising solutions. The plates will be irradiated in OSIRIS reactor in Saclay (CEA France) in 2005. This paper presents the FRM-II conversion schedule and particularly describes the conditions of this first irradiation program.

Research Reactor Utillization: A Justification for Existence?

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The majority of Research Reactors currently under operation are constantly faced with critical issues relating to decisions justifying their sustainable existence. These issues may relate to aspects such as the age and related state of safe operation, levels of political or environmentalist support, financial independence with regard to operational costs, all of which, together with several other factors, could contribute to justifiable existence in terms of levels of utilisation and safety of these reactors.

This presentation will endeavour to evaluate the mix of desirable characteristics regarded as essential justification to stakeholders for the extended operation and utilisation of a research reactor. The topic centres on the IAEA recommendations in terms of established Strategic Planning regarding such facilities. As an example, the model used to drive the sustained existence of the SAFARI-1 research reactor of South Africa will be evaluated.

THE RENEWED SPIRIT OF Y-12

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The Y-12 National Security Complex began operations in 1943 as a part of the Manhattan Project that developed the first atomic weapon. In recognition of the end of the Cold War, and the new War on Terror, Y-12 is modernizing its site to better meet these changing times. Although we are proud of our place in history, after 60 years, we have begun to write a new chapter that will enable us to meet the new challenges facing the world today by strengthening our security posture and utilizing existing Y-12 expertise in nuclear non-proliferation initiatives. The modernization of Y-12 will enable us to be agile enough to adapt and respond to all of our national security needs.

Y-12 considers nuclear non-proliferation as one of its primary missions. Some of the nuclear non-proliferation programs we support include our low enriched uranium (LEU) supply to research and test reactors. The LEU we provide to the research reactor community is derived from down blending the concentration of 235U contained in U.S. surplus highly enriched uranium (HEU) that we remove from dismantled nuclear weapons. Y-12 is also involved in various non-proliferation programs in Russia, in the material removal from Libya, and various activities supporting the Global Threat Reduction Initiative (GTRI).

The Y-12 National Security Complex stores significant quantities of HEU and therefore, has a security posture that must adapt to these new threats of global terrorism. This year, Y-12 has made real progress in modernizing its site so that it is better able to meet these new world challenges. Our modernization efforts will increase security, improve productivity, minimize health and safety risks and enable the Y-12 Site to continue to operate far into the future. This abstract will summarize how Y-12 modernization will provide a safer, more secure and stable supply of uranium to research and test reactors for many years into the future, and how Y-12 will continue to support nuclear non-proliferation initiatives.

THE NEED TO ADDRESS THE LARGER UNIVERSE OF HEU-FUELED REACTORS, INCLUDING CRITICAL ASSEMBLIES, PULSED REACTORS AND PROPULSION REACTORS

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The RERTR program has focused thus far primarily on ending shipments of HEU fuel to research reactors. This has resulted in giving highest priority to reactors with steady thermal powers of 1 megawatt or more, because they require regular refuelling.

Critical facilities and pulsed reactors can also of serious concern, because some of them contain very large amounts of barely-irradiated HEU and plutonium. They could be costly to convert -- and conversion to LEU may be impractical for fast-neutron critical assemblies. An assessment should be carried out first, therefore, as to which are still needed. Critical assemblies are required today primarily to benchmark Monte Carlo neutron-transport codes. Perhaps the world nuclear community could share a few instead of each reactor-design institute having its own.

There is also a whole universe of HEU-fuelled pressurized-water reactors used to power submarines and other types of nuclear-powered ships. These reactors collectively require much more HEU fuel each year than research reactors. The risk of HEU diversion from their fuel cycles is not zero but it is difficult for outsiders to discuss conversion because of the fuel designs are classified.

This makes the conversion of Russia's civilian icebreaker reactors of particular interest because issues of classified fuel design are less problematic and these reactors load annually fuel containing about 400 kg of U-235. Another reason for interest in developing LEU fuel for these reactors is that the KLT-40 icebreaker reactor is being adapted for a floating nuclear power plant. Finally, the research-reactor community is, in any case, faced with developing fuels that can operate at power-reactor-fuel temperatures because there are a few high-powered research reactors that operate in this temperature range.

TECHNICAL, ECONOMIC AND LEGAL ASPECTS OF SHIPMENT OF RESEARCH REACTOR SPENT NUCLEAR FUEL OF RUSSIAN ORIGIN TO THE RUSSIAN FEDERATION

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The purpose of the report is the search for principal decisions aimed at realization of the Agreement between RF Government and USA Government regarding cooperation in repatriating RR SNF to the RF. Russian approaches and general ideas to implement RR SNF shipment from the technical, economical and legislative points of view are considered in the report. Russian experience and opportunities of Russian technologies are considered to implement the program stipulated in the Agreement. The solutions resulted from international shipment experience and the most advanced technologies for RR SNF handling are suggested. It is shown that currently there is no single technology coping with the whole program of shipment and within the period specified in the Agreement. The solution is in a complex approach, elaboration of mobile and flexible systems, implementation of joint and parallel shipments.

SUCCESSFUL COMPLETION OF A TIME SENSITIVE MTR AND TRIGA® INDONESIAN SHIPMENTS

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Early this year, shipments of 109 MTR fuel assemblies were received at the Department of Energy's Savannah River Site from the BATAN reactor in Serpong, Indonesia and 181 TRIGA® fuel assemblies were received at the Idaho National Laboratory from the two BATAN Indonesian Triga reactors in Bandung and Yogyakarta, Indonesia. These were the first Other than High Income Countries shipments under the FRR program since the Spring 2001.

The Global Threat Reduction Initiative announced by Secretary Abraham will require expeditious scheduling and extreme sensitivity to shipment security. The subject shipments demonstrated exceptional performance in both respects. Indonesian terrorist acts and 9/11 impacted the security requirements for the spent nuclear fuel shipments. Internal Indonesian security issues and an upcoming Indonesian election led to a request to perform the shipment with a very short schedule. Preliminary site assessments were performed in November 2003. The DOE awarded a task order to NAC for shipment performance just before Christmas 2003. The casks departed the US in January and the fuel elements were delivered at the DOE sites by the end of April 2004. The paper will present how the team completed a successful shipment in a timely manner.

MANAGEMENT OF LEU ALUMINUM-CLAD SPENT FUEL IN ARGENTINA

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At present, the research and production reactor RA-3 is the only one in Argentina that generates aluminum-clad spent fuel. It was converted to reduced enrichment uranium in 1989 using LEU-U_3O_8 fuel elements, which were developed in CNEA in the frame of the Program on Reduced Enrichment for Research and Test Reactors.

The management strategy for the RA-3 spent fuel is presented in regard to interim wet and dry storage and the treatment prior to disposal. Particularly, different alternatives are discussed in relation to the processes being considered for the treatment of the spent fuel. In principle, these processes could be adjusted for spent fuels containing different fuel materials, e.g. U_3O_8, U_3Si_2 or U-Mo. A brief description of the available facilities for the spent fuel treatment is presented.

LOGISTICS OF THE RESEARCH REACTOR FUEL CYCLE : AREVA SOLUTIONS

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The AREVA Group Companies offer comprehensive solutions for the entire fuel cycle of Research Reactors comply with IAEA standards.·

CERCA and COGEMA LOGISTICS have developed a full partnership in the front end cycle In the field of uranium CERCA and COGEMA LOGISTICS have the long term experience of the shipment from Russia, USA to the CERCA plant..Since 1960, CERCA has manufactured over 300,000 fuel plates and 15,000 fuel elements of more than 70 designs. These fuel elements have been delivered to 40 research reactors in 20 countries.

· For the Back-End stage, COGEMA and COGEMA LOGISTICS propose customised solutions and services for international shipments.COGEMA LOGISTICS has developed a new generation of packaging to meet the various needs and requirements of the Laboratories and Research Reactors all over the world, and complex regulatory framework.Comprehensive assistance dedicated, services, technical studies, packaging and transport systems are provided by AREVA for every step of research reactor fuel cycle.

A MOBILE MELT-DILUTE MODULE FOR THE TREATMENT OF ALUMINUM RESEARCH REACTOR SPENT FUEL

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A mobile melt-dilute (MMD) module for the treatment of aluminum research reactor spent fuel is being developed jointly by Savannah River National Laboratory and Argonne National Laboratory. The process utilizes a closed system approach so that volatile fission products and fission gases are retained inside the sealed stainless steel canister after treatment. The MMD will allow spent fuel to be melted and diluted with depleted uranium to an isotopic content of less than 20%. Poisons such as gadolinium can be added to the melt that will partition to the uranium bearing phase reducing engineered criticality control requirements and increasing proliferation resistance. The final ingot is solidified inside the sealed canister and can be readily stored safely either wet or dry until final disposition. The MMD module can be staged at or near the research reactor fuel storage site to facilitate treatment of the spent fuel into a stable non-proliferable form.

RESEARCH REACTOR FUEL IN AUSTRIA

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In the past decades Austria operated three research reactors, the 10 MW ASTRA reactor at Seibersdorf, the 250 kW TRIGA reactor at the Atominstitut and the 1 kW Argonaut reactor at the Technical University in Graz. Since the shut down and decommissioning of the ASTRA reactor on July 31st, 1999 and the shut down of the ARGONAUT reactor in Graz by August 31, 2004 only one reactor remains operational in Austria which is the TRIGA Mark II reactor at the Atominstitut.

The TRIGA reactor Vienna is used intensively for students education and training, all reactor systems are in excellent condition, spare fuel elements are available to operate this reactor for another 10 to 15 years and at present there is no indication whatsoever that this reactor should be closed down in the coming years.

The Argonaut reactor Graz has been shut down on August 31st, 2004 and the fuel will be returned by fall 2005 using a US transport organisation. Although its plate type fuel elements are almost inactive due to the low power level and infrequent operation the fuel shipment procedure and the costs are almost equal to shipments of higher powered research reactors. The financing of the shipment, the transport route and the time schedule has been settled in mid-August 2004.

This paper will discuss the overall situation of the remaining TRIGA research reactor fuel in Austria and the shipment of the Argonaut reactor fuel to the USA.

FUEL BURNUP MEASUREMENT OF SPENT FUEL USING GAMMA SPECTROSCOPY TECHNIQUE

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This paper presents burn up results obtained for HEU and LEU fuel assemblies using gamma spectroscopy technique. The spectrums were got from an in-pool facility built in the RECH-1 Research Reactor to be mainly used to measure the burn up of irradiated fuel assemblies with short decay periods, where 95Zr is being evaluated as possible fission monitor. A campaign for measurements all spent fuel assemblies of the RECH-1 reactor was initiated in the frame of the Regional Project RLA/4/018: “Management of Spent Fuel from Research Reactors”.

The results presented here were obtained from the in pool facility for fuel assemblies with decay period greater than 100 days and 137Cs was used as fission monitor. The efficiency of the in pool system was determined using a slightly burnt experimental fuel assemblies, which has one fuel plate (one of the outer plates) and the rest are dummy plates. An average burn up of 2.8% of 235U was previously measured for the experimental fuel assembly utilizing the burn up facility installed in a hot cell, where 137Cs was used as monitor.

This paper shows burn up results for HEU (45% of 235U) and LEU (19.75% of 235U) fuel assemblies, which had relatively long cooling times. The experimental results are showing good agreement with those obtained using neutronic codes (WIMS-CITATION).

MANAGEMENT OF SPENT FUEL FROM RESEARCH REACTORS:
BRAZILIAN PROGRESS REPORT (WITHIN THE FRAMEWORK OF THE IAEA REGIONAL PROJECT RLA-4/018)

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There are four research reactors in Brazil, and, because of the low reactor power and low burn-up of the fuel, for three of them spent fuel storage is not a problem, except for the concern about ageing. However for one or the reactors, more specifically IEA-R1 research reactor, spent fuel storage is a major concern, because, according to the new proposed operation schedule of the reactor, by the year 2009 there will be no more racks available to store spent fuel, and so far, no alternative has been proposed. This paper gives a brief description of the type and amount of fuel elements utilized in each of the reactors with a short discussion about the spent fuel storage capacity at each installation The paper also describes the activities developed by Brazilian engineers and researchers during the period 2001-2004, within the framework of Project RLA-4/018-Management of Spent Fuel From Research Reactors. During this period, ten workshops, four coordination meetings and two short courses on SCALE nuclear codes were realized. The paper starts with some considerations about the research reactors ins Brazil, the importance of the Regional Project RLA-4/018 to understand the real problem related to storage of spent fuel from research reactors, followed by a description of the activities for each working group defined in the first coordination meeting of the project, namely, characterization, options, public communication, and legislation. As a conclusion, we can say that the advances of the project, and the integration promoted among the engineers and researchers of the participant countries are a reality.

EXTENDING THE FOREIGN SPENT FUEL ACCEPTANCE PROGRAM: POLICY AND IMPLEMENTATION ISSUES

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The May 2006 expiration date of the Foreign Research Reactor Spent Nuclear Fuel (FRR SNF) Acceptance Program is fast approaching. In April 2004, Energy Secretary Spencer Abraham instructed the Energy Department to "initiate actions necessary to extend the fuel acceptance deadline." However, extending the deadline may not be a simple task. The limits on the original program resulted from a delicate negotiation among many stakeholders. Any proposal to increase the duration and scope of the program will have to be considered in the context of DOE's failure since 1996 to develop viable treatment, packaging and long-term disposal options for FRR SNF. It is also unclear whether accepting additional low-enriched uranium FRR SNF can be justified on security grounds. This paper will propose criteria for acceptance of spent fuel under an extension that are intended to minimize controversy and ensure consistency with a threat-based prioritization of homeland security expenditures.

ANL PROGRESS ON THE COOPERATION WITH CNEA FOR THE MO-99 PRODUCTION: BASE-SIDE DIGESTION PROCESS

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Conversion from high-enriched uranium (HEU) to low-enriched uranium (LEU) targets for the Mo-99 production requires certain modifications of the target design, the digestion and the purification processes. ANL is assisting the Argentine Comisión Nacional de Energía Atómica (CNEA) to overcome all the concerns caused by the conversion to LEU foil targets. A new digester with stirring system has been successfully applied for the digestion of the low burn-up U foil targets in KMnO4 alkaline media. In this paper, we report the progress on the development of the digestion procedure with stirring focusing on the minimization of the liquid radioactive waste.

THERMOXID - A NEW SORBENT FOR THE SEPARATION AND PURIFICATION OF 99MO

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The Argonne National Laboratory Reduced Enrichment for Research and Test Reactors Program is performing R & D supporting conversion of 99Mo production from high-enriched to low-enriched uranium targets. One of the major obstacles to conversion is the fivefold increase of the amount of uranium needed to produce an equivalent amount of 99Mo. The additional uranium would lead to an increase in the volume of liquid processed and the volume of liquid waste. The use of an efficient, high capacity sorbent would allow for small purification columns and minimum liquid volumes throughout the process. Thermoxid has developed an inorganic sorbent that meets these requirements. Our batch tests show that Thermoxid sorbents have much higher Kd(Mo) values than the commonly used alumina under a wide variety of conditions (20 – 340 g U / L, 0.5 – 1.5 M HNO3, 4, 24, and 48 hours) relevant to acid-side 99Mo production and recovery processes. The Kd(Mo) values for the Thermoxid sorbents are inversely proportional to both uranium concentration and acidity. Column tests were conducted to determine the sorbents' capacity for Mo at various uranium and acid concentrations. Overall, these new sorbents appear to have superior performance and would allow for smaller separation/purification columns than are possible using alumina as the sorbent.

STATUS OF ANSTO MO-99 PRODUCTION USING LEU TARGETS

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The Australian Nuclear Science and Technology Organization (ANSTO) has produced Mo-99 using Low Enriched Uranium (LEU) UO2 targets for nearly thirty years. The Replacement Research Reactor (RRR) provides ANSTO with a good opportunity to review and improve the current Mo-99 production process. Uranium target design improvements were performed through a collaborative effort with the U.S. Department of Energy (DOE) to promote the use of LEU in research reactors. Annular cans targets including U foils were first designed, developed, and tested in cooperation with CERCA.

This paper, presents the latest results and conclusions of this program.

DEVELOPMENT OF THE FABRICATION TECHNOLOGY OF WIDE URANIUM FOILS FOR MO-99 IRRADIATION TARGET BY COOLING-ROLL CASTING METHOD

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An alternative fabrication method for polycrystalline uranium foils has been investigated using a cooling-roll casting method in KAERI since 2001, in order to produce a medical isotope 99Mo, the parent nuclide of 99mTc. The fabrication method of wide uranium foils produced by cooling-roll casting has been optimized to improve the quality of uranium foils and the economic efficiency of the foil fabrication with the modifications of the casting apparatus and the variations of the various process parameters. The injection control device of the uranium melt was applied to cooling-roll casting apparatus, in order to stabilize the fabrication process and to increase the yield of uranium foils through the prevention of the melt leakage. As the uranium has a low thermal conductivity, the collection apparatus was modified to fabricate the uranium foils without great defects soundly, leaded to improve the quality and the yield of the uranium foils. The dimension and the surface state of the uranium foils were also adjusted with the revolution speed of cooling roll, the ejection pressure of melt, the gap distance between nozzle slot and cooling roll, the superheat of the metal, and the atmosphere of melting and casting. Then, continuous polycrystalline uranium foils with a thickness range of 100 to 150? and a width of about 50 mm were fabricated with a better quality of uranium foils and a higher economic efficiency of the foil fabrication, through the modifications of the casting apparatus and the variations of the various process parameters.

GTRI AND CONVERSION OF ISOTOPE PRODUCTION TO LEU TARGETS

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The Global Threat Reduction Initiative was proposed in 2004 by U.S. Secretary of Energy Spencer Abraham to reduce risks from worldwide storage and use of weapon-usable nuclear materials – plutonium and highly enriched uranium (HEU). A key part of the initiative is converting the production of medical isotopes from reliance on HEU targets to alternative targets using low-enriched uranium (LEU), unsuitable for weapons. The main technical challenge to conversion has been surmounted, leaving political and economic hurdles. This paper reviews recent developments, including: U.S. government initiatives to promote conversion; Canada's stalled conversion program; prospective development of LEU-based isotope production in the United States; and industry lobbying efforts to rescind U.S. restrictions on HEU exports. The paper concludes with recommendations for expediting the conversion of isotope production to LEU targets.

MAKING OF FISSION 99Mo FROM LEU SILICIDE(S): A RADIOCHEMISTS VIEW

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The present-day industrial production of 99Mo is fission based and involves thermal neutron irradiation of highly enriched uranium, HEU, (20 % < 235U = 93 %) containing targets in research reactors, followed by radiochemical processing of the irradiated targets resulting in the final product, namely a 99Mo containing chemical compound of molybdenum.

In 1978 a program (RERTR) was started to develop and substitute HEU reactor fuel by low enriched uranium, LEU, (< 20 % 235U) fuel. In the wake of that program studies were undertaken to convert HEU into to LEU based 99Mo production. New targets and radiochemical separations of 99Mo were proposed; one of these is the now somewhat neglected but potential LEU silicide (U3Si2) target.

This paper aims at comparing two LEU uranium silicides with some other LEU target materials/targets and arriving at some preferences pertaining to 99Mo production.

MO-99 PRODUCTION ON A LEU SOLUTION REACTOR

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A pilot homogenous reactor utilizing LEU has been developed by the Kurchatov Institute in Moscow along with their commercial partner TCI Medical. This solution reactor operates at levels up to 50 kilowatts and has successfully produced high quality Mo-99 and Sr-89. Radiochemical extraction of medical radionuclides from the reactor solution is performed by passing the solution across a series of inorganic sorbents. This reactor has commercial potential for medical radionuclide production using LEU U2SO4 fuel. Additional development work is needed to optimize multiple 50 kilowatt cores while at the same time, optimizing production efficiency and capital expenditure.

Posters

TWENTY-FIVE YEARS SUPPORTING RERTR ACTIVITIES FROM CNEA MTR FUEL FABRICATION PLANT (ECRI)

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CNEA, the Atomic Energy National Commission of Argentina, has been participating in the RERTR Program effort since around twenty-five years ago. Most of this participation and support is related with fuel development and fuel manufacturing activities.

This paper describes the contributions from CNEA fuel related facilities. They comprise R&D and manufacturing activities. R&D started in the early eighties with the development and fabrication of miniplates for the international silicide fuel qualification effort. At present the R&D activities continue with the development of fabrication techniques and procedures suitable for the manufacturing of fuel plates using the U-Mo based alloys. The above mentioned facilities include also a fuel power production plant and a facility for fuel scrap treatment.

In the MTR fuel manufacturing field the contribution began with the fabrication of the fuel elements for the conversion of the Argentine RA-3 Reactor. LEU fuel elements were also produced for several reactors outside Argentina. Among them the core for the RP-0 facility in Perú, the conversion of the Iranian Reactor, the fuel for the Algerian Reactor and more recently the fuel for the MPR in Egypt. All these fuels use U3O8 as fissile material.

CNEA has qualified also as fuel manufacturer of fuels bearing silicide compounds. At present CNEA fuel fabrication plant is taking the manufacture of 64 fuel elements for the ANSTO Replacement Research Reactor in Australia.

As a result of the above mentioned activities more than 12000 fuel plates were fabricated and most of them have been successfully irradiated without any fuel failure reported.

CNEA contribution also covered the development and fabrication of targets for radio-isotope production. Argentina has converted the production of isotopes for medical applications to LEU in 2002. Up to present more than 600 targets bearing LEU aluminide have been fabricated irradiated and processed. Very recently CNEA has also become an international supplier of targets for Mo-99 production.

THREE-DIMENSIONAL MONTE CARLO NEUTRON TRANSPORT SIMULATION OF THE GHANA RESEARCH REACTOR-1

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Stochastic Monte Carlo neutron particle transport methods have been successfully applied to model in 3-D, the HEU-fueled Ghana Research Reactor-1 (GHARR-1) a commercial version of the miniature Neutron Source Reactor (MNSR) using the MCNP version 4c3 particle transport code. The preliminary multigroup neutronic criticality calculations yielded a k_{eff} = 1.00449 with a corresponding cold clean excess reactivity of 4.47 mk (447 pcm). The Monte Carlo simulations also show comparable results in the neutron fluxes in the HEU core and some regions of interest. The observed trends in the radial and axial flux distributions in the core, beryllium annular reflector and the water region in the top shim reflector tray were reproduced, indicating consistency of the results, accuracy of the model, precision of the MCNP transport code and the comparability of the Monte Carlo simulations. The results further illustrate the close agreement between stochastic transport theory and the experimental measurements conducted during off-site zero power cold tests.

RECH-1 TEST FUEL IRRADIATION

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Since May 2003, one RECH-1 fuel element has been submitted to irradiation at the HFR-Petten, Holland. This paper presents the objectives and progress up to date of this fuel qualification under irradiation.

Besides, a brief description of CHI/4/021, IAEA's Technical Cooperation Project that has supported this irradiation test, is also presented here.

A MODIFIED NITRIDE-BASED FUEL FOR RESEARCH REACTORS

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A modified nitride-based uranium fuel to support the small, secured, transportable, and autonomous reactor (SSTAR) concept is initiated at Lawrence Livermore National laboratory (LLNL). This fuel and material research project centers on the evaluation and manufacturing of uranium nitride fuel imbedded with other inert (e.g. ZrN) or neutron-absorbing materials (e.g. HfN) to enhance the fuel properties to achieve long core life. This paper discusses how a modified nitride fuel with chemically-compatible inert additives (ZrN, HfN, etc) could be suitable as replacement fuel for research and test reactors.

Mono-uranium nitride fuel pellet is manufactured at the LLNL. Existing facilities and equipment can be employed to fabricate modified uranium nitride fuel cladded in aluminum cladding. Preliminary fuel examination indicated that high uranium loading can be achieved in uranium nitride: at 80 theoretical density, 10.8 g/cc is uranium. Uranium nitride is also favorable in its thermal properties: the thermal conductivity of mono-nitride is compatible to that of silicate (~25 W/mK), and its melting temperature is much higher than that of metal (2630 ^{o}C for UN vs. 1100 ^{o}C for U metal). Out-of-pile experiment is planned to examine the corrosion properties of uranium nitride fuel in water coolant.

POSSIBILITIES TO USE TRIGA STEADY STATE REACTOR FOR RIA TESTS

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Studies were performed to investigate the possibilities of power pulse tests preirradiated fuel elements in Loop A irradiation device, in the following conditions :

steady state initial average linear power between 400 and 600 W/cm

coolant pressure in the range 5 MPa to 9 MPa and temperature in the range 265 °C to 300 °C

increase of the power between 6 and 10 times in 1 second with pulse half-width in the range 1 to 2 seconds

power reduced to shutdown levels by 5 seconds after the beginning of the pulse.

deposited energy in the range 90 cal/gUO2 to 210 cal/gUO2 in a five seconds period starting with the initiation of the power pulse.

The following analyses were conducted :

range of steady state linear average power rate on the test fuel as a function of initial enrichment and reactor power level for a 200 MWh/kgU pin.

transient simulations, done with PARET computer code in order to investigate the time shape of a power pulse resulted from a reactivity insertion in TRIGA SSR reactor and the corresponding energy deposition in the fuel pin, five seconds from transient inception.

ANALYSIS OF PARTIAL AND TOTAL FLOW BLOCKAGE OF A SINGLE FUEL ASSEMBLY OF AN MTR RESEARCH REACTOR CORE

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The lack of full understanding of complex mechanisms connected with the interaction between thermal-hydraulics and neutronics still challenge the design and the operation of nuclear reactors by the adoption of conservative safety limits. The recent availability of powerful computer and computational techniques together with the continuing increase in operational experience imposes the revisiting of those areas and eventually the identification of design/safety requirements that can be relaxed. Currently, the enlarged commercial exploitation of nuclear Research Reactors (RR) has increased the consideration to their corresponding safety issues. These later were so far been performed using conservative computational tools. Nowadays, the application of Best-Estimate (BE) methods constitutes a real necessity in order to increase their commercial productivity. For this purpose, as typical representative of research reactors, the IAEA 10 MW MTR Research Reactors problem is considered to investigate one of the most severe accident that may occur during a RR life time. The transient under consideration is a total and partial blockage of the cooling channel of a single Fuel Assembly. Such event constitutes a severe accident for this type of reactor since it may lead to local dry-out and eventually to loss of the FA integrity. Two cases are analyzed to emphasize the severity of the accident. The first one is a partial blockage of a single FA while the second one is an extreme scenario consisting of total blockage of the same FA. As a first attempt the calculations are performed by applying the BE thermal-hydraulic system code RELAP5. This study constitutes the first step of a larger work which consists in performing a 3D simulation using the Best Estimate coupled code technique. However, as a first approximation the instantaneous reactor power is derived through the point kinetic approach.

AN INDIAN PERSPECTIVE FOR TRANSPORTATION AND STORAGE OF SPENT FUEL

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The spent fuel discharged from the reactors are temporarily stored at the reactor pool. After a certain cooling time, the spent fuel is moved to the storage locations either on or off reactor site depending on the spent fuel management strategy. As India has opted for a closed fuel cycle for its nuclear energy development, reprocessing of the spent fuel, recycling of the reprocessed plutonium and uranium and disposal of the wastes from the reprocessing operations forms the spent fuel management strategy. Since the reprocessing operations are planned to match the nuclear energy programme, storage of the spent fuel in ponds are adopted prior to reprocessing. Transport of the spent fuel to the storage locations are carried out adhering to international and national guide lines.

PREDICTION OF FLOW INSTABILITY DURING NATURAL CONVECTION

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The occurrence of flow excursion instability during passive heat removal for Tehran Research Reactor (TRR) has been analyzed at low-pressure and low-mass rate of flow conditions without boiling taking place. Pressure drop-flow rate characteristics in the general case are determined upon a developed code for this purpose. The code takes into account variations of different pressure drop components caused by different powers as well as different core inlet temperatures. The analysis revealed the fact that the instability can actually occur in the natural convection mode for a range of powers per fuel plates at a predetermined inlet temperature with fixed geometry of the core. Low mass rate of flow and high sub-cooling are the two important conditions for the occurrence of static instability in the TRR. The calculated results are compared with the existing data in the literature.

HIGH FLUX REACTOR EVOLUTIONS AND IMPROVEMENTS

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Following the changes over the years in experimental and safety requirements at the ILL a great deal of work has been carried out on the installations :

- In 1985, a new cold source was installed, allowing the production of ultra-cold neutrons via a vertical channel.

- From 1991 to 1995 the reactor block was replaced, allowing us to perform reactivity calculations and determine other neutronic values.

- In 2003, a new hot source was installed with three beam tubes viewing it; the new system is now operating very efficiently.

- This year a major beam tube is to be replaced with a new zircaloy tube.

- And finally, from 2003 to 2006, the facility is being upgraded significantly to withstand newly-defined safe-shutdown earthquakes.

In parallel, developments are on-going on the efficiency of the instruments and the neutron guides under the Millennium Programme. These will result in overall gains in data collection of over a factor of 10. As the ILL's international convention has been extended to the end of 2013 the Institute is therefore now well-set to maintain its position as a centre of excellence in the scientific use of slow neutrons for the twenty years to come.

CORRELATION OF FAST NEUTRON FLUENCE AND POISONING OF BERYLLIUM BLOCKS IN THE MARIA REACTOR

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In the paper, the determination of fast fluence on the basis of computed He-3 and H-3 contents in beryllium blocks of the MARIA reactor is described. In beryllium irradiated with neutrons, isotopic transformations starting from the Be(n,2n) and Be(n,a) reactions cause beryllium damage. The reactions causing the damage lead also to the build-up of strongly absorbing Li-6 and He-3. The presence of the isotopes changes significantly the neutronic characteristics of the reactor and as such has been a subject of thorough analysis during the last years, using the REBUS code. The distribution of the absorbing isotopes in all beryllium blocks has been obtained through detailed fuel management calculations including the whole history of reactor MARIA operation. A correlation method has been developed to assess MARIA beryllium fluence on the basis of H-3 and He-3 block-wise REBUS distributions. The method relies heavily on fact that the Be (n,a) reaction has a nonzero cross section above 0.5MeV, i.e. only the fast neutron flux and hence the fast fluence is responsible for the damage. The values of fluence obtained using the correlation have been compared with the results of the fluence obtained using semi-empirical evaluation.

CONCEPTUAL ANALYSIS OF THE FUEL MANAGEMENT STRATEGY FOR THE RA-3 RESEARCH REACTOR AT 10 MW

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The Argentine Research Reactor RA-3 was designed to produce radioisotopes and it operates with LEU (U3O8) fuel since 1990. Its initial power was 5 MW and it has recently been upgraded to 10 MW. The National Atomic Energy Commission (CNEA) is both its owner and operator.

At the beginning of this year, the Nuclear Regulatory Authority extended its operation license to an authorised power of 10 MW after a series of modifications and tests carried out by the installation during 2002 and 2003.

As a consequence of this power increase, the installation introduced some non-systematic modifications in its fuel management strategy with the purpose of preserving the operation period in 20 days approximately. The main change was to load 2 fuel elements per cycle in some cycles (instead of 1 as it used to be at 5 MW).

The purpose of this work is to perform a conceptual analysis of possible fuel management strategies for the RA-3 reactor, that could provide quantitative elements for a safety assessment, as well as to evaluate the fuel management flexibility at 10 MW in compliance with standards in force.

It is concluded that operation at 10 MW with a 2 FE/cycle strategy leads to a significant excess reactivity at the beginning of cycle, but still in compliance with the margins established by the standards of application.

PHOTON SOURCE AND SHIELDING STUDIES RELATED TO THE SPENT FUEL STORAGE OF THE RA REACTOR

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The planning and activities related to the safe transport of spent fuel elements from the RA reactor at Vinca Institute to the reprocessing plant, performed during last few years, have resulted in development of valid methods for radiological characterisation of spent fuel and shielding analysis of storage containers and transport casks with spent fuel elements of the RA reactor.

Determination of the isotopic composition of the materials present in spent fuel of the RA reactor and subsequent derivation of the heat generation and radiation source terms has been the first study performed. The complexity of this study arises from the end region effect between fuel elements placed inside the fuel channel. This paper describes three procedures, being prepared for the analysis of spent fuel characteristics (radiation sources, decay heat, and spent fuel isotopic composition) of the RA reactor. The first procedure is based on the application of the SAS2H control module from the SCALE-4.4a code system and an approximate geometrical model. The second computational tool employed consists of the MCNP-4C and ORIGEN2.1 codes interfaced by the MOCUP driver. Recently developed, the third procedure is based on the applications of the KENO-V.a and ORIGEN2.1 codes.

Results representing the validation of the methods and geometrical models are included and radiation characteristics evaluations (radiation sources) of spent fuel slugs of the RA reactor are presented. Analysis of those results has allowed for further discussion in choosing the appropriate values of the radiation source terms needed for the shielding investigations of the stainless steel and aluminium containers with spent fuel slugs of the RA reactor.

In this paper, the design oriented shielding analysis was performed with the standard calculation model available in the SAS4 sequence of the SCALE-4.4a code system, which is based on the MORSE-SGC, a Monte Carlo code with the (27 neutron – 18 gamma) coupled energy groups cross-section data library. The calculation model used assumes that the materials of fuel assemblies and their sources have been smeared over the horizontal cross section of the fuel assembly.

For reference shielding calculation the MCNP-4C, a Monte Carlo code with proton cross section data library MCPLIB (designed by identifiers ending in .02p) and detailed geometry of spent fuel containers were used. As an example of validation work, the case with 30x6 spent fuel slugs in the aluminium barrel surrounded with 100 cm of light was analysed. The results obtained under assumption that all fuel slugs were irradiated at 6000 MWd/t (average fuel burnup of the RA reactor spent fuel with 2% 235U enriched metal uranium) and cooled for 5 years.

CNEA EXPERIMENTS TO APPLY FSW TO ENCASE U-MO FOIL IN ALUMINIUM

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U-Mo dispersed fuel can not fulfil the U density requirements for high flux reactors. Besides, in pile experiments have shown a high degree of interaction between the fuel particles and the matrix generating large porosity. Replacing dispersed fuel by an U-Mo foil has been proposed and is being investigated as a possible solution to these problems. Friction stir welding (FSW) is being experimented at ANL to fabricate fuel plates.

Supported by the ENDE - CAC - CNEA large experience in FSW, experiments have been started. Successful weldings have been obtained between Al and stainless steels as subrogate material instead U-Mo. The first objective is the implementation of the process, optimization of operative parameters and selection of the adequate tests to qualify the welding. The second objective is encase U-Mo foil in Al to be used as diffusion couples.

SET UP OF U-Mo POWDER PRODUCTION BY HMD PROCESS

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An industrial process for the fabrication of U-Mo powder have been set up with the development of the massive hydryding of uranium-molybdenum alloys in gamma phase at low temperature and pressure . The brittleness of the hydride allows performing conventional milling and dehydryding for obtaining U-Mo powder (hydryding-milling-dehydriding: HMD process).

The whole process involves the melting of the alloy, fractionating the alloy into lumps, two steps hydriding, size reduction of hydride by crushing, low impact milling, dehydriding and a very high temperature treatment. All this processes are performed in a licensed facility with the capability of working with enriched uranium.

The different steps in the process have been holly understood and optimized. Melting of the alloy is intended to be done in an arc furnace to reduce scrap and eliminate the fractionating step. The capability of massive hydriding in the gamma phase essentially depends on the previous introduction of hydrogen in traps that is performed at a slightly higher temperature than the intersticial filling. Previous crushing of the hydride is needed for the introduction of controlled size particles to the low impact mill. One pass threw the atmosphere controlled low impact mill produces specification size powder. A controlled temperature and pressure dehydriding is performed to avoid flying powder. A final step of high temperature treatment is performed to reduce even more remnant hydrogen, reduce oxide layer, homogenize composition, eliminate interior defects, increase density and with cooling, passivate the final product.

Known data of the whole process will be presented so as to thoroughly evaluate fabrication capabilities, performance and costs.

CORROSION OF SPENT NUCLEAR FUEL ALUMINIUM CLADDING IN ORDINARY WATER

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The Vinca Institute of Nuclear Sciences, Belgrade, Serbia and Montenegro, participates in the IAEA CRP on “Corrosion of Research Reactor Aluminium-Clad Spent Fuel in Water – Phase II”. Five racks were received and immersed in the RA reactor spent fuel storage pool near containers with spent fuel elements and were exposed to water influence for period of 6 months to 6 years.

All test racks are treated and examined visualy and under microscope according to the IAEA Test protocol. Crevice and pit corrosion covers both sides of aluminium coupons. Intensive corrosion processes at aluminium surfaces under ceramic rings and galvanic corrosion effects at aluminium coupons coupled sides to stainless steel coupons were noted. Uneven oxidation with pits of different shape and size are found.

It is believed that the final results of the Vinca’s studies will contribute to management and storage practices at research reactor interim spent fuel wet storage facilities.

A NEUTRONIC FEASIBILITY STUDY FOR HEU-LEU CONVERSION OF THE REACTOR PIK

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Earlier (Bariloche, 2002) we have shown the possibility of the conversion of reactor PIK from HEU(90%) to MEU(36%) fuel. Monte Carlo calculations have demonstrated that the [UO2(25vol.%)+Cu] meat with density 2.2gU/cm3 could be changed for the [UO2(38vol.%)+Al] meat with 3.6gU/cm3 with noticeable gain in reactor neutronics. The following attempt to reach LEU(19.75%) conversion with UMo(9w.%) meat at Al cladding (instead of stainless steel) failed due to the strains and stresses, which arise in the FE during the working cycle. There is a possibility of the cladding peeling off the meat.

In this talk we suggest for the PIK to use [UMo(9w.%)60vol.%+Mg] meat with 9.4gU/cm3 and stainless steel cladding. Thin tubes with such meat were manufactured and used during fifty years as 6% enriched fuel elements in the first Nuclear Power Station of Russia. We plan to perform the Monte Carlo simulations for such PIK-4 FE. In the case of success it would be the first example of HEU-LEU conversion of high-flux reactor without change of core geometry.

THERMOMECHANICAL DART CODE IMPROVEMENTS FOR LEU VHD DISPERSION AND MONOLITHIC FUEL ELEMENT ANALYSIS

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A collaboration agreement between ANL/USDOE and CNEA Argentina in the area of Low Enriched Uranium Advanced Fuels has been in place since October 16, 1997 under the "Implementation Arrangement for Technical Exchange and Cooperation in the Area of Peaceful Uses of Nuclear Energy. An annex concerning DART code optimization has been operative since February 8, 1999.

Previously, as a part of this annex a visual thermal FASTDART version and also a DART TM thermomechanical version were presented during RERTR 2002 and RERTR 2003 Meetings. During this past year the following activities were completed:

Optimization of DART THERMAL code Al diffusion parameters by testing predictions against reliable data from RERTR experiments.
Improvements on the 3-D thermo-mechanical version of the code for modeling the irradiation behavior of LEU U-Mo monolithic fuel

Concerning the first point, by means of an optimization of parameters of the Al diffusion through the interaction product theoretical expression, a reasonable agreement between DART temperature calculation with reliable RERTR PIE data was reached.

The 3-D thermomechanical code complex is based upon a finite element thermal-elastic code named TERMELAS, and irradiation behavior provided by the DART code. An adequate and progressive process of coupling calculations of both codes at each time step was reached. The coupling of the various components of the calculation was benchmarked and validated against RERTR PIE data.
Various results will be shown during RERTR2004 meeting

NUCLEAR PARAMETERS AT START UP FRM-II

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In 2004 the research reactor FRM-II in Garching started operation using U_3Si_2 high-density fuel. March 2nd first criticality and in August full power was reached. During the start up measured nuclear data agree very well to the precalculated values. This paper presents some of these comparisons as there are the first critical condition of the reactor and the power density variation in the core for the fresh element. Excellent agreement of the measured and precalculated power density variation was found; this includes smaller variations induced by experimental installations. Till Juli 26nd, in the start up phase 88 MWd were operated with a maximum power of 13MW; the precision of the power level so far was expected to be not more than 10%. The measured change in the control rod position for new criticality on Juli 26nd was 3.05 cm; the postcalculated value for the different power steps was 3.18 cm; this corresponds to a renormalisation of the reactor power by 4%. The same result came from the more exact heating margin of the primary circuit measured at nearly full power in August. With this actual validation of the fuel burn up procedures, taken for the design of the FRM-II, the expected lifetime of more than 52 days was given new evidence.

THE METAL AND CONCRETE CASK FOR SNF AND ITS RADIATION PROTECTION QUALITY g-CONTROL

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The transportation and packing module on the basis of metal and concrete cask for the long-term storage and shipment of the spent nuclear fuel is developed in compliance with the requirements of the national standards and IAEA recommendations. Such wares designed for the NPP and submarine reactors fuel may be remade also for the research ones.

A procedure and remote device to control radiation protection of this cask equally its integrity checks after dynamic testing also is described.

SPECIFICITY IN THE LICENSING PROCESS OF REDUCED ENRICHMENT IN THE BULGARIAN RESEARCH REACTOR

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The research reactors core conversion to low enriched (LEU) fuel and utilization of a newly developed fuel will be the essential challenge to the licensing and operation of these reactors. Some risk of suspending the operating license of the older research reactors can be provoked by their inability to meet new safety requirements. Many research reactors, which are under extended shut down conditions, also can be converted to the low enriched fuel but the conversion process should be adapted to the current state of the reactor installation. This leads to some specificity in the licensing process of reduced enrichment for such reactors.

The presented paper considers some specific questions of the licensing process regarding the reconstruction of the Bulgarian research reactor, which includes conversion to the low enriched fuel. This specificity has raised as a result of the following facts:

- permit for a design of the reactor reconstruction with highly enriched fuel;
- shipment of the fresh highly enriched fuel in Russia in the framework of the RERTR programme;
- storage of the highly enriched spent fuel at reactor site;
- lack of information concerning a new low enriched fuel.

These facts might involve some changes in both – in licensing and design processes. Re-analysis of the neutronic and thermal-hydraulic calculations has to be made on the base of the technical specifications of the new LEU fuel. To facilitate the licensing process the NRA has adopted regulatory acceptance criteria for approval of the reactor core.