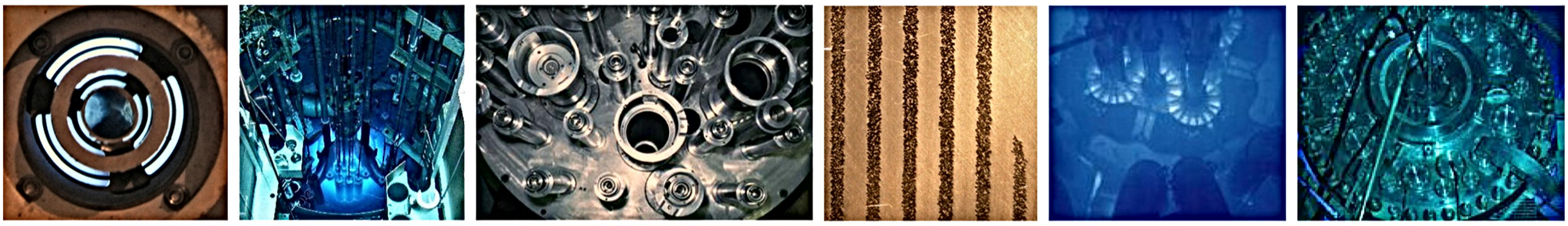
RERTR-2014

35th International Meeting on Reduced Enrichment for Research and Test Reactors



October 12-15, 2014
IAEA, Vienna, Austria

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RERTR-2014: 35th International Meeting on Reduced Enrichment for Research and Test Reactors

#	Session Title	Time	Paper Title	Presenter
Sunday, October 12 Registration: 3:00 – 6:00 pm, Vienna Marriott Hotel (Floor 1) Welcome Reception: 6:00 – 8:00 pm, Ballrooms A & B				
Monday, October 13 Meeting Room: IAEA M-Building (Level – M01) Conference Room M1				
	Opening and Welcome Jordi Roglans-Ribas	9:30 am	Welcome to the IAEA and RERTR-2014 International Meeting	J.C. Lentijo (IAEA)
1	**Global Progress in HEU Minimization** Chaired by J. Halse	9:45 am	1. NNSA's Role in HEU Minimization	A. Atkins (US-DOE NNSA)
			2. Conversion of ARGUS to Operation with LEU and IRT-3M LEU Fuel Qualification Progress	N. Arkhangelsky (SAEC-"Rosatom")
			3. Efforts Made for the Conversion of Ghana's MNSR to LEU Fuel	H.C. Odoi (NNRI)
			4. Analysis of Jamaican SLOWPOKE-2 Research Reactor for the Conversion from HEU to LEU Fuel	H. Dennis (ICENS)
			5. IAEA Activities in Support of HEU Minimization: 2014 Update	F. Marshall (IAEA)
			6. HERACLES - U-Mo Fuel Qualification in Europe	Y. Calzavara (HERACLES)
			7. U-Mo Monolithic Fuel Development for Conversion of High Performance Reactors	C. Landers (US-DOE NNSA)
12:15 pm Lunch Break				
2	**Fuel Utilization and Disposition** Chaired by N. Iyer	1:15 pm	1. Overview of Environmental Management Nonproliferation and Highly Enriched Uranium Minimization Mission Activities	E. DeLeon (US-DOE NNSA)
			2. Full Core Conversion and Operational Experience with LEU Fuel of the DALAT Nuclear Research Reactor	K.C. Nguyen (DALAT)
			3. Crossing the Finish Line: Ending Civil Use of HEU	M. Pomper (JMCNS)
			4. IAEA Cooperation with the RRRFR Programme: 2014 Update	S. Tozser (IAEA)
			5. Potential German Highly Enriched Uranium (HEU) Pebble Bed Fuel Disposition at SRS	M. Maxted (US-DOE SRS)
3:00 pm Coffee Break and Refreshments, M1 Poster Area				
3	**Poster Session I: HEU Minimization** Chaired by J. Holland	3:00 – 4:00 pm	1. Successful Operation of WWR-SM Research Reactor after Conversion to LEU Fuel	F. Kungurov (INP-KZ)
			2. Development of Low Enriched Uranium Targets for Mo-99 Production	A.L. Izhutov (SSC RIAR)
			3. Feasibility of Conversion to LEU-based Reactor Production of Mo-99	A.L. Izhutov (SSC RIAR)
			4. LEU Transition Core Optimization for the WWR-M Research Reactor in Ukraine	Y. Mahlers (INR)
			5. Brief History of MARIA Conversion From HEU to LEU	M. Migdal (NCNR)
			6. Construction of a New LEU Radioisotope Production Fission Plant in Argentina	D. Cestau (CNEA)
			7. Recovery of Uranium-Thorium from HTGR Fuel Using Salt-based Graphite Digestion	R. Pierce (SRNL)
4	**Perspectives on Fuel Development and Performance Issues** Chaired by S. Van den Berghe	4:00 pm	1. U.S. Progress in U-Mo Monolithic Fuel Development	M.K. Meyer (INL)
			2. Examination of High Uranium Density Research Reactor Fuel Performance and Endurance	G. Hofman (ANL)
			3. U.S. High Performance Research Reactor LEU Conversion Design Parameters	E.H. Wilson (ANL)
			4. Progress on the Development of U-Mo Fuel for Qualification in Korea	Y. J. Jeong (KAERI)
			5. Closing Nuclear Security Gaps: International Cooperation in LEU Fuel Development and HEU Minimization Norms	M.R. Burnett (PNNL)
5:30 pm Adjourn				

Tuesday, October 14 Meeting Room: IAEA M-Building (Level – M01) Conference Room M1				
5	**MNSR and Japan Conversions to LEU Operation** Chaired by J. Dix	8:30 am	1. GTRI Role in MNSR and Japan Conversions	B. Waud (US-DOE NNSA)
			2. The Physics Experimental Study for Prototype MNSR with LEU Core	Y. Li (CIAE)
			3. Status Report of Activities for the Core Conversion of Nigeria MNSR to LEU	S.A. Jonah (CERT)
			4. Utilization of Low-Enriched High Density Fuel at Dry Cores of Kyoto University Critical Assembly - Current Progress of the Feasibility Study	H. Unesaki (KURRI)
			5. Conversion of the KUCA "Type-A" Cores to LEU Fuel Preserving Reactivity and Central Flux Spectra	J.A. Morman (ANL)
10:00 am Coffee Break, M01 Coffee Corner *(Self-Hosted)*				
6	**High Performance Reactor Conversions** Chaired by E. Wilson	10:15 am	1. Safety Analysis of U-Mo LEU Fuel with Unfinned Cladding for the MIT Research Reactor	T. Newton (MIT)
			2. Continuing LEU Conversion Activities at the High Flux Isotope Reactor	D.G. Renfro (ORNL)
			3. Accident Analyses for the Conversion of the University of Missouri Research Reactor from Highly-Enriched to Low-Enriched Uranium	L. Foyto (MURR)
			4. Planning for the Conversion of the NIST Center for Neutron Research to LEU from HEU	D.S. O'Kelly (NIST)
			5. Enhanced Low-Enriched Uranium Fuel Element for the Advanced Test Reactor	S.R. Morrell (INL)
			6. Plasma Sprayed Zirconium for US HPRR LEU Conversion Fuel Diffusion Barrier	K. Hollis (LANL)
12:15 pm Lunch Break				
7	**Fuel Development – Irradiation Testing, PIE Analysis and Modeling** Chaired by H. Breitkreutz	1:15 pm	1. Design of the MP-1 Experiment for Irradiation in the Advanced Test Reactor	I. Glagolenko (INL)
			2. IVG.1M - LEU Fuel Test Plan	A.D. Vurim (IAE-NNC)
			3. SEM Characterization of U-7Mo Irradiated to High Fission Density at Relatively High Power, High Temperature, and High Fission Rate	D.D. Keiser (INL)
			4. Capabilities Developed for Measurement of Thermal Conductivity and Fission Gas Release of Irradiated Nuclear Fuels at PNNL	A.J. Casella (PNNL)
			5. Creep and Mass Relocation of U-Mo/Al Dispersion Fuel Meat During Mini-plate Irradiation	Y.S. Kim (ANL)
3:00 pm Coffee Break and Refreshments, M1 Poster Area				
8	**Poster Session II: Fuel Development and Fabrication** Chaired by J. Holland	3:00 – 4:00 pm	1. Ultrasonic Testing of Dispersion Type Fuel Miniplates Manufactured with Hydrided U-Mo Powder	M. Barrera (CCHEN)
			2. Characterization of Si-coated U-Mo Fuel Particles before and after Interaction Annealing	L. Olivares (CCHEN)
			3. Thermal Conductivity of In-pile Irradiated AFIP-1 Dispersion U-Mo Fuel	T. K. Huber (LANL)
			4. UAlx Plate Production: Analysis of Intermetallic Growth in UAl2/Al	B. Stepnik (AREVA-CERCA)
			5. Thermal Conductivity of U-Mo/Al Dispersion Fuel: Effects of Particle Shape and Size, Stereography, and Heat Generation	T.W. Cho (UNIST)
			6. Fabrication Procedures for Manufacturing UMo-Al Dispersion Fuel at IPEN	M. Durazzo (IPEN-CNEN)
			7. Metal Coating on Atomized U-Mo Particles to Suppress Interdiffusion between U-Mo/Al	J.M. Park (KAERI)
			8. Residual Stress Measurement for Highly Radioactive Samples	D.E. Dombrowski (LANL)
			9. Bonding Toughness Measurements in LEU Fuel Plates	D.E. Dombrowski (LANL)
			10. Intelligent Integrated Machining: Zr Thickness Measurements Using XRF for Process Control and Quality Assurance	D.E. Dombrowski (LANL)
			11. Heavy Ion Irradiation on U-Mo/Al Layer Systems: Dependence of IDL Thickness on Irradiation Temperature and Particle Flux	H. Breitkreutz (FRM2)
			12. Fuel Fabrication Process Optimization and Alternative Fabrication Development	D.M. Paxton (PNNL)

8	**Poster Session II** (Continued)		13. Design of a Full-Size Fuel Plate Irradiation for Monolithic Fuel Qualification	M. Meyer (INL)
			14. Xe irradiation on ZrN-coated U-Mo/Al dispersion fuel	Y.S. Kim (ANL)
9	**Russian-Designed Reactor Conversions** Chaired by V. Brusilovsky	4:00 pm	1. The Russian ARGUS Solution Reactor HEU-LEU Conversion: LEU Fuel Preparation, Loading and First Criticality	M. Sergey (NRC-KI)
			2. Current Status of Conversion at the WWR-K Research Reactor	A. Shaimerdenov (INP-KZ)
			3. Progress in Safety Assessment of the IR-8 Reactor During Conversion to LEU Fuel	Y. Pesnya (NRC-KI)
			4. HEU/LEU IGR Reactor Kinetics	R.A. Irkimbekov (IAE-NNC)
5:30 pm Adjourn				
Wednesday, October 15 Meeting Room: IAEA M-Building (Level – M01) Conference Room M1				
10	**Safety Analysis** Chaired by L.E. Kokajko	8:30 am	1. Comparative Safety Analysis of the MIR.M1 Reactor with Reference to Two Types of Low Enriched Fuel	A.L. Izhutov (SSC-RIAR)
			2. Analysis of Beyond Design Basis Accident for Conversion of IRT MEPhI Research Reactor to LEU Fuel	N.A. Hanan (ANL)
			3. Analysis of Beyond DBA Consequences of the IR-8 Reactor Primary Pipes Rupture during Conversion to LEU	A. Sidorenko (NRC-KI)
			4. Loss-of-Flow Simulations for the Conversion of BR2 to Low Enriched Uranium Fuel	J.R. Licht (ANL)
10:00 am Coffee Break, M01 Coffee Corner *(Self-Hosted)*				
11	**Fuel Development – Fabrication Technology** Chaired by C. Jarousse	10:15 am	1. Fabrication and Qualification of U_3Si_2 LEU Fuel Assemblies with Extrusion Technology for High Flux Reactor Petten	F.C. Klaasen (NRG)
			2. A Review of Fabrication Technologies for the MP-1 Experiment	D.E. Burkes (PNNL)
			3. Y-12 National Security Complex U-Mo Fabrication	H. Longmire (Y-12)
			4. LANL Progress on U-Mo Fuel Fabrication Process Development	D. Dombrowski (LANL)
			5. Protective Coatings for Long Term Wet Storage of Spent Aluminium-Clad Research Reactor Fuel	L. Ramanathan (IPEN)
			6. Performance and Fabrication Status of TREAT LEU Conversion Conceptual Design Concepts	I.J. van Rooyen (INL)
12:15 pm Lunch Break				
12	**Conversion Analysis and Methods** Chaired by Y. Calzavara	1:15 pm	1. IVG.1M Reactor Kinetics	R.A. Irkimbekov (IAE-NNC)
			2. Comparative Validation of Monte Carlo Codes for Conversion of IRT MEPhI Research Reactor to LEU Fuel	N.A. Hanan (ANL)
			3. Loss-of-Offsite-Power Simulations for the Conversion of RHF to Low Enriched Uranium Fuel	J.R. Licht (ANL)
			4. Evaluation of Numerical Methods for Transient Thermal-Hydraulic Reactor Analysis	P.L. Garner (ANL)
			5. Validation of the Fuel Plate Swelling Models used in the Multi-Physics Simulation of the E-FUTURE-1 Fuel Irradiation Experiment	A. Yacout (ANL)
3:00 pm Coffee Break and Refreshments, M1 Poster Area				
13	**Poster Session III: Conversion Analysis and Safety Licensing** Chaired by J. Holland	3:00 – 4:00 pm	1. Nigeria Research Reactor (NIRR-1) Conversion Programme Implementation - A Regulatory Approach	G. Omeje (NNRA)
			2. Practical Application of the Graded Approach on the Safety of NIRR-1 HEU to LEU Core Conversion	K.J. Adedoyin (NNRA)
			3. Vibration Diagnostics of Cooling Circuit in Maria Reactor Before and After Fuel Conversion from HEU to LEU	T. Krok (NCNR)
			4. Reversal of OFI and CHF in Research Reactors Operating at 1 to 50 Bar	B. Dionne (ANL)
14	4:00 pm **Summary and Closure** John Stevens (ANL) and Pablo Adelfang (IAEA)			
5:00 pm Adjourn				

Session Abstracts

Opening and Welcome
Plenary Session

Emcee: Jordi Roglans-Ribas
Director, Nuclear Engineering Division
Argonne National Laboratory

Welcome to IAEA and RERTR-2014 International Meeting

J. C. Lentijo
International Atomic Energy Agency
Vienna International Center, PO Box 100, 1400 Vienna, Austria

Session 1
Global Progress in HEU Minimization

Chair: Jessica Halse

1.1 NNSA's Role in HEU Minimization

Arthur Atkins
Assistant Deputy Administrator
U. S. Department of Energy/National Nuclear Security Administration
1000 Independence Ave, SW Washington, DC 20585 – USA

1.2 Conversion of ARGUS to Operation with LEU and IRT-3M LEU Fuel Qualification Progress

Nikolay V. Arkhangelsky
State Atomic Energy Corporation "Rosatom"
24 Bolshaya Ordynka st., Moscow 119017 – Russia

1.3 Efforts Made for the Conversion of Ghana's MNSR to LEU Fuel

H. C. Odoi, J. K. Gbadago, R. G. Abrefah, S. A. Birikorang, R. B. M. Sogbadji, E. Ampomah-Amoako
National Nuclear Research Institute
Ghana Atomic Energy Commission, Atomic Road, Kwabenya, Accra – Ghana

Ghana Research Reactor-1 is one of the Miniature Neutron Source Reactors in operation outside China, and it has been in operation since it was commissioned in March 1995. The fuel of the reactor is UAl_4 in an aluminum matrix and has an enrichment of 90.2 %. The reactor core has been earmarked for conversion from the 90.2 % enriched HEU to about 13.0 % enriched LEU; hence various studies have been undertaken in pursuance of this course. A Project and Supply Agreement for LEU fuel has been signed by all stakeholders. Currently,

SKODA in collaboration with CIAE is designing a cask which will be used for the shipment of the spent HEU fuel from Ghana to China.

1.4 Analysis of Jamaican SLOWPOKE-2 Research Reactor for the Conversion from HEU to LEU Fuel

H. Dennis, International Centre for Environmental and Nuclear Sciences
University of the West Indies Mona, Kingston7 – Jamaica
F. Puig, GTRI Program, Nuclear Engineering Division
Argonne National Laboratory, 9700 Cass Avenue Argonne Illinois, 60439 – USA

The Jamaican SLOWPOKE-2 (JM-1) is a 20kW research reactor manufactured by Atomic Energy of Canada Limited that has been operating for 30 years at the University of the West Indies, Mona Campus in Kingston, Jamaica. The University, with the assistance of the IAEA under the GTRI/RERTR program, is currently in the process of converting from HEU to LEU. Full-reactor neutronic and thermal hydraulic analysis of the reactor was performed, using MCNP5 and PLTEMP/ANL v4.1 respectively, on both the existing HEU and proposed LEU core configurations. With the conversion, although the reactor power will increase from a full nominal power of 20kW to approximately 22kW in order for the $10^{12}\,n\cdot cm^{-2}\,s^{-1}$ flux in the inner irradiation channels to be maintained, and the maximum fuel pin temperature will increase from ~82°C to ~108°C, the analyses illustrated that there will be increased safety margins. There will be no significant changes in reactor behavior and the inherent safety features, characteristic of the SLOWPOKE-2 reactor, will be preserved.

1.5 IAEA Activities in Support of HEU Minimization: 2014 Update

F. M. Marshall, P. Adelfang, J. M. Dix, E. E. Bradley, S. M. Tozser
Department of Nuclear Energy, Division of Nuclear Fuel Cycle and Waste Technology, Research Reactor Section, International Atomic Energy Agency
Vienna International Center, PO Box 100, 1400 Vienna – Austria

The IAEA has been involved for more than thirty years in supporting international nuclear non-proliferation efforts associated with reducing the use of highly enriched uranium (HEU) in research reactors and other civilian activities. IAEA projects have directly supported HEU minimization efforts by providing assistance related to fuel conversion from HEU to low enriched uranium (LEU), fuel shipments, and reduction of HEU use in other applications. Current activities on the subject, such as the development of a book on U-Mo fuel, are in response to Member State requests and future efforts will be designed to match requirements of international programmes. This paper presents the current status, as updated from 2012, of the IAEA efforts to support of the global objective to minimize the use of HEU in civilian applications.

1.6 HERACLES - U-Mo Fuel Qualification in Europe

Yoann Calzavara
Head of Service, Institut Laue-Langevin
71 avenue des Martyrs, Grenoble, 38000 – France

1.7 U-Mo Monolithic Fuel Development for Conversion of High Performance Reactors

Chris Landers
Convert Program Manager
U. S. Department of Energy/National Nuclear Security Administration
1000 Independence Ave, SW Washington, DC 20585 –USA

SESSION 2
Fuel Utilization and Disposition

Chair: Natraj Iyer

2.1 Overview of Environmental Management Nonproliferation and Highly Enriched Uranium Minimization Mission Activities

Edgardo DeLeon
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1000 Independence Ave., Washington DC 20585 – USA

Maxcine Maxted
U.S. Department of Energy
Savannah River Site, P.O. Box A, Aiken, SC 29802 – USA

This presentation will provide an overview of Environmental Management (EM) Nonproliferation and Highly Enriched Uranium (HEU) minimization mission activities. The presentation will discuss EM's role and partnership with the National Nuclear Security Administration's Global Threat Reduction Initiative. The presentation will also focus on the nuclear materials disposition process, primarily at the Savannah River Site, near Aiken South Carolina. Lastly, the presentation will also discuss ongoing and potential future mission activities involving international partners to support non-proliferation and HEU minimization activities.

2.2 Full Core Conversion and Operational Experience with LEU Fuel of the DALAT Nuclear Research Reactor

Nhi Dien Nguyen, Ba Vien Luong, Vinh Vinh Le, Ton Nghiem Huynh, Kien Cuong Nguyen
Reactor Center, Dalat Nuclear Research Institute, Vietnam Atomic Energy Institute
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After successful in full core conversion from HEU to LEU fuel at the end of 2011, normal operation and utilization of the Dalat Nuclear Research Reactor was emphasized and improved. During carrying out commissioning program for reactor start up, reactor characteristics parameters were measured and compared with design calculated data. Good agreement between experimental data and calculated data was archived and the working core has met all safety and exploiting requirements. Fuel and in-core management of LEU core were implemented by using MCNP-REBUS linkage computers code system with visual

interface to make friendly using. The neutron trap was modified to serve for producing I-131 by increasing 6 containers. Other irradiation channels inside the reactor core and horizontal beam tubes are being effectively used for neutron activation analysis, fundamental research, nuclear data measurement, neutron radiography and nuclear structure study.

2.3 Crossing the Finish Line: Ending Civil Use of HEU

Miles A. Pomper
James Martin Center for Nonproliferation Studies, Monterey Institute for International Studies
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One of the seminal achievements of the Nuclear Security Summits (NSS) initiated by President Barack Obama has been to win global support for longstanding U.S. efforts to phase out highly enriched uranium (HEU) in civilian use. However, as the final NSS approaches in 2016, the world still lacks a comprehensive multilateral strategy to minimize and ultimately eliminate HEU from the civilian sector. At the same time, U.S. budget pressures and the difficulty of some of the technical tasks that lie ahead risk slowing down future progress. In order to maintain momentum toward the goal of eliminating civilian HEU, the 2016 NSS will need to take a number of steps, such as endorsing a political framework that includes an explicit commitment in the NSS Communiqué to *end* civilian HEU use, when technically and economically feasible, not merely to *minimize* it.

2.4 IAEA Cooperation with the RRRFR Programme: 2014 Update

S. M. Tozser, E. E. Bradley, P. Adelfang, J. M. Dix
Department of Nuclear Energy, Division of Nuclear Fuel Cycle and Waste Technology
Research Reactor Section, International Atomic Energy Agency
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Since the 1990s, the IAEA has been providing continuous support to international nuclear non-proliferation activities associated with the reduction of utilization of highly enriched uranium (HEU) by eliminating stockpiles of HEU and encouraging eligible countries to convert their research reactors from highly enriched uranium (HEU) to low enriched uranium (LEU). In support of these efforts, the Russian Research Reactor Fuel Return (RRRFR) programme was launched in 2011. The IAEA has been a very active supporter of the RRRFR programme since its inception. Under the auspices of the RRRFR programme, the Agency has ensured a broad range of technical advice and organizational support for HEU fuel repatriation, and has provided training in research reactor conversion from HEU to LEU since core conversion is mandatory for reactors to participate in the RRRFR programme. This paper gives an overview of the achievements made by the RRRFR programme with special emphasis on the IAEA's contribution. This will include the shipments' history in terms of fresh and spent fuel, as well as a summary of experiences gained throughout the process. The paper discusses the consolidated knowledge of the unique international programme and shares the most important lessons learned.

2.5 Potential German Highly Enriched Uranium (HEU) Pebble Bed Fuel Disposition at SRS

Maxcine Maxted[1], Edgardo DeLeon[2]
[1]Spent Fuel Program Manager, [2]Environmental Management
US Department of Energy, Savannah River Site, PO Box A, Aiken, SC 29801 – USA

The Department of Energy- Environmental Management (DOE-EM) is currently evaluating the potential for receipt of the German Highly Enriched Uranium (HEU) pebble bed reactor fuel at the Savannah River Site. New technology has been developed by SRNL which allows for a chemical digestion of the graphite cladded fuel. If successful technology maturation is achieved, this technology could provide a disposition path for graphite fuel. This presentation will provide an overview of the history and current status of the project including the development of an Environmental Assessment (EA) under the National Environmental Policy Act (NEPA) requirements. This presentation will provide an example of how a spent nuclear fuel (SNF) with limited disposition options can be evaluated and new potentials developed for the SNF disposition.

SESSION 3
Poster Session I: HEU Minimization

Chair: John Holland

3.1 Successful Operation of WWR-SM Research Reactor after Conversion to LEU Fuel

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Institute of Nuclear Physics of Academy of Science of Republic of Uzbekistan (INP AS RU)
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WWR-SM research reactor, operating at 10 MW, was converted to LEU fuel in 2009. Active core of WWR-SM research reactor of the Institute of Nuclear Physics of Academy of Science of Republic of Uzbekistan (INP AS RU) contains now 24 IRT-4M type fuel assemblies made in Novosibirsk chemical plant, Russian Federation.

WWR-SM research reactor is pool type reactor intended for production of neutron fluxes in range of 10^{12}-$3x10^{14}$ neutron/cm^2 with energy of 0.1-9 MeV and operates on 10 MW. Reactor has 40 vertical and 10 horizontal channels for irradiation of samples. For realization of State scientific–technical programs, branch-wise tasks in the field of fundamental and applied scientific studies, solving scientific-technical tasks, 29 vertical and 9 horizontal channels are used and bound to scientific departments of the INP AS RU. Other channels are reserved and used for control over technical conditions of reactor and carry out orders of side organizations.

Important studies in fundamental and applied fields of nuclear physics, radiational physics of hard body, radiational material science, radiochemistry, activation analyses, information technologies and scientific devices construction are held using WWR-SM research reactor. These studies let solve actual for the Republic problems of geology, mining, metallurgic and jewelry industry, medicine, ecology, agriculture, criminalistics, and also develops

technologies of producing new radioisotope production, modify constructional materials, mineral raw materials and other different products.

Main tasks, solved at reactor are providing and servicing scientific departments with neutron fluxes and holding physical and nuclear-analytical studies, and also introduction of scientific and technological developments of the institute to get different radioisotopes (I-125, I-131, P-32, P-33, S-35, Tc-99m generator, Sm-153, Ir-192, Au-198), modification of materials and crystals.

Full technological cycle of reactor operation using 19.7% enriched by U-235 IRT-4M type fuel developed and realized at WWR-SM research reactor. Estimates of radiation levels in technological rooms and surrounding territory while using IR-4M type fuel have been made. Potential of technical possibilities of WWR-SM research reactor of INP AS RU is kept to perform scientific researches and carrying out industrial orders due to getting 1.5×10^{14} $ncm^{-2}s^{-1}$ in the active core of reactor while using IR-4M LEU fuel.

3.2 Development of Low Enriched Uranium Targets for Mo-99 Production

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JSC “SSC Research Institute of Atomic Reactors”, 433510 Dimitrovgrad-10 – Russia

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National Nuclear University MEPhI (Moscow Engineering Physics Institute)
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JSC “SSC RIAR” conducts the experiments to promote conversion of existing highly enriched uranium-based technologies. These technologies provide for using highly enriched uranium as nuclear fuel in research reactors as well as for Mo-99 production. The present paper presents the results of experiments related to the development of low enriched uranium-based targets with the focus on new design modifications (flat and tubular types). Presented here are the results of neutronic and thermal-hydraulic calculations. The present paper gives an overview of possible target design optimization ways aimed at increasing the Mo-99 production yield. Given here are also the results of comparative analysis of thermal and physical characteristics of targets. This work was done in cooperation with the National Research Nuclear University Moscow Engineering and Physics Institute with funding from the Ministry of Science and Education of the Russian Federation within the framework grant # 02.G25.31.0069.

3.3 Feasibility of Conversion to LEU-based Reactor Production of Mo-99

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At present JSC “SSC RIAR” employs fission Mo-99 accumulation technology with the use of

highly enriched uranium-based targets. In order to implement the conversion of existing nuclear technologies to low enriched uranium (LEU), a calculation data analysis was performed to study feasibility of standard technology conversion. The current stick-type target was used as a basis. A whole scope of scientific research and development study including calculations of neutronic and thermal-hydraulic characteristics was performed. Feasibility of conversion to LEU-based reactor production of Mo-99 was demonstrated. This work was done in cooperation with the National Research Nuclear University Moscow Engineering and Physics Institute with funding from the Ministry of Science and Education of the Russian Federation within the framework grant #02.G25.31.0069.

3.4 LEU Transition Core Optimization for the WWR-M Research Reactor in Ukraine

Y.P. Mahlers and V.M. Makarovsky
WWR-M Research Reactor
Institute for Nuclear Research, Prospect Nauki 47, Kiev 03680 – Ukraine

Because of full-core conversion of the WWR-M reactor in Ukraine with simultaneous replacement of all remaining HEU fuel by fresh LEU fuel, the number of fuel assemblies in the core and total power of the reactor were lowered essentially. As a result, fast and intermediate neutron fluxes in beam tubes were decreased. With fuel burnup, the number of fuel assemblies in the core and reactor power will be increased, so for equilibrium LEU core, flux in beam tubes will be almost the same as for HEU core. However, to solve this problem during transient period, core reload optimization should be applied. Thus, dependence of the number of fuel assemblies in the core and maximum allowed power of the reactor on LEU fuel burnup is estimated using calculations by MCNP-4C, WIMS-ANL and PLTEMP codes. Then, core configuration is optimized successively with increasing fuel burnup to provide sufficient fast and intermediate neutron flux in beam tubes and satisfy all the safety requirements.

3.5 Brief History of MARIA Conversion from HEU to LEU

M. Migdal and T. Krok
National Centre for Nuclear Research
05-400 Otwock-Świerk – Poland

MARIA reactor core initially operated with 80% enriched HEU fuel and converted to 36% HEU in 1999. Since 2005 in National Centre for Nuclear Research works on the fuel conversion from highly enriched uranium with 36% of ^{235}U to low enriched one (LEU) with ^{235}U content below 20% have been led. This is the most recent fuel conversion held in NCNR. This task sets before us a number of serious challenges. Main one was upgrading of reactor pump system, among others there are PIE tests of LTA's, measuring safety related coefficients, constant fuel channels monitoring using Fuel Integrity Monitoring System (FEIMS), vibration tests of new fuel and last but not least development of neutronic and thermal-hydraulic calculations inspired by the full core conversion. All this work is performed under supervision of some US experts within the framework of an US-government-funded international project to limit terrorist attack risk.

3.6 Construction of a New LEU Radioisotope Production Fission Plant in Argentina

Daniel Cestau, P. Cristini, A. Novello, E. Carranza, M. Milidoni,
F. Fraguas, C. Bravo, H. Spinelli, F. Nielli

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Molybdenum-99 and Iodine-131 from low enriched uranium-aluminium (LEU) targets is being produced in Argentina, at Ezeiza Atomic Centre, since 2002. Local isotope demand and part of regional markets are supplied by a weekly production. Before 2002, Argentina produced for more than 15 years fission molybdenum-99 from HEU targets. In 2010, Argentina initiates a project for constructing a new 30 MW LEU reactor, the RA-10. It is planned to be finished by 2018. In 2012, a project for constructing a new and bigger LEU Fission Radioisotope Production Plant (PPRF) was approved by the government. The project started in 2014. The new PPRF will increase production capacity several times compare with the present plant. A description of the new PPRF, its designed capacity, location, a brief description of the LEU method and the project timeline will be presented.

3.7 Recovery of Uranium-Thorium from HTGR Fuel Using Salt-based Graphite Digestion

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The Savannah River National Laboratory (SRNL) and Forschungszentrum Jülich GmbH are partnering in the development of a digestion technology for the processing of graphite-based high temperature gas reactor (HTGR) nuclear fuels. SRNL has successfully demonstrated the conversion of the graphite in HTGR fuel to digestion byproducts using a mixture of molten salts. The reactions with graphite occur effectively between 500 and 800 °C with the graphite reaction rate increasing linearly as a function of temperature. The same chemical system also reacts with the silicon carbide in TRISO fuels. The system can be configured to react with graphite only for BISO kernels or graphite and silicon carbide for TRISO fuel kernels. The process chemistry was successfully demonstrated on the processing of full HTGR BISO pebbles containing unirradiated U/Th oxide fuel kernels. Fission product partitioning studies with irradiated fuel kernels have also been completed.

SESSION 4
Perspectives on Fuel Development and Performance Issues
Chair: Sven Van den Berghe

4.1 U.S. Progress in U-Mo Monolithic Fuel Development

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Based on results from scoping irradiation tests, U-Mo monolithic fuel was selected for further development as the primary LEU conversion fuel for U.S. high performance research reactors. More focused testing in the RERTR-12, AFIP-6, and AFIP-7 irradiation test campaigns has confirmed that the U-Mo monolithic fuel system exhibits good irradiation behavior over the range of irradiation conditions required to support operation of high performance research reactors licensed by the Nuclear Regulatory Commission in the United States. Fuel plates from the RERTR-12 experiment, in particular, show stable behavior to very high burnup (>$1x10^{22}$ local fission density) at high power density (18-40 KW/cm^3 average), after which they failed by pillowing with no fission product release to the coolant. Based on these results, the U.S. Global Threat Reduction Initiative is proceeding with fuel qualification, beginning with the development of commercial-scale fabrication processes and selection of a fabrication process through the MinPlate-1 (MP-1) irradiation test. Fuel fabricated using the selected process will be qualified through a series of miniplate, full-size plate, and fuel element irradiations. Additional efforts are ongoing to define fuel performance limits through measurement and assessment of material properties, fuel performance modeling, and flow testing.

4.2 Examination of High Uranium Density Research Reactor Fuel Performance and Endurance

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This study evaluates the postirradiation data of the most recent monolithic mini plate test RERTR-12. The test consisted of mini plates of the currently accepted design, employing a Zr diffusion barrier between the U-Mo fuel foil and Al cladding. A range of U-235 enrichment was used and the test was performed in a flux trap in the Advanced Test Reactor ATR. During routine visual examinations four mini plates were found to have developed blisters or pillows all in the high enrichment fuel plates at fission densities well beyond attainable with LEU fuel. Examination of metallographic sections of the failed plates supports that the pillowing occurred during the final shut down of the irradiation. However, the determining factor is the accumulated radiation damage in the fuel resulting in high swelling, development of substantial fission gas porosity and embrittlement rendering the fuel

plates susceptible to modest thermal stress occurring at shutdown. All failures occurred in the highly enriched fuel plates and originated at peak fission density locations well in access of fission densities achievable with LEU fuel. The probability of failure in this mini plate test is quantified using the Weibull failure analysis method.

4.3 U.S. High Performance Research Reactor LEU Conversion Design Parameters

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Based on favorable irradiation behavior, U-10Mo monolithic fuel has been selected for qualification for the LEU conversion of the U.S. high performance research reactors (US HPRR). Good irradiation behavior has previously been demonstrated in test plate geometry across a range of irradiation conditions similar to those found in the current US HPRR. In order to allow a fabrication process selection, a mini-plate test, MP-1, is planned to test fuel performance across the range of reactor-specific plate geometries and plate histories. Design Parameters of the US HPRR LEU conversion cores have been documented for limiting plates. In a working group effort, reactor core designers and safety analysts, the fuel developers, and fabrication experts have defined and represented the range of limiting plate geometries and irradiation histories in the US HPRR for subsequent mini-plate and full-size plate irradiation testing.

4.4 Progress on the Development of U-Mo Fuel for Qualification in Korea

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KAERI has started the installation of the facility equipment and established the fabrication technology for supplying the plate-type fuel to the KIJANG research reactor (KJRR) which is under construction in Korea since 2011. In KJRR, 8gU/cc U-Mo dispersion plate-type fuel was chosen to be used for the first time in the world. Therefore, qualification test for plate-type U-Mo dispersion fuel must be conducted before getting license from our regulatory body for providing plate-type U-Mo dispersion fuel to KJRR. KAERI has plans to irradiate the U-Mo fuel mini-plate at HANARO for three times(HAMP-1, 2, 3) from December, 2013 as well as to irradiate the U-Mo LTA at ATR from June, 2015. In this paper, overall fabrication process flow and irradiation specification for the 1st, 2nd and 3rd irradiation test at HANARO(HAMP-1, 2 and 3) as well as U-Mo LTA irradiation test at ATR will be explained in detail.

4.5 Closing Nuclear Security Gaps: International Cooperation in LEU Fuel Development and HEU Minimization Norms

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Securing and eliminating the commerce of vulnerable nuclear material remains one of the most significant challenges to global nuclear security. President Obama, in his 2009 Prague speech, highlighted this challenge and committed the United States to increase efforts to ensure that vulnerable material never reaches the hands of a terrorist. Similarly, at the 2010 and 2012 Nuclear Security Summits, leaders from across the globe embraced this mission and are working to protect nuclear material, convert research reactors from the use of highly enriched uranium (HEU) fuel to low enriched uranium (LEU) fuel, and permanently remove fresh and spent HEU from civilian sites. Despite these high-level commitments, and vastly increased efforts by the United States, HEU minimization has not been universally accepted. In particular, conversion of high performance research reactors has faced resistance in the nuclear energy community. Research reactor fuels make up the majority of HEU exports by the United States and their commerce continues to pose a security threat, because of the fuels' vulnerability to interception and use in an illicit nuclear device. At the same time, the reactors that receive exports function to support vital scientific missions. To maintain operations and meet mission requirements, high performance research reactor conversion is only possible through the development of high density LEU fuels. This paper contends that, despite technical and political challenges, international LEU fuel development is key to maintaining HEU minimization momentum by bridging the gap between nonproliferation advocates and research reactor users. Additionally, high density LEU fuel development may positively impact the development of the next generation of LEU-fueled research reactors, mitigating the impact of HEU minimization on nuclear science.

SESSION 5
MNSR and Japan Conversions to LEU Operation

Chair: Joanie Dix

5.1 GTRI Role in MNSR and Japan Conversions

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This paper describes the role of the U. S. Department of Energy/National Nuclear Security Administration's Global Threat Reduction Initiative (GTRI) in support of efforts to convert Chinese-origin miniature neutron source reactors (MNSRs) and several Japanese research reactors from the use of highly enriched uranium (HEU) fuel to low enriched uranium (LEU) fuel. While these various programs fall within the broad concept of HEU minimization, the role of GTRI varies with each project. From direct bilateral cooperation to technical support to Institutes to indirect technical or financial support provided via the International Atomic Energy Agency (IAEA), GTRI seeks to work with each country and organization in a manner

that is both appropriate and results driven. GTRI is working directly with the government of China and the Institute of Atomic Energy (IAE) to convert the Prototype MNSR within China. GTRI is also working with China, through the IAEA, to support conversion activities for MNSRs outside of China (Ghana, Nigeria, Pakistan, Iran and Syria). One avenue for this support is the IAEA's MNSR Working Group, which meets once per year to discuss progress, challenges and next steps. Finally, GTRI is working directly with the Kyoto and KINKI Universities to provide technical support in their efforts to convert their various reactors and critical assemblies to LEU fuel.

5.2 The Physics Experimental Study for Prototype MNSR with LEU Core

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$MNSR_S$ (Miniature Neutron Source Reactor) are low power research reactors designed and manufactured by China Institute of Atomic Energy (CIAE). $MNSR_S$ are mainly used for NAA, training and teaching, testing of nuclear instrumentation. The first MNSR, the prototype MNSR, was put into operation in 1984, later eight other MNSRS had been built both at home and abroad.

For Prototype MNSR, highly enriched uranium (90.2%) is used as the fuel material, Al alloy as cladding, metal Be as reflectors and light water as moderator and coolant.

Without changing the core dimensions of the Prototype MNSR but substituting the HEU fuel with LEU fuel and Al cladding with Zircaloy cladding, the critical mass, the control rod worth, top Be reflector worth and neutron flux distribution are measured and the final loading of fuel elements are determined. The experiment was done on the Zero Power Experiment equipment of MNSR.

5.3 Status Report of Activities for the Core Conversion of Nigeria MNSR to LEU

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The Nigeria Miniature Neutron Source Reactors (MNSR) code named NIRR-1 is a low power, tank-in-pool research reactor currently fueled with about 1 kg of HEU. NIRR-1 is currently in its first fuel cycle and was designed mainly for neutron activation analysis and production of some short-lived radioisotopes. Over the years, studies under the aegis of the IAEA CRP and RERTR programme have been performed to convert MNSRs in general and NIRR-1 in particular to LEU.

This report contains a summary of the results of design and safety analyses performed under the aegis of the IAEA CRP entitled "Conversion of MNSR to LEU" 2006 – 2012 for the conversion of NIRR-1. The objective of the CRP was to identify an LEU core with similar operational capabilities as the original HEU core and with acceptable safety margins under both normal and accident conditions. In order to provide comparisons between the proposed LEU core and the initial HEU core, thorough analyses were performed for both cores. Results obtained indicate that the conversion of the NIRR-1 to LEU fuel does not present any new potential accidents nor does the conversion increase the consequences of any of the postulated design basis accidents identified in the current approved SAR. At present, activities leading to the eventual conversion have slowed down. The challenges responsible for the lull in activities currently being encountered and the way forward are presented.

5.4 Utilization of Low-Enriched High Density Fuel at Dry Cores of Kyoto University Critical Assembly - Current Progress of the Feasibility Study

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This paper describes the updated results of the feasibility study for utilizing the low enriched uranium (LEU) at the solid moderated / reflected core (A- and B-core; "dry core") of the Kyoto University Critical Assembly (KUCA). Due to the variety of neutron spectrum and core composition of KUCA cores, the substitution of the existing HEU fuel with LEU fuel will be of notable interest from reactor physics study and thus this feasibility study is being pursued within the framework of joint scientific study between Kyoto University Research Reactor Institute (KURRI) and Argonne National Laboratory (ANL). The updated results from Kyoto University on neutronic characteristics analysis of the dry core, using both deterministic and statistical analysis using U-Mo and U-silicide as the fuel material, is described in this paper. The results are summarized from the viewpoint of critical mass (number of fuel plates required) and neutron spectrum index, as well as sensitivity of fuel fabrication tolerance to neutronic characteristics.

5.5 Conversion of the KUCA "Type-A" Cores to LEU Fuel Preserving Reactivity and Central Flux Spectra

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Recent studies demonstrated the feasibility of converting the 93% High Enriched (HEU) KUCA cores to the use of 19.75% Low Enriched (LEU) U-10Mo fuel by preserving the same reactivity. However, with the conversion of the KUCA cores it is preferable to preserve not only the reactivity but other important neutronic features as well. The present paper discusses the feasibility of converting the KUCA cores while preserving the neutron flux spectrum at the core center as well as the reactivity. Initial results indicate that a 12-mil thick U-10Mo foil is a good candidate for the conversion. Preliminary studies were also done on the sensitivity of the KUCA cores to the dimensional tolerances that are likely to be associated with the fabrication of the fuel plates. The results indicate that extremely high fabrication accuracy is required or the reactivity effects could be non-negligible.

SESSION 6
High Performance Reactor Conversions
Chair: Erik Wilson

6.1 Safety Analysis of U-Mo LEU Fuel with Unfinned Cladding for the MIT Research Reactor

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The Massachusetts Institute of Technology Reactor (MITR) is a 6 MW research reactor operating with highly-enriched uranium (HEU) finned plate-type fuel. The conversion objective is to design a low-enriched uranium (LEU) fuel element that could safely replace the current 15-plate MITR HEU fuel element and maintain performance while requiring minimal, if any, changes to the reactor structures and systems. Recent design analyses of alternatives to 0.25 mm clad finned LEU fuel plates have shown a 19-plate unfinned LEU fuel element with increased cladding thickness and thinner fuel meat thickness on the outer plates to be a feasible alternative. This study presents the results of neutronic and thermal hydraulic steady-state analyses of power distribution, and the heating impact of conversion on in-core experiments with use of this alternative fuel. Accident analyses are also given, including thermal margin analysis of loss-of-flow and reactivity insertion accidents.

6.2 Continuing LEU Conversion Activities at the High Flux Isotope Reactor

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ORNL has collaborated with the GTRI program since 2005 to convert HFIR to LEU. During 2014, alternatives for HFIR complex fuel were evaluated for reactor performance and safety to select a new design that can be manufactured reliably and economically while maintaining HFIR's world-class scientific performance. Neutronics modeling added the capability to explicitly represent the geometry of the fuel plates. Gaps in data required to benchmark steady-state and transient neutronics analyses of HFIR LEU fuel were identified. Thermal-hydraulic modeling employed a legacy code to evaluate alternative fuel designs while continuing to develop multi-physics models to demonstrate that adequate safety margins can be preserved. ORNL coordinated with GTRI in fuel testing, fabrication, and qualification to ensure that a converted HFIR is safe, reliable, and meets regulatory requirements. ORNL began the "safety-in-design" process to document assumptions and plans for HFIR LEU conversion. Plans for 2015 and beyond will be discussed and key issues will be highlighted.

6.3 Accident Analyses for the Conversion of the University of Missouri Research Reactor from Highly-Enriched to Low-Enriched Uranium

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The University of Missouri Research Reactor (MURR®) is one of five U.S. high performance research and test reactors that are actively collaborating with the Global Threat Reduction Initiative (GTRI) Reduced Enrichment for Research and Test Reactors (RERTR) Program to find a suitable low-enriched uranium (LEU) fuel replacement for the currently required highly-enriched uranium (HEU) fuel. Analyses of accident scenarios for a proposed core loaded with U-10Mo monolithic LEU fuel have been completed. The models include both fresh and irradiated fuel assemblies. Furthermore, a series of branch cases to evaluate the impact of the uncertainties in core operating conditions or fuel thermophysical properties that may affect the severity of the accidents are considered. Results for a positive reactivity insertion accident, a loss of coolant accident, a loss of flow accident, and the maximum hypothetical accident (MHA) are presented in this paper. All accident scenarios demonstrate an acceptable margin to potential fuel damage, or acceptable dose consequences in the case of the MHA.

6.4 Planning for the Conversion of the NIST Center for Neutron Research to LEU from HEU

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The NIST Center for Neutron Research (NCNR) with the assistance of the Brookhaven National Laboratory and the Global Threat Reduction Initiative (GTRI) program continues to perform safety analysis and review in preparation for the conversion of the NIST reactor (NBSR) from HEU to LEU fuel. The conversion is tentatively scheduled to occur in 2027. The NBSR's user facility mission is primarily related to the production of cold-neutrons for materials and neutron physics research. The conversion of the NBSR will minimize changes to the reactor fuel element design and cycle length while maintaining the nominal operating

power of 20 MW. Analyses of steady state equilibrium conditions and HEU to LEU transition core characteristics have been completed. Accident analysis of the converted core is ongoing with comparison to previous HEU core characteristics. A draft Safety Analysis Report is planned for completion by the end of CY14.

6.5 Enhanced Low-Enriched Uranium Fuel Element for the Advanced Test Reactor

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Under the current US Department of Energy (DOE) policy and planning scenario, the Advanced Test Reactor (ATR) and its associated critical facility (ATRC) will be reconfigured to operate on low-enriched uranium (LEU) fuel. This effort has produced a conceptual design for an Enhanced LEU Fuel (ELF) element. This fuel features monolithic U-10Mo fuel foils and aluminum cladding separated by a thin zirconium barrier. As with previous iterations of the ELF design, radial power peaking is managed using different U-10Mo foil thicknesses in different plates of the element. The lead fuel element design, ELF Mk1A, features only three fuel meat thicknesses, a reduction from the previous iterations meant to simplify manufacturing. Evaluation of the ELF Mk1A fuel design against reactor performance requirements is ongoing, as are investigations of the impact of manufacturing uncertainty on safety margins. The element design has been evaluated in what are expected to be the most demanding design basis accident scenarios and has met all initial thermal-hydraulic criteria.

6.6 Plasma Sprayed Zirconium for US HPRR LEU Conversion Fuel Diffusion Barrier

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Plasma spraying of zirconium (Zr) is being investigated for the diffusion barrier between the U-Mo fissile material and the aluminum cladding for US HPRR LEU fuel. Interest in plasma spraying is in part due to the application of the Zr to the U-Mo late in the manufacturing process allowing for the more efficient recycle of scrap fuel material which allows higher LEU utilization. Recent results have explored the scale up from mini-plate (25 mm x 100 mm) sized samples to full sized HPRR fuel samples (127 mm x 610 mm). In addition, coatings on the edges of the fuel are being investigated to prevent all contact between the U-Mo and the Al. Activities involving plasma sprayed Zr for the upcoming MP-1 reactor test in ATR are also discussed. A subset of the MP-1 irradiation test samples will utilize plasma sprayed Zr for the diffusion barrier.

SESSION 7
Fuel Development – Irradiation Testing, PIE Analysis and Modeling
Chair: Harald Breitkreutz

7.1 Design of the MP-1 Experiment for Irradiation in the Advanced Test Reactor

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Under the umbrella of the Global Threat Reduction Initiative (GTRI) the United States is working on the development of the new monolithic Low Enriched Uranium (LEU) nuclear fuel for conversion of the US High Performance Research Reactors (USHPRR). Successful implementation of the new type of fuel depends not only on demonstration of its safe and reliable irradiation performance, but also on the efficient and economical, i.e. commercially viable fabrication process. Miniplate-1 (MP-1) is the first in the series of experiments focused on irradiation testing of U-Mo monolithic fuel specimens manufactured using several fabrication processes that meet commercial viability requirements. The testing of the fuel will be accomplished in ATR under the conditions prototypic of the research reactor environments. At the conclusion of the test, performance of the fuels manufactured using different commercially viable processes will be compared, resulting in down-selection of the fuel fabrication process for subsequent large scale irradiation testing, qualification and licensing. This paper is focused on the details of the design of the MP-1 irradiation experiment in ATR.

7.2 IVG.1M - LEU Fuel Test Plan

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Fuel Assembly with low-enriched uranium (LEU FA) is designed for the conversion of IVG.1M reactor operated by the National Nuclear Center of the Republic of Kazakhstan. Pilot low-enriched Water-Cooled Technological Channels (LEU WCTC) are produced by Russian manufacturer Federal State Unitary Enterprise Scientific-Research Institute LUCH and delivered to Kazakhstan to conduct IVG.1M in-reactor tests.

Two pilot LEU WCTC as part of existing reactor core will be used for testing; one having 800-mm long fuel elements (FE) and the other having 600-mm long FE. Irradiation testing of two LEU-WCTC is to provide the burnout identical to the values of HEU-WCTC No. 4 unloaded from the IVG.1M for testing in 2004. As of the time of testing, WCTC No. 4 had an energy-producing 35 MWh under the fluence 10^{18} n/cm^2 and burnup 1.82 g U^{235}. Testing results shall prove the feasible application of identical LEU-WCTC as part of LEU reactor core or necessity for further design debugging.

7.3 SEM Characterization of U-7Mo Irradiated to High Fission Density at Relatively High Power, High Temperature, and High Fission Rate

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This paper investigates the extent that changes in irradiation conditions may impact the microstructural evolution of U-7Mo up to high fission density. This includes the effects of changing temperature, fission rate, and power. For a fuel irradiated in a reactor like BR-2, the power, temperature, and fission rate will all be relatively high, and it is of interest to determine if U-7Mo powders respond uniquely to these conditions, compared to the conditions of other reactors where the temperature, fission rate, and power may not be as aggressive. In order to gain insight about the microstructural response of U-7Mo irradiated at relatively aggressive conditions, U-7Mo samples were generated from fuel plate R9R010, which was irradiated as part of the RERTR-8 experiment at high temperature, high fission rate, and high power, up to high fission density. The irradiation conditions were similar to those used for the SELENIUM experiment that was irradiated in BR-2. From a mounted sample that was metallographically prepared, a focused ion beam was employed to generate samples with little surface damage that were then characterized using scanning electron microscopy. This paper will describe the as-irradiated U-7Mo microstructure observed for low and high-flux samples taken from R9R010. The microstructure observed for the high-flux sample will be compared to those observed for RERTR-7 fuel plates that were irradiated to similar fission density, but at lower power, fission rate and temperature. Finally, the potential for interconnection of fission gas bubbles in the R9R010 sample will be discussed.

7.4 Capabilities Developed for Measurement of Thermal Conductivity and Fission Gas Release of Irradiated Nuclear Fuels at PNNL

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Instrumentation necessary for measuring thermal properties of irradiated fuels and materials has been installed and commissioned within hot cells in the Radiochemical Processing Laboratory at the Pacific Northwest National Laboratory. The instrumentation and developed capabilities consists of a laser flash apparatus to measure thermal diffusivity, a differential scanning calorimeter to measure specific heat, a gas pycnometer for density measurements, and an optical microscope to perform metallography from which composite and layer thicknesses and variability across the sample are extracted. Additionally, a thermogravimetry/ differential thermal analysis coupled with a mass spectrometer has been installed for fission product release data. With this instrumentation, small-scale samples are annealed according to specified thermal profiles while continuously monitoring fission gas releases. This work has assisted in understanding of the changes in thermo-physical properties of the multi-layered fuel design as a function of temperature and burnup and provided fission product release data associated with the fuel.

7.5 Creep and Mass Relocation of U-Mo/Al Dispersion Fuel Meat during Mini-plate Irradiation

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ABAQUS finite element simulation to analyze creep and mass relocation of U-Mo/Al dispersion fuel meat observed from PIE was performed. The model considered that uniform-sized U-Mo particles were embedded in a close-packing array in the Al matrix. Fuel meat swelling is the driving force for creep and mass relocation in the meat end regions. The fuel meat swelling is a sum of fuel particle swelling by fission products and swelling in the interaction layers (IL). The net meat volume increase was obtained by subtracting the meat swelling with the volume consumptions of fuel particle and Al matrix by IL growth. ABAQUS was applied for two plate cases: one with pure Al matrix and the other with Al-5% Si. The creep rate constants for Al, IL and U-Mo were obtained that gave the best simulation for the measured meat swelling. The corresponding stresses were also calculated.

SESSION 8
Poster Session II: Fuel Development and Fabrication

Chair: John Holland

8.1 Ultrasonic Testing of Dispersion Type Fuel Miniplates Manufactured with Hydrided U-Mo Powder

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In order to evaluate the bonding integrity in dispersion fuel plate, the transmission ultrasonic technique is applied. This technique was development for monolithic fuel type and dispersion type manufactured with U-Mo atomized powder. This paper reports the results obtained from

the development, analysis and discussions conducted at CCHEN using miniplates manufactured with hydride powder and atomized U-Mo. In summary, the results for density of $6gU/cm^3$ miniplates, manufactured with atomized powder U-Mo, a signal intensity of 32.84%, in relation with the initial amplitude of ultrasonic signal was detected. However, for miniplates manufactured with particles U-Mo obtained by hydriding, the ultrasonic testing presented lower values of signal intensity, close to 10.32%. Considering the characteristics of this powder and the fabrication process of miniplates, has been verified that the porosity and brittleness of U-Mo hydrided particles produces an excessive increase in fine particles, starting with 61% vol. particles below 45 microns, checked by quantitative metallography, this percentage increases up to 93% volume after compacting (U-Mo-54 miniplate) and 96% vol. at the end of the miniplate manufacturing process. Based on these results, this paper discuss about the possibility that the excessive increase in the fraction of fine particles of fuel, besides the residual porosity present in the meat, would be the main cause for increasing of attenuation of the ultrasonic signal intensity through the fuel meat.

8.2 Characterization of Si-coated U-Mo Fuel Particles before and after Interaction Annealing

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In order to study the Si-coating layer formation and its effect as interaction barrier, cylindrical compacts, based on Si-coated atomized U-Mo fuel particles, dispersed in aluminum matrix, were manufactured. The starting microstructure (as - coated) and after interaction annealing were characterized through both, optical and SEM with EDS analysis. After Si-coating treatment by pack cementation, conducted at 1000°C/3 hours in vacuum, the silicon coating film consists of a thin layer (about 10 μm of thickness), comprised by 35%at of Si and 65% at of U plus oxygen. Regarding the coating treatment conducted under inert atmosphere (argon), migration of silicon in U-Mo particles, up to 22%at was detected, besides 24%at of uranium and 53%at of oxygen at the surface of U-Mo particles. Nevertheless, any interaction layer was observed for coated U-Mo particles treated in Argon. After interaction annealing, conducted at 500°C by 7 hours, in vacuum, the Si-coated U-Mo particles, surrounded by aluminum, forms a non-continuous interaction layer containing Al, U, Si and Mo. For the case of U-Mo particles coated in argon, the aluminium contents was very low (4%at) at the center of a very small particle. This beneficial effect can be caused by the presence of 1,7%at of silicon in the surface and 0,7%at in the center of U-Mo particle, likely forming solid solution. This paper presents and discusses in detail these results of silicon coating over U-Mo particles, as an attempt to control the formation and composition of the deleterious fuel/matrix interaction layer observed in dispersion type fuel.

8.3 Thermal Conductivity of In-pile Irradiated AFIP-1 Dispersion U-Mo Fuel

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The evolution of the thermal conductivity of research reactor fuel during in-pile irradiation plays a significant role in performance of the fuel element. To correctly simulate the heat fluxes and temperatures in the fuel meat during reactor operation and also potential accident scenarios, it is crucial to investigate the change in thermal conductivity depending on the fission density and temperature.

The Idaho National Laboratory provided a high ($6.0 \cdot 10^{21}$f/cc) and a low ($4.8 \cdot 10^{21}$f/cc) burnup fuel segment from the AFIP-1 irradiation test for thermal property measurements. The fuel consists of atomized U-7wt.% Mo powder in an aluminum matrix containing 2.1wt.%Si and is clad in Al 6061.

The Pacific Northwest National Laboratory set up instruments in their hot cells for the measurement of thermal properties. A Laser Flash Apparatus was used to determine thermal diffusivity, a Differential Scanning Calorimeter for specific heat measurements, and a Pycnometer to obtain the density of the irradiated AFIP-1 samples. The thermal conductivity can be calculated from the results of those measurements. Comparison with non-irradiated fuel shows that the thermal conductivity strongly decreases with increasing burnup.

8.4 UAl_x Plate Production: Analysis of Intermetallic Growth in UAl_2/Al

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UAl_x materials used as fuel-plates or irradiation targets are produced by reactions between UAl_2 and Al powders, motivating a comprehensive study of the diffusion mechanisms in this system. The present study aims to evaluate experimentally some fundamental features of the growing of the interaction product by means of diffusion couple experiments. The UAl_2-Al interfaces were investigated by SEM-EDS analysis, after heating the couples in the temperature range 550-645°C for dwell periods up to 100 hours. The major interaction phase, UAl_3, forms in the diffusion zone with planar morphology whereas for longer annealing times and at higher temperatures UAl_4 forms preferentially at the vicinity of the open pores and cracks of the sample. In terms of mechanism, it can be proposed that the interdiffusion reactions occur in two steps (i) formation of UAl_3 from the reaction between UAl_2 and Al, (ii) formation of UAl_4 from the reaction between UAl_3 and Al.

8.5 Thermal Conductivity of U-Mo/Al Dispersion Fuel: Effects of Particle Shape and Size, Stereography, and Heat Generation

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This paper describes the effects of particle sphericity, interfacial thermal resistance, stereography, and heat generation on the effective thermal conductivity of U-Mo/Al dispersion fuel. ABAQUS FEM was used to calculate the effective thermal conductivity of U-Mo/Al dispersion fuel by applying the models that consider particle sphericity, interfacial thermal resistance, stereography, and heat generation in the U-Mo particles. The ABAQUS simulation results were compared with measured data available in the literature. The particle sphericity effect decreases with the fuel volume fraction. It is known that two-dimensional models, in general, predict lower values of thermal conductivity than three-dimensional models due to the difference in the surface to volume ratio. By using ABAQUS FEM, we obtained consistent results. By using the results, we developed a conversion factor between the two-dimensional models and the three-dimensional models. We also investigated the effect of heat generation from the dispersed fuel particles. The effect was considerable for the cases with large interaction layer growth.

8.6 Fabrication Procedures for Manufacturing UMo-Al Dispersion Fuel at IPEN

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This work forms part of the studies presently ongoing at IPEN investigating the feasibility of fabricating U10wt%Mo-Al dispesion fuel. Miniplates were fabricated with UMo alloy powder prepared by hydriding-milling-dehydriding of the gamma phase (HMD). Hydrided pieces were fragile enough to be hand milled to the desired particle size range. Hydrogen was removed by heating the samples under high vacuum. Based on IPEN previous experience in developing and manufacturing dispersion type fuel, the objective of this work was to promote an adjustment to the current fuel manufacturing procedures, allowing the incorporation of higher uranium concentrations to the fuel. The goal is to increase the uranium concentration up to 7 gU/cm^3 by using the UMo-Al dispersion. This paper describes the main procedures used for manufacturing UMo-Al miniplates.

8.7 Metal Coating on Atomized U-Mo Particles to Suppress Interdiffusion between U-Mo/Al

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For the suppression of the interaction between U-Mo/Al, there have been numerous researches such as addition of Si into Al matrix, multi-wire monolithic fuel. Among them adding Si (over 5 wt. %) into the matrix showed very good irradiation performance due to suppression of interdiffusion between U-Mo and Al matrix. However, too much Si addition to the Al matrix would be detrimental when it comes to the subsequent reprocessing of spent fuel. Furthermore, it was reported that the themo-mechanical properties of the matrix with its high Si content may induce buckling-like deformation of fuel plates after the irradiation. For this reason, KAERI already suggested the silicide coating method using heat-treatment. In this study, we not only improved the previous silicide coating method, but also succeeded in Si, Nb coating on U-Mo powders by PVD (sputtering) method. Annealing tests were also performed to see the potency of coating layers as a diffusion barrier.

8.8 Residual Stress Measurement for Highly Radioactive Samples

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The interface bonds are critical to in-reactor performance for a high density monolithic plate fuel system which uses low enriched uranium 10wt% molybdenum foils co-rolled with Zr and clad with 6061Al. Stresses induced during reactor shutdown have been identified as a source of concern for that integrity. Because of the post-reactor activity, those residual stresses will have to be measured in a shielded nuclear radiation containment chamber or “hot cell” with remote handling of specimens and instrumentation, which limits measurement options. This study tested options for stress measurements by using surrogate fuel plates, such as with depleted uranium, but using only hot-cell appropriate equipment. Several measurements were performed using the incremental slitting method (a.k.a. crack compliance) but, for hot cell use, using a milling cutter instead of wire EDM for making the cut and a displacement sensor instead of a strain gauge. Measurements were also performed using incremental hole drilling using an interferometry system instead of strain gauges. For both measurement techniques, special data reduction development was required in order to handle discontinuities in the stress profiles across the layers. The results were encouraging, and the slitting method is now being implemented for use in a hot cell.

8.9 Bonding Toughness Measurements in LEU Fuel Plates

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Recent research into U-10Mo/Zr/Al plate fuel assemblies has illustrated the importance of fundamentally understanding interfacial mechanical behavior both before and after exposure to irradiation environments. The parameters and phenomena that have been noted include existence of stress gradients at interfaces and their influence on bond strength, and strength and fracture behavior of the various interfaces before and after irradiation. Bend testing and pull testing have been used to gain some insight on the mechanical behavior of the composite plate, but neither method can isolate the mechanical behavior of a specific bond. Here we present the fracture behavior of Al/Zr and Zr/U-10Mo bonds as measured by two newly developed methods: 1) Miniaturized bulge test, and 2) MiniCantilever beam bending. Using both methods, a crack was successfully initiated along the interface, providing for quantitative measurement of upper bound and lower bound values for the fracture energy release rate associated with fracture in the vicinity of the specific interfaces. Detailed discussions of the schemes of preparing and conducting both testing types, and computing various quantities required for the determination of the energy release rate are presented.

8.10 Intelligent Integrated Machining: Zr Thickness Measurements Using XRF for Process Control and Quality Assurance

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Reactor operation is dependent upon precise control of fuel plate geometry and thickness of individual layers, however there is currently no method to assess Zr diffusion barrier thickness over the entire surface area of the foil. An energy-dispersive x-ray fluorescence spectrometry (XRF) technique was developed to map Zr coating thickness over the surfaces of co-rolled fuel foils. Two different instruments and calibration methods were employed. An inexpensive handheld XRF analyzer was chosen for rapid non-destructive estimation of Zr coating and/or Al cladding thickness *in situ* on the shop floor, while a laboratory-grade scanning macro-XRF was used to confirm the handheld measurements and for R&D applications. For Zr foil calibration standards, results from the handheld analyzer and the macro-XRF instrument are in excellent agreement with each other and within ±5% of known values. The method was successfully applied to map Zr on 48” LEU foils and to study effects of cold-rolling.

8.11 Heavy Ion Irradiation on U-Mo/Al Layer Systems: Dependence of IDL Thickness on Irradiation Temperature and Particle Flux

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Heavy ion irradiation with 127Iodine at 80MeV is an economic out-of-pile technique to predict in-pile irradiation effects in U-Mo/Al fuels, e.g. the formation of an interdiffusion layer (IDL) between the U-Mo and the Al [1,2]. Recently, it was shown that this technique generates amorphous IDLs, provided a well-defined irradiation temperature and particle flux is chosen which is in full agreement with in-pile observations [3]. In this work, we present the dependence of the final IDL thickness after heavy ion irradiation as a function of irradiation temperature and Iodine particle flux. Both have a strong impact on the created IDL thickness, again in full agreement with in-pile data.

8.12 Fuel Fabrication Process Optimization and Alternative Fabrication Development

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Pacific Northwest National Laboratory (PNNL) is integral to the Fuel Fabrication Capability (FFC) responsibility to develop a low-enriched uranium-molybdenum fuel (LEU-Mo) in several areas, including predictive tools for optimizing the baseline process of rolling U-10Mo fuel coupons as well as development of alternate methods for application of a Zr barrier on the U-Mo fuel foil. The predictive tool has produced simulations of hot rolling U-10Mo coupons encapsulated in low-carbon steel can following two different schedules. Model predictions of the roll separation force and roll pack thicknesses at different stages of the rolling process were compared with experimental measurements. This presentation will discuss various attributes of the rolled coupons revealed by the model (e.g., waviness and thickness non-uniformity). The finite-element model developed was also used to conduct parametric studies on rolling process parameters including can material and friction. The influence of these process parameters on the rolling defects was investigated and simulation results will be discussed. PNNL is developing a process to plate Zr from a molten salt bath onto U-10Mo fuel foils and a method to co-extrude Zr and U-10Mo producing an encapsulated fuel meat. The experimental methods and concept feasibility results from these development activities will be presented. Both methods have the potential to produce a Zr barrier layer on a complex shaped fuel foil.

8.13 Design of a Full-Size Fuel Plate Irradiation for Monolithic Fuel Qualification

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The US fuel development team is focused on developing and qualifying the uranium-molybdenum monolithic fuel. Several previous irradiations have demonstrated the favorable behavior of the monolithic fuel. The overall irradiation program strategy was recently revamped to provide an opportunity to test, examine, and select the most advantageous fabrication processes for producing monolithic fuel. Specimens fabricated by the selected processes are planned to be irradiated in a series of tests in order to populate the data set needed for fuel qualification through the Nuclear Regulatory Commission. This paper summarizes the overall irradiation testing plan with an emphasis on the rigorous approach to design, analysis, and requirements flowdown for qualification tests. The recently designed Full Size Plate 1 irradiation test is also summarized as an example of this strategic approach to qualifying monolithic fuel.

8.14 Xe Irradiation on ZrN-Coated U-Mo/Al Dispersion Fuel

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High-density U-Mo/Al dispersion fuels, developed for high-performance research reactors under the Global Threat Reduction Initiative (GTRI) program, were irradiated with 80 MeV Xe ions up to various doses (1.8-2.9×10^{17} ions/cm^2). Fuel samples containing ZrN-coated and uncoated U-Mo particles were irradiated. This experiment is aimed to investigate microstructural responses of the U-Mo particles dispersed in Al matrix as a function of irradiation damage. The responses include the fuel-matrix (U-Mo-Al) interaction (FMI) and the Xe gas bubble formation. Post irradiation examinations (PIE) reveal that ZrN coating can effectively eliminate FMI when the coating is intact. FMI forms on the surface of uncoated U-Mo particles and the locations where ZrN coating layers were compromised. Xe gas bubbles appear in both U-Mo and FMI regions, and some large bubbles (~ 200 nm) are seen to form by interlinking small bubbles. Current results and further PIEs can help understand U-Mo/Al dispersion fuel irradiation behavior in reactor.

SESSION 9
Russian-Designed Reactor Conversions
Chair: Val Brusilovsky

9.1 The Russian ARGUS Solution Reactor HEU-LEU Conversion: LEU Fuel Preparation, Loading and First Criticality

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ARGUS is a 20 kW solution reactor, which has operated at NRC “Kurchatov Institute” since 1981. Fuel was highly enriched (90%) uranyl (HEU) sulfate solution. The reactor has an inherent safety.

Feasibility of the reactor conversion to low enriched uranium (LEU) fuel was studied in 2010-2012. Positive results of neutron-physical and thermal-hydraulic calculations allowed taking a decision about the reactor conversion.

Preparation work was conducted at the reactor ARGUS in the period 2012-2014 and LEU-fuel was prepared and loaded, reaching first criticality in July 2014. LEU-fuel was produced by mixing current HEU-fuel and fuel with 1.8% enrichment. Mixing was conducted in the reactor ARGUS vessel. As a result the following fuel characteristics were obtained: enrichment 19.8%, uranium concentration 380 g/l, volume 25.7 l.

Work was conducted in cooperation with ANL of the USA and supported by USDOE under the Global Threat Reduction Initiative.

The next stage of work is to get a new license of Rostechnadzor and conduct experimental study of the LEU-fuel reactor ARGUS characteristics.

9.2 Current Status of Conversion at the WWR-K Research Reactor

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Activities on conversion of the WWR-K research reactor at the Kazakhstan Institute of Nuclear Physics to LEU fuel are in progress. On the base of calculations, uranium dioxide in aluminum matrix of the uranium density 2.8 g/cm^3, enriched in U-235 to 19.7%, was chosen as fuel composition; a new design of the eight-tube fuel assembly (FA) with thin-walled fuel elements (1.6 mm), named as VVR-KN, was developed.

Three lead test assemblies (LTAs) were fabricated at the Novosibirsk Chemical Concentrate Plant. The LTAs passed through irradiation trials (average burnup above 40% guaranteed by manufacturer), after which VVR-KN fuel was approved for use in the WWR-K reactor core. New FAs were used for conversion of the critical assembly, which models the WWR-K reactor core. In the critical assembly, the critical load and the working load were determined experimentally, as well as its excess reactivity and worthies of control rods. Experimental results are in good agreement with calculations.

At present, the reactor safety analysis report is under development, new instrumentations and CPS drives and control rods are being fabricated, and the emergency cooling system is being upgraded.

Startup of the WWR-K reactor with low-enriched fuel is expected in 2015. This work is supported by the US Department of Energy under the Global Threat Reduction Initiative.

9.3 Progress in Safety Assessment of the IR-8 Reactor During Conversion to LEU Fuel

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As a part of the RERTR program NRC KI is performing the studies to establish the feasibility of converting the IR-8 research reactor to low enriched uranium (LEU) 19.7% enrichment of U^{235} fuel with financial support from U.S. Department of Energy. In 2012 after the completion of the IR-8 conversion feasibility study the work to assess safety of IR-8 reactor during conversion to LEU fuel was begun. Conversion is possible using the IRT-3M fuel assemblies (FAs) with U-9% Mo LEU fuel instead of high enriched uranium (HEU) FAs with UO_2. Now the IRT-3M FA with LEU is under development and licensing of U-9% Mo fuel. The main neutron and thermal-hydraulic characteristics of the core during conversion were determined. The radiation safety analysis for the IR-8 reactor during normal operation and possible accidents was finished. Currently analysis of the postulated accidents consequences for design basis accidents (DBA) is completed and continuing for beyond DBA.

9.4 HEU/LEU IGR Reactor Kinetics

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This paper presents results of solving the problem of IGR reactor kinetics implemented within the framework of the project on fuel enrichment reduction. IGR reactor is owned by National Nuclear Center of the Republic of Kazakhstan (NNC RK) and operated since 1961. IGR is a pulse research nuclear thermal reactor with a homogeneous uranium-graphite reactor core of heat capacity type. Its efficiency is governed by maximum temperature of core. The main modes of operation of the reactor are the self-quenching neutron burst mode and the regulated mode.

The possibility of the reactor conversion from high to low-enriched uranium (HEU, LEU) fuel was shown by calculations of reactor parameters in stationary critical conditions. For modeling dynamic modes of the reactor the "kinetic" account code was created. This code has been verified on parameters of neutron bursts of IGR-HEU reactor. The results of modeling of IGR kinetics show that from the point of view of neutron physics IGR-LEU reactor has similar dynamic characteristics of IGR-HEU reactor.

SESSION 10
Safety Analysis

Chair: Lawrence E. Kokajko

10.1 Comparative Safety Analysis of the MIR.M1 Reactor with Reference to Two Types of Low Enriched Fuel

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Presented here are the results of calculation data analysis of the MIR.M1 accidents with the most conservative initiating events with reference to two types of low enriched fuel, i.e. UO_2+Al and U9%Mo+Al. Analysis was performed for the loss-of-heat removal and positive reactivity insertion accidents resulted in the worst case probable consequences. According to the results of calculation data analysis, the instantaneous break LOCA leads to melting fuel assemblies with the highest power density as to both types of nuclear fuel. The positive reactivity-induced accident does not cause any fuel failure. This experiment was conducted with funding from the Argonne National Laboratory (USA).

10.2 Analysis of Beyond Design Basis Accident for Conversion of IRT MEPhI Research Reactor to LEU Fuel

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This paper presents the results of the analysis of Beyond Design Basis Accident − fuel channel blockage (drop of foreign object into the inlet of the core). It is assumed that one fuel assembly will melt for this accident and radiological consequences are analyzed. The core with tube-type low enriched uranium (LEU, 19.7 w/o, U-9%Mo) fuel and oxide high enriched uranium (HEU, 90 w/o) fuel is investigated. This work is the part of safety analysis for conversion of the IRT MEPhI research reactor to LEU fuel.

10.3 Analysis of Beyond DBA Consequences of the IR-8 Reactor Primary Pipes Rupture During Conversion to LEU

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As a part of the RERTR program NRC KI is performing the studies to establish the feasibility of converting the IR-8 research reactor to low enriched uranium (LEU) 19.7% enrichment of U^{235} fuel with financial support from U.S. Department of Energy. In 2012 after the completion of the IR-8 conversion feasibility study the work to assess safety of IR-8 reactor during conversion to LEU fuel was begun. Currently the analysis of beyond DBA consequences is carried out. The analysis of reactor primary pressure and suction pipes rupture consequences is finished. Calculations of full instantaneous rupture of pressure pipes for initial loading of the core with HEU, four mixed loadings, first and equilibrium loadings with LEU were performed using one-dimensional BEREZA code. For confirmation of the obtained results the same calculations for initial loading of the core with HEU and equilibrium loading with LEU were performed using system ATHLET code.

10.4 Loss-of-Flow Simulations for the Conversion of BR2 to Low Enriched Uranium Fuel

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The RELAP5/Mod 3.3 code is being utilized for the safety analyses of a representative core configuration of the BR2 research reactor for conversion from Highly-enriched to Low Enriched Uranium fuel (HEU and LEU, respectively). Flow distribution and pressure drop data from a mock-up facility and the BR2 reactor, both with several core configurations, were used to determine generic minor loss coefficients for the reactors flow channels. The model was then applied to the 1963 BR2 loss-of-flow experiments and shown to obtain good agreement with the measured peak cladding temperatures. Following this, the model was applied to the representative core and used to predict cladding temperatures for HEU and LEU fuel at nominal conditions. Future work includes repeating the 1963 BR2 loss-of-flow experiment conditions with the representative core configuration for both fuel types.

SESSION 11
Fuel Development – Fabrication Technology

Chair: Christophe Jarousse

11.1 Fabrication and Qualification of U_3Si_2 LEU Fuel Assemblies with Extrusion Technology for High Flux Reactor Petten

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The manufacturing of silicide-based LEU fuels for research reactors is currently mostly relying on rolling technology. During the last years, NCCP and NRG collaborated on developing and qualifying LEU silicide fuels, manufactured by extrusion technology. Two prototype fuel assemblies were manufactured by NCCP and qualified for use in the High Flux Reactor (HFR) at Petten.

Before irradiation, the thermohydraulic resistance and reactivity were measured and compared with a reference HFR fuel assembly. The assemblies were then irradiated for 15 irradiation cycles, corresponding to 387 full power days.

In this irradiation time, they reached an average burn-up of over 72-73% U-235. Coolant gap measurements, as well as detailed visual inspections were performed after each irradiation cycle. These showed good irradiation behavior, similar to reference HFR fuel assemblies. Early 2014, the successful qualification was approved by the Dutch regulatory body.

The manufacturer’s qualification of extrusion-based silicide fuels opens opportunities to exploit the advantages of extrusion technology towards silicide-based LEU fuels for other research reactors, as well.

11.2 A Review of Fabrication Technologies for the MP-1 Experiment

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The Fuel Fabrication Capability (FFC) is responsible for establishment of a fabrication process for low-enriched uranium-molybdenum fuel (LEU-Mo) currently under development. A key component of the fuel qualification program is the Mini-Plate-1 (MP-1) irradiation test. The MP-1 test will evaluate performance as a function of fabrication process and resulting initial microstructure using a systematic approach to specifying an initial fuel microstructure, properties, and other design features that favor acceptable fabrication criteria, adequate performance, and satisfactory blister threshold temperature. This testing ensures that fabrication / microstructure / performance relationships for monolithic LEU-10Mo fuel are understood. The objective of the test is to provide information on fuel performance, in concert with fabrication studies, to down-select an LEU fuel design that meets performance and commercial fabrication requirements. The FFC is tasked with fabrication of products to support the MP-1 experiment. This presentation will focus on technologies that will be pursued in the coming year as part of the MP-1 fabrication effort.

11.3 Y-12 National Security Complex U-Mo Fabrication

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Y-12 National Security Complex (Y-12 NSC) participates in the Fuel Fabrication Capability (FFC) pillar of the U.S. Department of Energy's (DOE) National Nuclear Security Administration (NNSA) Global Threat Reduction Initiative (GTRI) Convert Pillar system. Y-12 NSC is primarily responsible for the establishment of the fabrication process of a low-enriched uranium-molybdenum (LEU-Mo) feedstock. The baseline LEU-Mo fabrication process included a two-step casting process. Y-12 NSC is examining the feasibility of transitioning to a single step casting process. This presentation will focus on the strategy of transitioning to a single step casting process and the initial data from the feasibility trials.

11.4 LANL Progress on U-Mo Fuel Fabrication Process Development

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Recent progress at LANL in process development, prototype fabrication and alternate process development for monolithic LEU fuel will be discussed. Master alloy manufacture using vacuum arc re-melting (VAR) has produced several homogenized DU-Mo ingots. The effects of mold design are demonstrated for several designs of the triple plate mold. The feasibility of using HIP cans fabricated by sheet forming is demonstrated and the fabrication cost advantages are discussed. A new fixture for solid state bonding of electron beam welded aluminum cladding is shown and results of its use are discussed. Comparison of a finite element model with experimental HIP can measurements is made. The effect of foil annealing temperature on the microstructure of Zr/U-Mo interfaces is shown. The as-cast surfaces of DU-Mo machined by electric discharge machining are shown and possible advantages in material utilization are discussed.

11.5 Protective Coatings for Long Term Wet Storage of Spent Aluminum-clad Research Reactor Fuel

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Pitting corrosion of the aluminum cladding of spent research reactor (RR) fuels in wet storage has been reported and attributed to certain water parameters acting synergistically. Hence, use of conversion coatings was proposed to protect the Al cladding of spent RR fuel during long term wet storage. This paper presents the results of: (a) preparation and corrosion characterization of boehmite and hydrotalcite (HTC) coatings on AA 1100 and AA 6061 alloys; (b) field studies in which boehmite and HTC coated Al alloy coupons as well as full size plates assembled in a dummy fuel element (with or without post coating treatments) were exposed to the IEA-R1 RR spent fuel basin for a year. The cerium modified HTC coating imparted the highest corrosion resistance. The extent of corrosion protection of the different coatings was compared and this paper discusses the mechanism by which the alloy's corrosion resistance was increased. Scale-up of the overall coating processes to protect spent RR fuel elements will be also presented and discussed.

11.6 Performance and Fabrication Status of TREAT LEU Conversion Conceptual Design Concepts

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Resumption of transient testing at the TREAT facility was approved February 2014 to meet U.S. Department of Energy objectives. The National Nuclear Security Administration's Global Threat Reduction Initiative Convert Program has established a program to convert TREAT from its existing highly enriched uranium core to a new low enriched uranium (LEU) core using the existing core to test and qualify the LEU fuel. The screening decisions for the initial pre-conceptual designs are briefly described with more detailed discussions on current feasibility, qualification and fabrication approaches. Feasible fabrication will be shown for a LEU fuel element assembly that can meet TREAT design, performance, and safety requirements. The statement of feasibility recognizes that further development, analysis, and testing must be completed. Engineering challenges such as cladding oxidation, high temperature material properties, fuel block fabrication along with neutronics performance, will be highlighted. Preliminary engineering evaluations provided confidence that conceptual designs could be achieved.

SESSION 12
Conversion Analysis and Methods

Chair: Yoann Calzavara

12.1 IVG.1M Reactor Kinetics

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This paper represents the results of problem solution related to the IVG.1M reactor kinetics being implemented within the Program on Conversion of the Research Reactors to LEU fuel. Being operated since 1972, the IVG.1M reactor is currently subordinated to the National Nuclear Center of the Republic of Kazakhstan (NNC RK). IVG.1M is an updated, high-temperature, gas-cooled channel and light water-cooled channels, water-moderated, thermal reactor with beryllium reflector. The core of the reactor has water-cooled technological channels consisting of two-ribbed fuel elements.

Kinetics calculation code (KCC) was created to provide calculations of IVG.1M with high-enriched and low-enriched uranium (HEU, LEU) fuel under various scenarios which cause positive and negative reactivity feedback. KCC is verified based on the experimental data having been obtained in IVG.1M transient power startup. From these verification tests it may be concluded that the KCC properly describes IVG.1M core behavior affected relation

between the reactor power and reactivity feedback. The KCC is recommended to make calculations for safety case of IVG.1M conversion.

12.2 Comparative Validation of Monte Carlo Codes for Conversion of IRT MEPhI Research Reactor to LEU Fuel

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In the framework of the conversion feasibility studies for 2.5 MW pool type research reactor IRT MEPhI of the National Research Nuclear University MEPhI the detailed neutronic core analysis using different Monte Carlo codes was performed. In order to validate the obtained results the comparison of these codes on some test problems calculation was carried out. The test problems for IRT-type reactor with tube-type low enriched uranium (LEU, 19.7 w/o, U-9%Mo) fuel and oxide high enriched uranium (HEU, 90 w/o) fuel were developed. The static cases and the depletion problem were examined. The calculations have been performed using continuous energy Monte Carlo codes: MCNP (+MCREB for burnup calculation) and MCU-PTR. The impact of cross-section libraries used for a particular problem on the calculated results was investigated. Calculated results for IRT MEPhI operational core with HEU fuel and fresh and operational core with LEU fuel are also presented.

12.3 Loss-of-Offsite-Power Simulations for the Conversion of RHF to Low Enriched Uranium Fuel

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The code RELAP5/Mod 3.3 was used to simulate a loss-of-offsite-power accident in the RHF research reactor for highly-enriched and low enriched uranium fuels (HEU and LEU fuels, respectively). The steady-state nominal condition results (initial conditions) were in agreement with the known temperature, pressure and flow distributions. Qualitatively the transient behavior of the reactor was found to reasonably agree with previous simulations performed with the CATHARE code. Good agreement was obtained for the magnitude in peak fuel cladding temperature but its evolution differed due to modeling choices in discretizing the core coolant volumes. The magnitude in natural circulation flow for RELAP5 was comparatively lower due to the specified minor loss coefficients at tee junctions but their justification could be inferred from the pump coast down measurements. These preliminary simulations of the RHF reactor show that the fuel type had little impact on the loss-of-offsite-power accident scenario.

12.4 Evaluation of Numerical Methods for Transient Thermal-Hydraulic Reactor Analysis

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Improvement in computer speeds, usage of computer networks, and availability of newer computer codes allows one to bring a large amount of computational power to the analysis of transients in reactors. Using these many resources is not always practical on a day-to-day basis. Instead one may be limited to using a system-level code which less spatial detail. Two Computational-Fluid-Dynamics (CFD) codes STAR-CCM+ and ANSYS have been used to model the transient behavior of a twisted-pin fuel element and compared with results obtained using system-level code RELAP5-3D.

12.5 Validation of the Fuel Plate Swelling Models used in the Multi-Physics Simulation of the E-FUTURE-1 Fuel Irradiation Experiment

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The swelling of the fuel plates used in research reactors can have a significant effect of the maximum fuel temperature reached during the irradiation cycle and can lead under certain conditions to cladding blistering and plate failure. The plate swelling is the result of multiple physical phenomena, including fuel burnup, interaction layer formation, and oxide layer growth, which in turn affect the fuel and cladding temperatures and the fission gas pressure. As detailed fuel plate swelling measurements became available from the E-FUTURE-1 experiment and 3-D multi-physics analyses of this experiment were performed using more accurate and experimentally validated fuel behavior models, it is possible to validate the fuel plate swelling models in the context of the E-FUTURE-1 experiment analyses. A Computational Fluid Dynamics (CFD) model based on the STAR-CD code and the SIMDIF Fuel Behavior (FB) model for the simulation of UMo dispersion are used at Argonne National Laboratory (ANL) for the multi-physics analysis of the E-FUTURE-1 fuel irradiation experiment conducted in the BR2 reactor at SCK•CEN in Belgium. The fuel plate swelling results obtained from a multi-physics simulation of the E-FUTURE-1 three irradiation cycles are compared with the corresponding experimental results obtained from post-irradiation plate swelling measurements. Plate swelling model enhancements including a simplified cladding creep model are shown to improve significantly the agreement between the measured and calculated plate swelling results. The relationships between the fuel plate swelling, oxide layer

growth, and the fuel meat and cladding temperatures during the E-FUTURE-1 experiment are examined.

SESSION 13
Poster Session III: Conversion Analysis and Safety Licensing

Chair: John Holland

13.1 Nigeria Research Reactor (NIRR-1) Conversion Programme Implementation - A Regulatory Approach

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Nigeria is embarking on a fuel conversion program in view of her being a signatory to the Non-Proliferation Treaty and by extension, acceptance of the non-proliferation program which encompasses a global effort to convert her type of research reactor to low enriched uranium fuel. The NIRR-1 is a 31kW miniature neutron source reactor situated at the Centre for Energy Research and Training Ahmadu Bello University, Zaria. The reactor was acquired through the tripartite project and supply agreement between the Federal Government of Nigeria, International Atomic Energy Agency and China Institute of Atomic Energy. The reactor attained criticality on the 3rd February 2004 and has since been used for Neutron Activation Analysis, Experiments and Training in Nuclear Science and Technology. NIRR-1 uses U-235 fuel enriched to about 90.2%.

This paper analyzes the status of NIRR-1 conversion program from a regulatory perspective, especially the major milestones fulfilled towards the submission/review of the feasibility/leu conversion report. The paper considers the legal framework including the Act, Regulations and guidance documents developed or in the process of development for effective regulation of the conversion from project schedule to shipping requirements. The various international instruments endorsed by Nigeria as a demonstration of her commitment to conversion program in form of Treaties, Conventions and Agreements are highlighted. The status of the draft Regulations on Research Reactors and the key elements of the Regulations are discussed.

13.2 Practical Application of the Graded Approach on the Safety of NIRR-1 HEU to LEU Core Conversion

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Objective: (1) To bring out the Fundamental safety requirements of Research Reactors using the Graded approach, to justify the safety and security of NIRR-1(LEU) Core conversion and to ensure that the same Safety Objectives and Engineering design requirements are met, (2) Graded approach is a method in which the stringency of the design measures and analyses applied are commensurate with the level of risk posed by the reactor facility.

Safety justification & conclusion: (1) Increase ratio of neutron flux in the irradiation site to the core thermal power, (2) The LEU has a better shut down margin, because it has a central control rod worth of 7.7mk, (3) Fuel integrity & Dose to public are maintained under all operating conditions, (4) Reactivity coefficients meets required limits and are comparable to the existing HEU core, (5) The melting temperatures of the LEU fuel meat and cladding has a better safety margin.

13.3 Vibration Diagnostics of Cooling Circuit in Maria Reactor Before and After Fuel Conversion from HEU to LEU

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Fuel conversion has increased 20% of flow rate per fuel channel and forced replacement of the main pumps in cooling circuit. Long term observation of pipeline vibrations provide additional information about state of new pumps assembly and its influence on cooling installation. State of installation has been evaluated by comparison archival measurements of power spectrum density of acceleration and accepted limits of root mean square velocity (Vrms). Acceleration is measured by sensors grouped in pairs at an angle 90 degrees in 3 points of pipeline. Measurement includes 100 sets of 1 second acceleration signals. Before conversion of reactor core we have created database of power spectrum density and Vrms. The parameters are still stored and changes are observed. Comparison of archival results with present measurements allows early detection of changes in cooling installation.

13.4 Reversal of OFI and CHF in Research Reactors Operating at 1 to 50 Bar

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The condition at which the critical heat flux (CHF) and the heat flux at the onset of Ledinegg flow instability (OFI) are equal, is calculated for a coolant channel with uniform heat flux as a function of five independent parameters: the channel exit pressure (P), heated length (L_h), heated diameter (D_h), inlet temperature (T_{in}), and mass flux (G). A diagram is made by plotting the mass flux and heat flux at the OFI-CHF intersection (i.e., reversal from CHF > OFI to CHF < OFI as G increases) as a function of P (1 to 50 bar), for 36 combinations of the remaining 3 parameters (L_h , D_h , T_{in}). The application of the diagram *to scope* whether a research reactor is OFI-limited (below the curve) or CHF-limited based on the five parameters of its most-limiting coolant channel is described. Justification for application of the diagram to research reactors with axially non-uniform heat flux is provided.

In order to make the OFI-CHF intersection diagram, two world-class CHF prediction methods (the Hall-Mudawar inlet-conditions correlation and the extended Groeneveld 2006 Table) are compared for 216 combinations of the five independent parameters. Also, two widely used OFI correlations (the Saha-Zuber and the Whittle-Forgan with $\eta = 32.5$) are compared for the same combinations of the five parameters. The extended Groeneveld Table and the Whittle-Forgan OFI correlation are selected and used in making the diagram. Using the five design parameters, the operating state of any research reactor can be located on the reversal diagram that will readily show whether CHF or OFI is most-limiting. The scoping results of the OFI-CHF diagram for five research reactors (ATR, HIFR, MITR, MURR, and the ANS Design) are found to agree with the results reported by their owners. Due to its limitations (uncertainties not included), the diagram cannot replace the detailed thermal-hydraulic analysis required for a reactor safety analysis.