# SESSION ABSTRACTS

## SESSION 1

## Opening Remarks

MC: Chris Landers

### Opening Remarks

**Ambassador W. Robert Kohorst**
U.S. Embassy Zagreb, Ulica Thomasa Jeffersona 2, 10010 Zagreb – Croatia

### Welcome to 40th RERTR International Meeting

**Paul Kearns**
Director, Argonne National Laboratory, 9700 S. Cass Ave., Lemont, IL 60439 – USA

## Global Progress in HEU Minimization

Session Chair: Chris Landers

### 1.1 NNSA Nonproliferation Initiatives and Partnerships

**Brent Park**
Deputy Administrator for Defense Nuclear Nonproliferation, US Department of Energy's National Nuclear Security Administration, 1000 Independence Ave., Washington DC 20585 – USA

### 1.2 International Atomic Energy Agency Efforts to Promote Nuclear Nonproliferation

**Christophe Xerri**
Director, Division of Nuclear Fuel Cycle, Waste Technology and Research Reactors, IAEA, International Atomic Energy Agency Vienna International Centre, PO Box 100 A-1400, Vienna – Austria

## SESSION 2

## HEU Minimization Structures Plenary

Session Chair: John Stevens

### 2.1 Reactor Conversion: By the Numbers

**Chris Landers**
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### 2.2 Nuclear Material Removal Collaboration with Research Reactor Conversions

**Scott Roecker**
US Department of Energy's National Nuclear Security Administration, 1000 Independence Ave., Washington DC 20585 – USA

### 2.3 IAEA and RERTR: A History of Collaboration

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The IAEA has been involved for more than thirty-five years in supporting international nuclear non-proliferation efforts associated with reducing the amount of highly enriched uranium (HEU) in use in civilian research reactors and related activities. IAEA fuel cycle projects have directly supported the Reduced Enrichment for Research and Test Reactors (RERTR) program, through dissemination of information related to fuel conversion from HEU to low enriched uranium (LEU), HEU fuel shipments, and reduction of HEU in other applications. These activities are completed in conjunction with the requesting Member States, and in collaboration with other international experts. This paper presents some of IAEA's HEU minimization efforts that have been completed during the past three decades, as well as look at IAEA's ongoing and planned activities to support the global objective of reducing the use of HEU in civilian applications.

# SESSION 3

## Life After Conversion I

Session Chair: Sunday Jonah

### 3.1 Technical Highlights from 40 RERTR Conferences and Reactor Conversions

**John Stevens**
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The effort to eliminate Highly Enriched Uranium (HEU) from civilian commerce wherever possible has required a wide array of technical advancements, from materials science for higher density Low Enriched Uranium (LEU) fuels and isotope-production targets, through modeling methods development for the neutronics and thermal-hydraulics of safety analyses, to implementation infrastructure and processes that have leveraged similarities among classes of research reactors. Since the beginning, the effort – and the successes along the way – have been based upon collaborations among many organizations in many nations, facilitated by the annual RERTR Conferences. This paper will provide a brief review of notable progress in reactor and Mo-99 conversion, as the foundation for shared reactor conversion successes in the future.

### 3.2 Framatome-CERCA™ Successes in Research Reactor Conversions

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CERCA™ is the Framatome division dedicated to Research Reactor Fuel and Mo-99 medical target production. It is one of the major actors in metallic fuel manufacturing and transport in the world. Since 1957, CERCA™ has produced about 1,000,000 fuel plates and 6,000 fuel assemblies; and is involved in international conversion programs conducted by Europe, US DoE or IAEA. It is a major partner for HEU minimization and Research Fuel conversions.

This paper focuses on CERCA™ contribution to HEU minimization in the world in the past, today and for the future: LEU $U_3Si_2$ conversion, LEU UZrH TRIGA fuel conversion and LEU UMo developments. Many successes will be reviewed and ongoing challenges will be presented.

## 3.3 Status of Hungary after the HEU-LEU Conversion

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The Budapest Research Reactor, the first nuclear facility of Hungary used HEU fuel from 1966 to 2012. The reactor was converted to LEU with the help and kind assistance of the RERTR Program. The spent HEU fuel and all the unused nuclear fuel was shipped back to the country of origin, i.e. to Russian Federation in the frameworks of the RRRFR Program between 2008 and 2013. After the last shipment was completed, Hungary was declared to be free of HEU. The removal of the entire inventory of HEU from the country represents a substantial contribution to nuclear risk mitigation.

The operation of the Budapest Research Reactor was planned up to 2023, the technical conditions allow even to extend this period for ten more years. During and after the core conversion there was no problem with the LEU fuel assemblies, only a small penalty had to be paid in neutron flux, there was no significant difference for the users.

In this year the procurement of the fresh fuel assemblies for the next seven years was arranged. Fuel diversification is in progress, in cooperation with Technicatom (France) a new LEU type fuel is in development.

As Hungary has an intensive extension program in nuclear energy, i.e. new NPP units are planned, there is a hope that the government will support the use of the research reactor on the long term.

# SESSION 4

## Life After Conversion II

Session Chair: Ross Finlay

## 4.1 Towards a HEU-free World: INVAP's Experience, Challenges, Milestones, and Perspective

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INVAP S.E., Av. Cmte. Luis Piedrabuena 4950 (R8403CPV), S. C. de Bariloche, Rio Negro – Argentina

During the fourth edition of the Nuclear Security Summit, INVAP, together with CNEA, received the Industrial Innovation Award from the Nuclear Energy Institute recognizing the work done towards the reduction of enrichment in research reactors through numerous institutional and commercial projects.

INVAP projects range between the design and construction of high performance LEU-fueled reactors for different purposes (including Mo-99 production from LEU targets), core conversion from HEU to LEU, the design and construction of associated facilities for the manufacturing of LEU fuels and targets and the processing of radioisotopes from LEU targets and exports of LEU fuels and LEU targets.

INVAP is a recognized nuclear vendor of this technology at local, regional and international levels, with presence in five continents.

This paper presents the contribution of INVAP to enrichment reduction as a backbone of its business and summarizes the experience and challenges faced along the different projects, the milestones reached throughout its 43 years of existence and the perspective for the future identifying the strategy for coming ventures.

## 4.2 Support of ÚJV Řež, a. s. for Shipments of Spent Nuclear Fuel within the $M^3$ Program

**Josef Podlaha**, Karel Svoboda

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In 2005, ÚJV Řež, a. s. (ÚJV, the former Nuclear Research Institute Řež), Czech Republic joined the Russian Research Reactor Fuel Return (RRRFR) program under the US-Russian Global Threat Reduction Initiative (GTRI) (now Material Management and Minimization program, $M^3$). The primary goal of the program is to advance nuclear nonproliferation objectives encouraging eligible countries to convert their research reactors from highly enriched uranium (HEU) to low enriched uranium (LEU) fuel and eliminating stockpiles of HEU.

ÚJV`s daughter company, Research Centre Řež, operates the LVR-15 research reactor with output of 10 $MW^{th}$, which has been in operation since 1957. After more than 50 years of operation of the reactor, a large amount of spent nuclear fuel (SNF) of Russian origin had been accumulated.

The ŠKODA VPVR/M high capacity casks were used for SNF shipment from the LVR-15 research reactor back to the Russian Federation (RF). The ŠKODA VPVR/M cask is a type B(U) and S cask system with a unique design that allows for easy use at almost any research reactor facility. The cask is closed by means of a system of two upper and two lower lids. The cask is loaded from the bottom, being placed above the SNF storage pool. It eliminates the need for a transfer cask, thereby reducing the number of manipulations and increasing the level of nuclear safety and radiation protection. However, the transfer cask may be used too, if necessary. The cask has a capacity of 36 fuel assemblies, and 16 casks are available. This means that 576 FA can be transported in one shipment.

Two SNF shipments of HEU and LEU SNF from ÚJV were realized in 2007 and 2013. The combined road and railway transport of sixteen casks with 549 fuel assemblies (Czech Republic – Slovakia – Ukraine – RF) took place in 2007. The second shipment (six casks with 112 fuel assemblies) was carried out in 2013. Combined road, railway and marine transport (Czech Republic - Poland - RF) was used. The shipments were realized with significant technical and financial aid from the US administration and US DOE. After the shipments were completed, only LEU nuclear fuel remained on the territory of the Czech Republic.

Proven to be an excellent and very sophisticated system for SNF transportation, the ŠKODA VPVR/M casks were further employed for SNF shipments from Bulgaria, Hungary, Poland, Ukraine, Belarus, Serbia, Vietnam, Uzbekistan and Georgia to Russia. Some shipments were carried out as air shipments with use of the TUK-145/C cask system developed in Russia that included the ŠKODA cask.

In 2016, the new ŠKODA MNSR casks were developed on the base of the ŠKODA VPVR/M casks in the Czech Republic and currently are being employed in the shipments of HEU cores of Chinese Miniature Neutron Source Reactors (MNSR) to China. The TUK-145/C-MNSR cask system developed in Russia is used for transportation. The shipment from Ghana was carried out in 2016 and the shipment from Nigeria was carried out in 2018. Shipments from Syria, Iraq and Pakistan are planned.

ÚJV`s participation in shipments of SNF from other countries comprises leasing of ŠKODA VPVR/M or MNSR casks, including service and maintenance inspections of the casks, transportation of the empty casks, providing cask documentation, training of personnel in cask use and SNF loading, technical oversight and expertise during the cask handling, fuel loading and cask closing and sealing, the drying and helium leak testing of casks and the return transportation of the empty casks.

Seventeen shipments from twelve countries using a total of 110 ŠKODA casks have already been completed without any incident or accident and more than 3500 fuel assemblies or cores have been shipped to Russia and China.

Summary

In 2005, ÚJV Řež, a. s. (ÚJV), Czech Republic joined the Russian Research Reactor Fuel Return program under the US-Russian Global Threat Reduction Initiative and started the process of spent nuclear fuel (SNF) shipment from the LVR-15 research reactor to Russia using the ŠKODA VPVR/M casks. Two SNF shipments from ÚJV were carried out in 2007 and 2013.

ÚJV also participates in shipments of SNF from other countries. Seventeen shipments from twelve countries have already been completed without any incident or accident and more than 3500 fuel assemblies have been shipped to Russia and China. When the Party Ends: Maintaining Support for Reactor Conversion in Challenging Times

### 4.3 Comparison of Thermal Hydraulics Characteristics of NIRR-1 LEU Core with HEU Core

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The Nigeria Research Reactor-1 (NIRR-1) has been converted from HEU to LEU and it is therefore, the third Miniature Neutron Source Reactor (MNSR) currently fueled with $UO_2$ pellets enriched to 13%. Like the HEU cores of MNSR facilities, a major safety characteristic is the self-power regulating capability even with the single control completely withdrawn and the reactor allowed to operate unattended to. This was tested in the NIRR-1 LEU core during the on-site Commissioning. NIRR-1 was operated in the manual mode with all safety and protection features disabled so as to monitor the transient behavior of the facility with the insertion of total cold core excess reactivity of 3.94 mk. Results obtained indicate that the reactor power reaches a peak value of 70 kW and power gradually reduces over the entire transient time of 3500 seconds. Results of measured and simulated reactivity insertion transients of 3.94 and 3.0 mk for NIRR-1 LEU core compare well with data obtained for the HEU core and will be used to develop a PARET LEU model that would be used for the analyses of reactivity insertion transient in MNSR facilities and low power research reactors with similar thermal hydraulics characteristics

# SESSION 5

## HEU Removal Operations and Fuel Transportation

Session Chair: Scott Roecker

### 5.1 History of the U.S. Origin Program and Its Current End State

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The United States (U.S.) Department of Energy (DOE), in consultation with the Department of State, adopted the Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel in May 1996. The policy, which expired on May 12, 2019, provided Foreign Research Reactors (FRRs) an opportunity to return spent nuclear fuel (SNF) to the United States.

The goal of the program was to repatriate certain U.S.-origin SNF and other weapons-grade nuclear material. Managed by the National Nuclear Security Administration's Office of Material Management and Minimization, the program was successful in reducing the amount of highly enriched uranium in civil commerce by returning U.S.-origin SNF to the United States for secure storage, management, and disposition. This paper provides an update on the program and discusses program accomplishments and current initiatives.

## 5.2 Current Status of the Russian Research Reactor Fuel Return Program

**I. Bolshinsky**, J. Dewes, S. Moses, K. Bateman
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In 1999, the U.S. Department of Energy (DOE), together with the International Atomic Energy Agency (IAEA) and the Russian Federation, agreed to start a new nuclear non-proliferation initiative to return Russian/Soviet-origin highly enriched uranium (HEU) back to its country of origin. Trilateral discussions among the U.S., the Russian Federation, and the IAEA identified more than 20 research reactors in 16 countries (Belarus, Bulgaria, China, Czech Republic, DPRK, Germany, Hungary, Kazakhstan, Latvia, Libya, Poland, Romania, Serbia, Ukraine, Uzbekistan, and Vietnam) that had Russian/Soviet-origin HEU. In 2000, the IAEA Director General sent a letter to 16 countries asking for their willingness to return this spent HEU fuel to the Russian Federation. Fourteen countries responded positively to the Director General's letter.

The primary goal of the Russian Research Reactor Fuel Return (RRRFR) program, managed by DOE's National Nuclear Security Administration, Office of Material Management and Minimization, is to advance nuclear nonproliferation objectives by encouraging eligible countries to convert their research reactors from HEU to low-enriched uranium (LEU) fuel and to eliminate stockpiles of HEU. In May 2004, the United States and the Russian Federation signed a Government-to-Government Agreement that established the legal framework necessary for cooperation between the United States and the Russian Federation for the return of research/test reactor fuel from eligible countries.

To date, the RRRFR program has completed sixty-three successful operations to repatriate almost 2,300 kg of Russian-origin HEU fresh and spent fuel. All Russian-origin HEU has been completely removed from 12 countries: Bulgaria, the Czech Republic, Hungary, Georgia, Latvia, Libya, Poland, Romania, Serbia, Ukraine, Uzbekistan, and Vietnam. At the present time Russian-origin HEU remains in Kazakhstan, Belarus, Germany, DPRK, and China. This paper describes the current status of the RRRFR program and preparations for upcoming HEU removal and down-blending activities.

## 5.3 NAC's OPTIMUS™ Packaging for Research Reactor Wastes

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NAC OPTIMUS™ packagings are a Type B(U)F-96 transportation package designed for maximum flexibility and cost-efficiency to support shipment of a wide range of challenging reactor wastes and materials. They are small, modular packaging meeting DOT weight limits for road transport. Contents can be accommodated in multiple configurations including standard drums of up to 416 liters in size. The packagings also accommodate unique waste contents and fuel designs where cost-effective packaging options have been a challenge. The packagings have a simplicity of design and operational flexibility to meet both IAEA and U.S. NRC Type B requirements. Flexibility is assured by the modular design for containment, shielding and criticality control. The containment boundary design permits adaptability to content requirements by relying on interchangeable internal components or dunnage.

The OPTIMUS™ packaging utilizes the same cask containment vessel (CCV) in the OPTIMUS™-H and OPTIMUS™-L. The OPTIMUS™-L is a lightweight transportation packaging with a capability of up to ten (10) OPTIMUS™-L packages per truck shipment. The OPTIMUS™-H has an Outer Shield Vessel (OSV) made of ductile cast iron and Impact Limiters (IL) with a higher payload capacity then the OPTIMUS™-L. The large cavity size of the CCV, can accommodate standard drums up to 416 liters, combined with the small size and modularity of the OPTIMUS™ packaging provides unmatched flexibility for the user. OPTIMUS™ can accommodate more radioactive waste in each drum (up to a Fissile Gram Equivalent (FGE) limit of 395g Pu 239) than larger packages can in 10 drums.

In this paper, NAC provides the technical capabilities of the OPTIMUS™ packagings and identifies the design features and technology advancements making the OPTIMUS™ product line a readily adaptable and flexible solution for packaging reactor and decommissioning wastes, especially in smaller facilities with limited crane access.

## 5.4 Impacts to the Advanced Test Reactor Fresh Fuel Shipping Container (ATR FFSC) for the Transport of LEU Fuel

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The U.S. National Nuclear Security Administration (NNSA) Office of Conversion through the U.S. High Performance Research Reactor (USHPRR) Project is developing and qualifying a new low-enriched uranium (LEU) fuel to convert five research reactors and one critical assembly from the use of high-enriched uranium (HEU) fuel to LEU fuel. The ATR FFSC package (for unirradiated fuel element transport) was developed approximately a decade ago for the transport of the HEU versions of these fuel elements and its license has been amended over the last several years to accommodate shipment of various USHPRR experiments. In the future, test elements and ultimately full core loads of the LEU fuel elements will need to be transported. Differences between the HEU and LEU fuel need to be considered including the increase in fuel element mass and changes in fissile content. This paper discusses the potential impacts to the ATR FFSC for the transport of new LEU fuels.

## 5.5 Corrosion Protection of Spent Aluminum-Clad Research Reactor Fuel During Long Term Wet Storage

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Protection of the aluminum cladding of spent research reactor (RR) fuels was proposed to prevent pitting corrosion during long term wet storage. This paper will present the effect of HTC (a lithium aluminum-nitrate-hydroxide hydrate that forms on Al alloy surfaces immersed in an appropriate alkaline lithium salt solution) coatings prepared from different baths on AA 6061 alloy surfaces pre-treated to simulate spent fuel surface features. The results of field studies in which dummy fuel elements, consisting of Al alloy plates coated with HTC from different baths (with or without post coating treatments), were immersed in the IEA-R1 reactor's spent fuel basin for 30 months indicated that the cerium modified HTC coating imparted very high corrosion resistance. This paper will discuss the HTC corrosion protection mechanism and present a mock-up of the arrangement required to remotely handle and coat the dummy-fuel elements.

# SESSION 6

## Poster Session

Session Chair: Caryn Warsaw

## 6.1 Calculated Studies in Support of the Creation of a Uranium-Zirconium Hydride Critical Assembly with Low Enriched Uranium Zirconium Carbonitride Fuel

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Uranium zirconium carbonitride UZrCN is a high density, high temperature fuel with high thermal conductivity, which has potential for use in various types of reactors, comprising research reactors, including of conversion HEU or LEU fuel. Within the Russian Research Reactor Fuel Return Program the Joint Institute for Power and Nuclear Research – Sosny of the National Academy of Science of Belarus is developing a uranium-zirconium hydride critical assembly with LEU UZrCN fuel, which will be placed on the critical facility Crystal. The critical assembly is a hexagonal lattice comprising a core, containing fuel and control rods assemblies with stainless steel claddings, surrounded by reflector assemblies and units. The fuel material is uranium zirconium carbonitride ($U_{0.9}Zr_{0.1}C_{0.5}N_{0.5}$) with ~ 12.5 g/cm$^3$ density, ~ 11.5 g/cm$^3$ uranium density and 19.75% uranium-235 enrichment. The moderator is $ZrH_{1.89}$ with 5.08 g/cm$^3$ density. Neutron absorber in control rods - natural $B_4C$. Each fuel assemblies contains zirconium hydride and 3 fuel rods in a niobium or stainless steel cladding. Side reflector: inner layer - assemblies with zirconium hydride and stainless steel claddings, outer layer - stainless steel units. This paper presents the configuration of the core and the reflector, the material composition and geometric dimensions of the components of the critical assembly, and the results of calculating by the MCU-PD code.

## 6.2 USHPRR Fuel Fabrication Pillar Fabrication Process Status

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The Fuel Fabrication (FF) Pillar, a project within the U.S. High Performance Research Reactor Conversion program of the office of NNSA's Material Management and Minimization (M$^3$), is tasked with the scale-up and commercialization of the high-density monolithic U-10Mo fuel for the conversion of appropriate research reactors. The key steps of the fabrication process are baselined and will be utilized in each upcoming fabrication campaign. These keys process steps are 1) vacuum induction melting to cast U-10Mo ingot, 2) Hot-roll bonding of zirconium to ingot, 3) Cold rolling zirconium clad U-10Mo to finish in tolerance and 4) hot isostatic pressing to clad in aluminum. Future fabrication campaigns, such as Plate Demonstrations and Assembly demonstrations, are designed to gain additional process knowledge and further demonstrate the fabrication line capabilities. The results of these campaigns will support optimization studies, process modeling, and characterization activities which will provide continued process understanding.

## 6.3 Current Status on the Development of High-density LEU $U_3Si_2$ Fuel in KAERI

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As a part of a Reduced Enrichment for Research and Test Reactors (RERTR) Project, a lot of efforts for the conversion of high enriched uranium (HEU) fuel to low enriched uranium (LEU) fuel in research reactors have been made for last decades. Until now, HEU fuel has been converted to LEU fuel with a uranium density of 4.8 gU/cc using uranium silicide alloys such as $U_3Si$ and $U_3Si_2$ in more than 90% of the research reactors. However, the development of U-Mo LEU fuel for high flux research reactors (HFRR) using HEU fuel has been still in progress and some problems to be solved have been remained for using it in HFRR. Currently, high-density $U_3Si_2$ LEU fuel is being investigated for its use in HFRR as a backup solution of U-Mo LEU fuel. In order to use the LEU $U_3Si_2$ fuel at the HFRR, U loading density of each fuel plate should be increased from the commercial value of 4.8 gU/cc to the high-density value of 5.3 gU/cc. The atomized powder manufactured by KAERI provides sufficient minimum cladding thickness at the dog-bone area of a fuel plate due to its better mobility compared to comminuted powder in the hot rolling process. In this paper, manufacture and inspection results regarding a 5.3 gU/cc high-density LEU $U_3Si_2$ fuel plate and atomized $U_3Si_2$ powder are introduced and discussed.

## 6.4 Update of Regulatory Oversight of NIRR-1 Core Conversion.

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This paper contains details of safety assessments performed during the NIRR-1 core conversion from Highly Enriched Uranium (HEU) fuel to Low Enriched Uranium (LEU) fuel and the Lessons Learnt during the issuance of license for the NIRR-1 conversion programme; it also highlights the Inspection observations.

Some of the safety assessments done were: NIRR-1 vessel integrity inspection, subcriticality measurement of NIRR-1, operational and licensing procedures of HEU core discharge and LEU loading. The Nuclear Safety and Radiation Protection Act 19 of 1995 established the Nigerian Nuclear Regulatory Authority (NNRA). The NNRA has responsibility for Nuclear Safety and Radiological Protection Regulation in Nigeria. Among the activities regulated by the NNRA is the Miniature Neutron Source Reactor (MNSR) at CERT.

Conclusions
The inspection of inner and outer surfaces of vessel and neutron measurement was completed by CIAE team and CERT team, under the supervision of the NNRA.

## 6.5 European Developments for Monolithic UMo Fuel: UMo Foil Manufacturing Project

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## 6.6 Depleted Uranium Manufacturing Studies for KUCA LEU Conversion Fuels

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## 6.7 Y-12 National Security Complex LEU-Mo Casting Update

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The Y-12 National Security Complex (Y-12) participates in the Fuel Fabrication Pillar of the National Nuclear Security Administration's (NNSA's) Office of Material Minimization and Management (M3) Office of Conversion Pillar. Y-12's primary responsibility is to establish the fabrication process for the low-enriched uranium-molybdenum (LEU-Mo) feedstock. This update focuses on recent efforts to demonstrate consistency and repeatability in the LEU-Mo fabrication process for a production environment.

## 6.8 First-Principles Study of Surface Properties of Crystalline and Amorphous Uranium Aluminides

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Surface energy is a crucial material property required to model gas bubble behavior in nuclear materials. In this work we systematically investigate the surface properties of uranium aluminides by first-principles calculations. A total of 13 surface orientations with a maximum Miller index (MMI) up to 3 are studied for cubic $UAl_3$, while 19 surfaces for orthorhombic $UAl_4$. Using the predicted surface energies, we obtain the surface properties of equilibrium single crystal $UAl_3$ and $UAl_4$, including surface area weighted surface energy, dominant surface orientations and surface anisotropy. To understand gas bubble behaviors in amorphous uranium aluminides, we study the surface properties of amorphous $UAl_3$ and $UAl_4$. Compared to the crystalline phases, the amorphous phases have lower surface energies. The currently obtained surface properties of uranium aluminides can be used for the future modeling of gas bubble behaviors in the interaction layer in UMo/Al dispersion fuel.

## 6.9 Microstructural Characterization of U-7Mo Dispersion Fuel Plates Irradiated at High Power

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The EMPIRE experiment was successfully irradiated in the Advanced Test Reactor. The purpose of the experiment is to generate data necessary for completion of the "Comprehension Phase" of the HERACLES international fuel development program. Microstructural characterization will play an important role for generating the required data from the EMPIRE irradiation experiment. Microstructural characterization has been performed on fuel plates from past mini-plate irradiation experiments to bolster the future information that will come from the EMPIRE irradiation experiment (e.g., plates have been irradiated in the past under different conditions that can be investigated to understand how fission rate/temperature affects swelling). The data from historical fuel plates, which were irradiated under relatively aggressive conditions, allow for an improvement of the understanding of fuel performance, in terms of identifying what needs to be “fixed” with respect to fuel performance in order to identify a dispersion fuel system that can survive during irradiation to relatively high fission density without failure. It is of interest to determine if during irradiation things like performance of the U-Mo fuel particles, formation of interaction layers in the fuel meat, formation of defects (e.g., cracks or pores) in the fuel meat, or a combination of these factors are causing fuel failure and must be addressed. Furthermore, new destructive examination data from the historical plates can be made available in a relatively short timeframe so that new information can be assessed without having to wait for only the new destructive examination data that will result from the EMPIRE irradiation experiment. This presentation will discuss microstructural characterization information that has been generated recently that can be used to support the successful completion of the goals of the EMPIRE irradiation experiment, and ultimately the “Comprehension Phase” of the HERACLES international fuel development program.

## 6.10 Fabrication of Atomized LEU-7wt.%Mo Powder for KUCA Core Conversion

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The conversion program of the Kyoto University Critical Assembly (KUCA) from highly enriched uranium (HEU) to low enriched uranium (LEU) is being conducted by the collaboration with Korea, the US, France, and Japan. The US supplied LEU materials for the fabrication of fuel powder and Korea has fabricated the centrifugally atomized LEU-7wt.%Mo powder. A total of 13 batches were fabricated to meet the specification of maximum impurity and power size distribution with about 95% yield ratio. D10, D50 and D90 of the total powder are about 40 µm, 70 µm and 100 µm, respectively. The powder will be transported to CERCA, France using TN-BGC1 casks at the end of this October for manufacturing fuel coupons of KUCA. If this project is successfully completed, KUCA will be the first LEU conversion facility using the centrifugally atomized U-Mo powder.

## 6.11 Transition Cores Accident Analyses for the Conversion of the University of Missouri Research Reactor from Highly Enriched to Low-Enriched Uranium

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In support of transition from highly enriched uranium (HEU) to low-enriched uranium (LEU) fuels for the University of Missouri Research Reactor (MURR®), safety analyses are performed on transition cores with the LEU fuel elements. The initial LEU core upon conversion will be loaded with eight fresh fuel elements. In transition to an LEU core containing depleted fuel that is prototypic for MURR operations, it is necessary to have a sequence of transition cores wherein LEU fuel is brought into an equilibrium depletion state. For the sequence of transition cores, predictions for postulated reactivity insertion accidents (RIAs), loss of coolant accidents (LOCAs), and loss of flow accidents (LOFAs) have been conducted. Results show acceptable margins to the safety margins to the fuel temperature safety limit for all considered scenarios. Moreover, the margins are bounded by the preliminary equilibrium LEU core, which confirms the safety of the proposed fuel management during the initial cores after LEU startup.

## 6.12 UMo Benchmark Experiment Data Needed to Support Computational and Nuclear Data Validation

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Currently there are almost no benchmark experiment data available directly supporting neutronics modeling and simulation of ~20 wt.% $^{235}$U, UMo fuel contained within the Organisation for Co-operation and Development Nuclear Energy Agency (OECD NEA) international benchmark handbooks from the International Criticality Safety Benchmark Evaluation Project (ICSBEP) and International Reactor Physics Experiment Evaluation Project (IRPhEP). The benchmarks within these two handbooks represent the international standards used to support computational and nuclear data validation. Recent activities in nuclear data development supporting new neutronics thermal scattering cross sections for UMo fuel indicate the necessity to provide quality benchmark experiment data that can be utilized to support their validation and development. Preparation of high-integrity neutronics benchmarks from UMo experimental data will ensure proper integral testing of these cross sections, as well as serve to support computational validation of contemporary modeling and simulation software.

## 6.13 Scale-Up of Atomic Layer Deposition Coating and Heat Treatment of Uranium-Molybdenum Powder

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As part of the fabrication efforts for the EMPIrE irradiation experiment, a procedure for using Atomic Layer Deposition (ALD) to coat ZrN on U-Mo powders was developed, as well as a process for heat-treating U-Mo powders to increase the grain size of the powders. As developmental processes, the batch sizes for both ALD coating and heat treatment were limited to approximately 150g. However, if these processes are selected as part of the finalized fabrication process for LEU-Mo dispersion fuel, larger batch sizes will be needed for full-scale fabrication. This report details efforts to increase the batch size for each of these processes to the pilot scale, with a batch size of approximately 500g. Design modifications and preliminary results will be presented.

## 6.14 DART Simulation of Plate-Type Fuels to High Burnup

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The DART computational code is developed to simulate the in-pile irradiation behavior of research and test reactor (RTR) fuels. The code has been significantly upgraded and restructured in order to meet the modeling requirements of full-size fuel plates irradiated at high power to high burnup. These enhancements include a 2-D heat transfer module, updated interdiffusion layer growth correlations at high power for dispersion fuels, the parallelization scheme to facilitate microstructural modeling for full-size plates, and the addition of the monolithic-fuel branch. The code is able to simulate fuel plate swelling for both dispersion and monolithic types up to high burnup using either mechanistic or empirical approaches. Evolution of fission gas bubble morphology during irradiation are calculated through the rate-theory-based GRASS module. Validation and verification of the code is concurrently performed by comparing to experimental data and benchmarking with other codes. The DART code is applied to understand the governing physics of fuel swelling and its dependence upon fission density and rate and to provide information of how as-fabricated microstructures and degradation of key materials properties may affect fuel performance.

## 6.15 Microstructure-Based Process Modeling and Integration of U-10Mo

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Low-enriched uranium alloyed with 10 wt% molybdenum (U-10Mo) has been recognized as a promising candidate to replace highly enriched uranium fuel for the U.S. high-performance research reactors. Manufacturing the U-10Mo fuel involves a series of thermomechanical processing steps, which include casting, homogenization, hot rolling, annealing, cold rolling, and hot isostatic pressing (HIP). In order to develop processing windows and provide quick feedback to the fabricators, modeling methods have been developed for the individual processes. Modeling the interaction and coupling between individual processes using integrated computational materials engineering (ICME) aims to bridge the information between the interacting models so the effects of manufacturing processes on material characteristics can be better understood. Implementation of ICME leads to better understanding of the processing windows, helps meet the desired/specified form of the fuel, predicts the microstructure evolution and second-phase particle redistribution across multiple processes, and provides guidance for accelerated and more cost-effective fabrication development. This paper summarizes the individual and the integration of U-10Mo fabrication processes from homogenization to final annealing after cold rolling.

## 6.16 Evaluation of Fuel Swelling and Irradiation Creep Behavior for a MURR LEU U-10Mo Monolithic Plate: A Finite Element Analysis Based Study

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Finite element analysis (FEA) is used to model and simulate the thermo-mechanical behavior of a selected LEU MURR fuel plate under prototypic irradiation conditions of the proposed LEU MURR core. Fuel swelling was described considering three swelling correlations: an initial Kim correlation, a recent best-estimate swelling correlation, and the 95% upper confidence limit of a correlation including all available data, where the last one represents an extreme case (most conservative estimate of fuel swelling). Furthermore, irradiation creep strain rate in the U-10Mo fuel was modeled such that it linearly varies with equivalent stress, fission rate and empirical creep rate coefficient. Deformation profile and corresponding increase in the thickness of the simulated plate were determined considering various combinations of the aforementioned three swelling correlations and creep rate coefficients of 750, 500, 250, and 5 ($\times 10^{-25}$ $cm^3$/MPa-fission). Plate deformation at EOL was determined considering various combinations of the aforementioned swelling correlations and creep rate coefficient values. Results showed that the maximum change in plate thickness due to radiation effects is less than the allowances in the Preliminary Safety Analysis Report (PSAR) for a change in the plate thickness that contributes to a decrease in the coolant channel thickness.

## 6.17 Verification and Validation of Thermal-Hydraulic Analysis Software: The University of Missouri Research Reactor as a Case Study

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In support of the conversion of U.S. High Performance Research Reactors (USHPRR) from highly enriched uranium (HEU) to low-enriched uranium (LEU) fuels, various thermal-hydraulic analysis software is used to assess reactor designs and safety for both normal steady-state operations and postulated transient accident scenarios. To ensure the integrity of LEU reactor conversion designs, there are ongoing efforts for the verification and validation (V&V) of the analysis software according to the applicable sections of ASME NQA-1, “Quality Assurance Requirements for Nuclear Facility Applications.” This paper discusses the general V&V methodology within the Research and Test Reactor (RTR) Department at Argonne National Laboratory for three main thermal hydraulics software: PLTEMP/ANL for modeling steady state

operations, and PARET/ANL and RELAP5/MOD3.3 for modeling transient accident scenarios in the reactor core region and for system transients, respectively. Thermal-hydraulics analyses for the University of Missouri Research Reactor (MURR®) are used as a benchmark to demonstrate the performance of the qualified software for safety analyses.

# SESSION 7

## LEU Fuel Design and Qualification

Session Chair: Bruno Baumeister

### 7.1 Status of KJRR Fuel Qualification - Update

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As a part of the South Korean Ki-Jang Research Reactor (KJRR) fuel licensing and qualification effort, an irradiation and post-irradiation examination (PIE) test of the KJRR Lead Test Assembly (LTA) is conducting at U.S. Idaho National Laboratory (INL). Started from November 2015, the irradiation test was successfully completed at Advanced Test Reactor (ATR) of INL by reaching 216.6 EFPD and maximum 83% U-235 depletion at the end of February 2017. After 11 months of cooling period, both non-destructive and destructive Post-Irradiation Examination (PIE) has commenced at Material and Fuel Complex (MFC) of INL.

For the non-destructive PIE, the KJRR-LTA was first visually inspected and confirmed no significant defect was occurred during irradiation test. Neutron radiography imaging technology was used to review fuel plate channel gap distances. Profilometry for entire 21 fuel plates are completed to calculate oxide thickness. To support burn-up level calculation, gamma scanning for five selective plates are mostly completed as well. The result of the non-destructive PIE will be reported by end of October 2019 and followed by destructive PIE such as optical metallography, burn-up analysis and blistering endurance test.

In addition to the LTA irradiation qualification, the first mini-plate irradiation test (HAMP-1) was completed in HANARO at KAERI with max 65 % U-235 depletion. The PIE on HAMP-1 mini-plates revealed that there were not any abnormalities during the irradiation. The second and third test series (HAMP-2&3) will be also commenced in the nearest future. on-destructive PIE has been undergoing currently.

### 7.2 Review on the Development of Very High Density Fuels by CMAD Group

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Since 2002, the CMAD group of the CNEA has been working on different types of high density nuclear fuels, either in its dispersed form with cladding of aluminum and UMo core or in its monolithic form, employing different kind of alloys for the core and with zirconium alloy as cladding.

In this work, we take a review of the observed details focused in the development of monolithic fuels, the observations made in the interlayer between cladding and core at different temperatures, the variation of dog bone in the core, the dependence of these observations with work temperature and reductions made in the miniplates.

According with our observations we determined the best conditions to fabricate the miniplates and to avoid the layer interaction, the dog bond and the pillowing.

## 7.3 SEMPER FIDELIS: Post Irradiation Examination Results

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## 7.4 Non-Destructive Examination Preliminary Results of EMPIrE

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Non-destructive post irradiation examinations of the EMPIrE experiment have been completed at the hot fuels examination facility at the Idaho National Laboratory. These preliminary results including neutron radiography, gamma scanning, immersion density testing, and profilometry are summarized herein along with initial fuel performance analysis and destructive examination plans.

## 7.5 A Qualification Base Report on High-Density $U_3Si_2$/Al Dispersion Fuel for High-Power Research Reactors

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$U_3Si_2$ fuel dispersed in an Al matrix, $U_3Si_2$/Al, is the densest dispersion fuel qualified for use in research and test reactors (RTR). The safety evaluation report (SER) issued by the US NRC (NUREG-1313) licensed the fuel for use with densities up to 4.8 gU/cm$^3$, at the power level interrogated in the demonstration test. Since then, more test data, particularly at high powers, has become available, in an attempt to demonstrate the possibility of using this fuel under higher power conditions. A stand-alone data compilation that enables review and analysis of the available higher-power data in conjunction with the NUREG-1313 data is needed. This report is intended to serve as a base for a fuel qualification report in the future. Knowledge gaps for higher-power applications of the fuel have been identified, indicating the desired work areas. In this presentation, a status review of the fuel, performance issues related to the high power applications, and gap items will be discussed.

## 7.6 Status of LEU U-10Mo Monolithic Fuel Testing and Qualification Efforts to Support US High Performance Research Reactor Conversions

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The US High Performance Research Reactor Project has begun a new phase of fuel testing and post irradiation examination (PIE) of plate-type low-enriched uranium (LEU) U-10Mo Monolithic fuel. The

testing campaigns are intended to first confirm that the fuel performance behavior of commercially produced fuel matches that of the lab-scale fabricated fuel tested during development, followed by qualification of the U-10Mo monolithic fuel system through the US Nuclear Regulatory Commission. Planned tests include miniature-plate, full-size plate, and fuel element irradiations, during which fuel test specimens fabricated with qualified processes will be irradiated under conditions that bound the NRC-regulated reactors in order to demonstrate that the fuel meets the overarching fuel performance requirements of 1) maintaining mechanical integrity 2) maintaining geometric stability and 3) exhibiting stable and predictable behavior. The MP-1 test, with the objective of confirming fuel performance behavior, is currently undergoing irradiation in the Advanced Test Reactor and PIE in INL hot cells. Status of upcoming qualification and licensing test design activities and highlights of initial as-fabricated characterization and testing results from the MP-1 test will be provided.

# SESSION 8

## Fuel Fabrication Technology

Session Chair: Scott Ravenhill

### 8.1 USHPRR Fuel Fabrication Pillar Fabrication Status, Process Optimizations and Future Plans

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The Fuel Fabrication (FF) Pillar, a project within U.S. High Performance Research Reactor Conversion program of the office of NNSA's Material Management and Minimization (M3), is tasked with the scale-up and commercialization of the high-density monolithic U-10Mo and $U_3Si_2$ "silicide" fuel for the conversion of appropriate United States based research reactors. The FF Pillar has made significant fabrication demonstration progress in the form of the delivery of the MP-1 Medium Power and Low Power (LEU) irradiation experiment specimens. This was a successful demonstration of the Foil Fabrication Demonstration Line established at BWXT. Recent casting optimizations have been investigated and a process design has been selected for upcoming experiments, fabrication demonstrations, and long term production. Characterization from optimization studies, along with continued process modeling, has led to significant process improvements including developing methods for assessing process qualification to demonstrate and optimize the baseline co-rolling process using commercial-scale equipment.

### 8.2 Y-12 Past, Present and Future Supplying Uranium

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The Y-12 National Security Complex (Y-12) has a rich history rooted in the Manhattan Project and the early nuclear era which included enriching uranium using the Calutron process. After World War II, Y-12 focused its mission on the production of weapons components. As the cold war ended, Y-12 became heavily involved in the dismantlement of the secondary components. The uranium which became excess to national security needs was provided for supply. This highly enriched uranium (HEU) would also be down blended at Y-12 to produce high assay low enriched uranium (HALEU). These excess materials have been distributed around the world for research reactors. These collective skills have assisted the broader nonproliferation mission of converting research reactors from HEU to HALEU. In addition HEU from around the world has been repatriated to Y-12 for future peaceful uses such as the production of HALEU for research reactors and medical isotope production.

## 8.3 USHPRR Fuel Element Specifications and Plate Demonstration Plans

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The conversion of five US High Performance Research Reactors (USHPRRs) and one critical assembly from the use of highly enriched uranium (HEU) to low-enriched uranium (LEU) fuel is underway. As a part of the Reactor Conversion (RC) Pillar work, led by Argonne and the reactor organizations, LEU fuel element specifications and drawings for initial plate and element trials have been developed for all USHPRRs that will convert on high-density monolithic fuel made of alloy of uranium-10 wt% molybdenum (U-10Mo). These specifications and drawings and their future revisions will be used as a part of the future work to demonstrate and evolve the maturity of the fuel fabrication capability.

A campaign for plate (PD) and element fabrication demonstrations has been planned jointly by the RC Pillar and the Fuel Fabrication (FF) Pillar, led by Pacific Northwest National Laboratory. The PD campaigns will be the first opportunity for the commercial fabricator to make full-size plates and to exercise the fabrication process at a near-production scale. The fabrication experience gained during these campaigns will be used to improve the quality control measurement methods and equipment, as well as the fabrication processes, to prepare finalized fuel element specification requirements for each of the USHPRR safety and operational design bases.

## 8.4 Impact Assessment for the MIT Research Reactor LEU Fuel Fabrication

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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 HEU fuel element and maintain neutron flux performance while requiring minimal changes to the reactor structures and systems. The selected monolithic U-10Mo LEU fuel element design is a 19-plate unfinned fuel element with three different fuel foil thicknesses, denoted as “FYT” element design. The LEU fuel design has been shown to deliver 7 MW safely with increased primary flow to maintain the neutron flux performance of the 6 MW HEU core. Engineering drawings of LEU fuel elements are completed as an initial step to develop fuel specifications. Impact assessment for fuel fabrication tolerances on fuel safety and performance such as coolant channel gaps, U-235 enrichment and loading is currently ongoing. This study uses several computer codes including a neutronics code, MCNP5, and a thermal hydraulic code, STAT7, that have been used for previous analyses for MITR LEU fuel conversion. The criteria that are adopted for off-nominal geometry and composition evaluation include plate/element level power peaking, safety margin to onset of nucleate boiling, shutdown margin 1% $\Delta k/k$ for U-235 enrichment and loading variation at the full core level. Results of this study, and additional work, will inform stakeholders of the impact of fabrication tolerances of selected LEU fuel specifications on the MITR safety basis developed for conversion.

## 8.5 Manufacturing of the HiPROSIT Irradiation Experiment: High density $U_3Si_2$ Fuel Plates

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### 8.6 Future HALEU Supply – A Front-end Industrial Actor's View

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With a nearing end of downblended declassified HEU stockpile, and the prospect of HALEU needs for future fuel and reactor concepts, HALEU supply for research reactor fuel may present new challenges around the 2030s. Specific metal HALEU production facility may well be needed to ensure stable supply. As a front-end industrial actor with the technologies, expertise and industrial experience both in enrichment and in uranium metallization, Orano will present its view of the specific challenges for a future metal HALEU production, and what is necessary to be ready in time.

# SESSION 9

## High Performance Reactor Conversions

Session Chair: David Jaluvka

### 9.1 US High Performance Research Reactor LEU Conversion Design and Qualification Progress

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US reactors that regularly refuel with highly enriched uranium (HEU) have undertaken conversion to low-enriched uranium (LEU) using a high-density alloy of uranium-10 wt% molybdenum (U-10Mo) that is under qualification by the US High Performance Research Reactor (USHPRR) portion of the DOE/NNSA Material Management and Minimization (M3) Reactor Conversion program. The effort to convert the USHPRR is conducted by eleven US organizations, including the reactors, national laboratories, universities, a uranium processing plant, and a commercial fuel fabricator working to convert five reactors and an associated critical assembly. Since feasibility designs for conversion to LEU were completed in 2009, as of 2018 all five USHPRR have now completed LEU U-10Mo designs including three Preliminary Safety Analysis Reports (PSARs) for LEU conversion.

USHPRR regulated by the US Nuclear Regulatory Commission at the University of Missouri (MURR), Massachusetts Institute of Technology (MITR), and the National Institute of Standards and Technology (NBSR) have each completed a distinct PSAR. As a part of Reactor Conversion (RC) Pillar work, led by Argonne and these reactor organizations, the final two PSARs were submitted to the NRC by MURR and MITR in 2017. In 2018, USHPRR reactors regulated by the US Department of Energy (DOE) at Idaho National Laboratory (ATR) and Oak Ridge National Laboratory (HFIR) also completed optimized designs, with ATR confirming a 'base fuel' design of the same fabrication complexity as required for MURR, MITR and NBSR. Along with the previous complex U-10Mo designs, in 2018 HFIR also evaluated an option to convert on available $U_3Si_2$-Al dispersion fuel at a 'standard' 4.8 gU/cm$^3$ density, and is studying whether an increased fuel density reduces fabrication complexity. The Conversion Program is now pursuing plans

for HFIR to convert on silicide fuel. This requires irradiation testing due to plates containing boron, and since HFIR exceeds existing silicide qualification limits.

USHPRR design and safety bases are discussed along with the steps for future work to demonstrate fabrication at a commercial scale. A campaign for plate and element fabrication demonstrations has been planned jointly by the RC Pillar and the Fuel Fabrication Pillar, led by Pacific Northwest National Laboratory and is now underway. Using reactor-specific prototypic design parameter data provided by the RC Pillar, plate- and element-level irradiation testing, specific to each reactor, is being planned by the Fuel Qualification Pillar. Full-element flow testing, where required by design changes, will then be performed in order to finalize the SARs required before conversion to LEU fuel.

## 9.2 Progress Update on the MIT Research Reactor (MITR) Conversion from Highly Enriched Uranium to Low Enriched Uranium Fuel

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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. It delivers a neutron flux comparable to light water reactors in the compact core, and has a demonstrated track record in performing advanced materials, fuel, and instrumentation irradiation tests in light water reactors or high temperature reactors conditions. The conversion objective is to design a low-enriched uranium (LEU) fuel element that could safely replace the current 15-plate HEU fuel element and maintain performance while requiring minimal changes to the reactor structures and systems. The selected monolithic U-10Mo LEU fuel design is a 19-plate unfinned fuel element with three different fuel foil thicknesses, denoted as “FYT” element design. The LEU fuel design has been shown to deliver 7 MW safely to maintain the neutron flux performance of the 6 MW HEU core. The preliminary safety analysis report has been submitted to the U.S. NRC. The transition core analysis, from 22 fresh LEU elements gradually to 24 elements equilibrium core configuration, evaluated a fixed pattern refueling scheme where three fresh LEU fuel elements and three end-of-life elements are regularly introduced and discharged after each 10-week cycle. Results show that an equilibrium core is achieved after seven fuel cycles. Neutronic and thermal hydraulic modeling results demonstrated an adequate margin below the proposed fission density limits, a significant safety margin to onset of nucleate boiling, and all other core parameters can be maintained within the safety envelope as well. Engineering drawings of LEU fuel elements are completed as an initial step to develop fuel specifications. The LEU core startup plan is near completion. Impact assessment for fuel fabrication tolerances such as including coolant channel gaps, U-235 enrichment and loading is currently ongoing.

## 9.3 Conversion Status of the University of Missouri-Columbia Research Reactor from Highly Enriched to Low-Enriched Uranium Fuel

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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 U.S. National Nuclear Security Administration (NNSA) Office of Material Management and Minimization (M3) to find a suitable low-enriched uranium (LEU) fuel replacement for the currently required highly enriched uranium (HEU) fuel. In August 2017, a Preliminary Safety Analysis Report (PSAR) for a proposed core loaded with U-10Mo monolithic LEU fuel, which is currently being tested for qualification and licensing, was submitted to the U.S. Nuclear Regulatory Commission (NRC) for review. The PSAR includes detailed analyses of steady-state and accident/transient conditions that demonstrate, with the proposed fuel form and an uprate in power from 10 to 12 MW, sufficient margins to safety and operational performance. More recently, detailed analyses were conducted to plan fuel management, calculate safety margins, and estimate operational and experimental performance during the LEU transition core cycles without the use of poisoned LEU fuel elements, which are similar to the current HEU fuel. Earlier work anticipated that a burnable poison would be needed in up to twelve (12) LEU fuel elements in order to provide some reactivity hold-down during the initial, fresh LEU core startup cycles to limit perturbations to the axial flux profile in the flux trap and graphite reflector regions that would negatively impact the reactor's mission. Also presented is a discussion of the analyses performed for the storage of LEU fuel elements in the existing HEU storage locations. The analyses include: criticality safety; structural/mechanical loading, due to the increased weight of the LEU fuel element; storage planning, including mixed HEU/LEU fuel; dose rates from stored fuel elements; and decay heat removal during storage.

## 9.4 Recent Work on Conversion of the NIST Research Reactor

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The NIST Center for Neutron Research (NCNR) and Brookhaven National Laboratory (BNL) continue to work toward the conversion of the NCNR test reactor (aka the NBSR). The conversion to monolithic U-10Mo low-enriched uranium (LEU) fuel is supported by the National Nuclear Security Administration. At NCNR the focus has been on design and construction of a liquid deuterium cold source to replace the current liquid hydrogen cold source in order to mitigate experiment flux losses due to conversion. Several components for the new source have been received and tested including a new 7 kW refrigeration unit. At BNL the emphasis has been on refining the fuel specifications and operational parameters that feed into the fuel qualification program. A more detailed neutronics model has been introduced to reduce uncertainty in these parameters. Fuel qualification is also important because of the effort at NCNR to design a new neutron source to replace the NBSR and that design would also incorporate U-10Mo LEU fuel.

## 9.5 Present Status of the ATR LEU Fuel Project and the Element Test Campaign

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Progress continues in the ATR LEU Fuel Project, with the final design of the new ATR low-enriched uranium fuel element (referred to as the LOWE design). The Lead Test Elements will be inserted into driver positions in the ATR core during the Element Test Campaign (ET-1 through ET-3), with the number of elements placed in-core increasing with each ET irradiation campaign. The data gathered in the ATR LEU Fuel Project supports the Base Fuel Qualification for other U.S. High Power Research Reactor Program facilities, MITR, MURR, and NBSR. Work also progresses on the necessary addenda for the ATR and ATRC Safety Analysis Reports to support obtaining regulatory approval of the LOWE element insertions, and the eventual transitions to LEU cores in both ATR and ATRC. This presentation provides an overview of the recent progress, the technical challenges, and upcoming work.

### 9.6 High Flux Isotope Reactor Conversion from High-Enriched to Low-Enriched Uranium Fuel – A 2019 Progress Update

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In collaboration with the US reactor conversion program, Oak Ridge National Laboratory (ORNL) has been performing engineering evaluations on the conversion of its High Flux Isotope Reactor (HFIR) from high-enriched uranium (HEU) to low-enriched uranium (LEU) fuel since 2005. Design studies have been performed for HFIR cores loaded with LEU-10Mo monolithic and LEU$_3$Si$_2$-Al dispersion (4.8 and 5.3 gU/cm$^3$) fuels. Neutronic and thermal-hydraulic studies with the Shift and HFIR Steady State Heat Transfer Code tools, respectively, indicate that HFIR can preserve its HEU-like performance with the aforementioned fuel systems while maintaining adequate thermal safety margins if, among other considerations, fabrication of the fuel design features is demonstrated and qualification of the fuel is complete under HFIR-specific conditions. Recent advances have also been made in design optimization, high-fidelity COMSOL Multiphysics modeling, and RELAP transient analysis methods. ORNL continues to coordinate with the conversion program pursuing $U_3Si_2$ to ensure that a LEU-fueled core would result in safe, efficient, reliable, and cost-effective operation of HFIR to support its neutron science missions. Recent progress and plans regarding HFIR conversion studies will be discussed.

# SESSION 10

## International Conversion Progress, LEU Reactor Development and Post - Conversion Performance

Session Chair: Tom Hanlon

### 10.1 History and Current Status of the KUCA Dry Core Conversion Project

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The Kyoto University Critical Assembly (KUCA) is a multi-core type, thermal spectrum critical assembly consisting of one light-water moderated core and two solid-moderated cores, each fueled with highly enriched uranium (HEU). Starting several years ago, a program sponsored by the DOE/NNSA Office of Material Management and Minimization evaluated the feasibility of converting the solid-moderated (dry) cores to use low enriched uranium (LEU) fuel and started an ongoing research program to develop a suitable LEU fuel for the conversion. Initial feasibility studies indicated that U-7Mo dispersion-type material, compacted into a core then encapsulated to produce a fuel coupon would be a suitable LEU fuel. Using the research capabilities and expertise of Framatome-CERCA™, various cladding approaches were tested to determine the best approach to produce the LEU fuel coupons. As the coupon design evolved, complementary analysis and qualification of the LEU coupons continued. This paper presents a concise history of the conversion project, including the initial feasibility analysis, the research and development that led to the LEU fuel coupon design, a summary of the latest analysis results and the status of the KUCA dry cores conversion project.

## 10.2 Status of the IVG.1M Fuel Test

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The IVG.1M research reactor is operated by the National Nuclear Center of Republic of Kazakhstan. The reactor was designed to work with high-enriched uranium (HEU) fuel, however, in accordance with the agreement on the non-proliferation of nuclear weapons, HEU fuel should be replaced with a low-enriched uranium (LEU) fuel.

To date, two experimental channels with LEU fuel have been loaded into the IVG.1M reactor for carrying out complex tests. On August 2019 IAE NNC completed the 39$^{th}$ startup totaling 771.4 MW-h reactor energy, which is equivalent to 71% of the test plan; there continues to be no indication of fuel leakage. An important role in the faster completion of the fuel testing is played by the chiller installation. Its commissioning and successful start took place on June 26. Thus, we expect the remaining 29% of LEU irradiation testing to occur in the next few months. After that the two LEU fuel assemblies will be removed from the reactor for Post Irradiation Examination.

The aim of irradiation of the pilot WCTC-LEU is to reach same burn-up as in WCTC-HEU #4, which was extracted from IVG.1M reactor for examination in 2004. At the moment of carrying out of the given examination, the channel #4 had generated energy equal to 35 MW-h with thermalfluence 1018 n/cm$^2$ and burn-up 1.82 g of U-235. These parameters are reached after 1080 MW-h of operation of the reactor with the two LEU fuel assemblies inserted into the core.

## 10.3 Experimental and Analytical Transient Studies of Material Movements Inside Critical Configurations Using Low Enriched Uranium Fuel

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Experimental and analytical transient studies of the water moderator and the control rods movements inside the GIACINT critical configurations have determined the change of the neutron flux during the transient events. In these critical configurations, the fuel material is low enriched uranium zirconium carbon nitride ($U_{0.9}$-$Zr_{0.1}$-$C0_{.5}$-$N_{0.5}$) with ~12.5 g/cm$^3$ density and ~11.3 g/cm$^3$ uranium density. Water is the moderator and the reflector material, and the clad material is steel or niobium alloy. The fuel arrangement has a triangular lattice. The control rods material is born carbide with fuel rod follower or Plexiglas to avoid any disturbance for the neutron flux field. Two types of transients were studied. In the first transient experimental set, the water moderator level inside the critical configuration was changed with different speed. In the second set, the control rods were inserted with different speed. The neutron flux was measured in all the experiments as a function of the time starting from the critical condition. MCNP and SERPENT Analytical simulations were performed to calculate the neutron flux during these transients. The MCNP neutron tracking process was updated to simulate the geometrical model changes during the transient. In this paper, the experimental and the analytical results are presented and compared successfully.

## 10.4 Non-Reactor Tests of HEU and LEU Fuel and Reflector Material (Beryllium Oxide) as Part of the IGR Reactor Conversion

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In the framework of contract between Argonne National Laboratory and NNC, non-reactor tests of high and low enriched uranium (HEU, LEU) fuel and reflector material for the pulse-graphite reactor (IGR) were carried out at the IAE.

Graphite with impregnated uranium 90 % enriched in U-235 is used as HEU fuel for the IGR research reactor. The proposed LEU fuel is enriched to 20% U-235. To transfer the reactor to LEU fuel, in order to preserve its unique characteristics of neutron fluence, the use of beryllium-oxide (BeO) reflector in the central channel was assumed. The following main tasks were identified to assess the feasibility of the IGR conversion, to develop technical requirements for LEU fuel and to obtain data for safety analysis:

1. To conduct thermal cyclic tests of fuel and reflector samples;
2. To investigate the change in compressive strength due to thermal loading of HEU fuel, LEU fuel and beryllium oxide;
3. To study the process of high-temperature interaction of BeO with graphite; and
4. To measure the thermophysical properties of fuel and reflector materials over wide range of temperature.

The results of these investigations will be presented.

## 10.5 Progress of the Kijang Research Reactor (KJRR) Project in Korea

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KJRR is a new research reactor under construction for radioisotope production and neutron transmutation doping (NTD) of silicon in Korea. The reactor uses low enriched and high density alloy of uranium-molybdenum. This U-Mo plate type fuel is the first in a full-scale reactor in the world. The uranium density of the fuel is 8g-U/cc for standard fuels. For the qualification, the lead test assembly (LTA) had been irradiated at Advanced Test Reactor (ATR) of Idaho National Laboratory (INL). It is undergoing inspection and testing at INL. The other fuel samples for the evaluation of KJRR fuel performance will be irradiated at HANARO in Korea.

The construction permit (CP) of KJRR was issued on May 10, 2019. Application for CP was filed in November, 2014. During the review by regulatory body, two large earthquakes were occurred near the KJRR site. Additional time was needed to evaluate the characteristics of the seismic events from the geological point of view. In result, the project was in hibernation until the evaluation completed.

After CP approval, the project team was reorganized. Some long lead equipment will be procuring in 2019 to maintain the new milestone schedule. Construction work is scheduled to begin by the end of 2020. And HANARO reactor will return to routine operating cycle soon after some regulatory issues are resolved. The FSAR will be submitted to regulatory body by the end of 2022 to apply for an operation license (OL).

# SESSION 11

## Fuel Irradiation Testing and Characterization

Session Chair: Jong Man Park

### 11.1 High-Temperature FLiBe Salt and Materials Irradiation Tests Supporting Molten Salt Reactors Development at the MIT Research Reactor

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The Massachusetts Institute of Technology Reactor (MITR) is a 6 MW university research reactor that is part of the Nuclear Reactor Laboratory (NRL). It is well-suited for carrying out both basic and integrated studies because of its relatively high-power density (similar to that of a LWR), its easy-access geometric configuration, and the proven capability of the MITR staff to effectively design and execute proof-of-concept experiments. The MITR became the first partner facility of Department of Energy's Nuclear Science User Facilities (NSUF) in 2008, and has been utilized by a wide user base from universities, national labs and the nuclear industry. One of the recent accomplishments is the successful completion of four fluoride salt (FLiBe) irradiation experiments at 650-700°C to support the development of a new Molten Salt Reactor (MSR) concept, Fluoride salt-cooled High-temperature Reactor (FHR), with funding from two U.S. Department of Energy Integrated Research Projects, led by MIT with collaborators including the University of California-Berkeley (UCB), and the University of Wisconsin-Madison (UW). The objectives of these irradiation experiments are: (1) to assess the corrosion and compatibility of proposed FHR materials 316 stainless steel, Hastelloy® N, SiC and SiCf/SiC composites, nuclear graphites, Cf/C composite, and surrogate TRISO fuel particles in molten FLiBe, (2) to measure the fast neutron activation products 16N ($t_{1/2}$ = 7.1 s,) and 19O ($t_{1/2}$ = 26.9 s,) that are significant radiation dose contribution in the gas phase, (3) to examine the partitioning of tritium, produced from neutron interactions with FLiBe, among the various media in the experiment, and (4) to test tritium permeation coating materials for tritium control and management. New irradiation facilities were specifically designed for three in-core irradiation experiments and one 3-in. vertical neutron beam port. In addition, a dedicated lab space is set up for handling the irradiated FLiBe. The MITR is currently the only facility that has the demonstrated capabilities and infrastructure to perform high-temperature in-core irradiations of FLiBe salt to support MSR and FHR development in the US.

### 11.2 Low Enrichment Nuclear Fuel Based on Uranium-Zirconium Carbonitride: Performance of the Methodical Reactor Experiment

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Uranium-zirconium carbonitride UZrCN has been developed at the FSUE SRI SIA "LUCH" and is a high-density high-temperature fuel with high heat conductivity capable of being used in various types of reactors, comprising research reactors, including of conversion HEU on LEU fuel. The main problem hindering wide application of this fuel is insufficient knowledge of its behavior under irradiation, especially at high burnup. To eliminate this drawback preparations are currently being done to conduct a long-term reactor experiment whose goal is to study the characteristics of UZrCN fuel at reaching a burnup of ≈ 40% fifa in the SM-3 research reactor at the JSC "SSC RIAR". The parameters of the reactor experiment are as follows: temperature at the claddings of the experimental capsule – not exceeding 1000 K, power density in UZrCN tablets – up to 550 W/cm$^3$, duration of the reactor experiment – 5 years. In the process of preparation for the long-term reactor experiment the principal diagram of the irradiating device with the experimental capsule equipped with UZrCN tablets enriched to 19.75% by U-235 was developed and a number of auxiliary tasks were completed in order to contribute to achievement of the goal specified, neutronic and thermophysical analysis was performed and the pre-pile experiments were conducted, namely, the effective heat conductivity factors of the contact spots of the irradiating device elements were experimentally determined, thermal cycling tests of the irradiating device cladding were performed, the interaction between UZrCN fuel and the cladding material was demonstrated and experimentally confirmed. The analysis of the neutronic parameters of the experiment was conducted using MCU-RR code. The code uses the algorithm of solution the neutron transport equation by Monte Carlo method. The goal of the neutronic analysis was to determine the level of power density in the fuel pellets and structural elements of the irradiating device at the beginning and the end of the campaign. Basing on the data of the neutronic analysis thermophysical and mechanical analyses were conducted. Thermophysical analysis was carried out using ANSYS and PARAM-TG codes. The analysis was carried out iteratively, taking into account the changes of the model's geometry and dimensions resulting from thermal expansion, as well as the temperature dependence of thermophysical properties of solids, liquids and gases. Modeling of the temperature conditions was carried out taking into account the changeability of the geometry and dimensions of UZrCN fuel and the fission gas release. The analysis was conducted for two states of the core corresponding to the beginning and the end of the campaign to achieve a burnup of 40 % fifa.

The results of the works performed made it possible in July 2019 to conduct the methodical experiment in the SM-3 reactor in order to substantiate the developed irradiating device under the specified experimental parameters. Testing of the irradiating device was conducted in a reflector cell of the SM-3 reactor. The device was being irradiated for 23.3 effective days. Mean value of the power density in the pellets tested throughout the duration of the methodical reactor testing equaled 516 W/cm$^3$. Burnup achieved in the pellets tested – 0.63% fifa. The experimental capsule currently undergoes the post-irradiation examination.

## 11.3 Current Plans for Irradiation Testing of Low Enriched Uranium Silicide Fuel in Support of High Flux Isotope Reactor (HFIR) Conversion

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Faced with challenges in fabrication of contoured Low Enriched Uranium (LEU) U-10Mo monolithic fuel with burnable absorbers for conversion of High Flux Isotope Reactor (HFIR), United States High Performance Research Reactor (USHPRR) Program turned to potential alternative solutions. Taking into account decision by some of the European partners to qualify LEU uranium silicide fuel with high U loading for use in European High Performance Research Reactors, and noting additional potential benefits of silicide, USHPRR program conducted initial studies to assess feasibility of conversion of HFIR core to LEU silicide. Given positive outcomes of the study, the decision was made to develop a strategy for qualification of silicide fuel for use in HFIR conversion. This strategy includes several irradiation tests that will be discussed herein.

## 11.4 Overview and Status of the FUTURE-HFIR Irradiation Experiment

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The U.S. Department of Energy National Nuclear Security Administration Office of Material Management and Minimization US High Performance Research Reactor (USHPRR) Low-Enrichment Uranium (LEU) Conversion program has been provided an opportunity to participate in the ongoing COBRA irradiations taking place at SCK•CEN, in the BR2 reactor. The timing of the irradiation allows for an assessment of $U_3Si_2$ dispersion, plate-type fuel intended to be used in the LEU conversion of the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory. Additionally, U-10Mo plates will be evaluated as a fallback option for HFIR, to further elucidate monolithic fuel's performance at high power, and align the FUTURE-HFIR effort with USHPRR experiments performed at Idaho National Laboratory's Advanced Test Reactor. This presentation will provide an overview of the FUTURE-HFIR experiment and its design, and provide a current status of activities, requirements, logistics, and the predicted performance of the various plate types relative to their irradiation targets.

## 11.5 Design of Full Size Plate Irradiation Test for U-10Mo Monolithic Fuel Qualification

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Qualification of U-10Mo monolithic fuel is phased in three stages: (i) mini-plate (MP) irradiations, (ii) large or full size plate (FSP) tests, and (iii) element tests (ET). The FSP-1 irradiation design was recently completed. The experiment includes fuel conditions representative of U.S. High Performance Research Reactors (HPRRs) research reactors using three plate types. FSP-1 will show the correlation of fuel behavior between a large U-10Mo monolithic fuel plates and a small, mini-plate sized fuel plates. It is designed to reach the highest power and fission densities achieved in LEU core designs for HPRRs regulated by the U.S. Nuclear Regulatory Commission. Six plates, two of each plate type, will be irradiated in the north east flux trap of ATR for up to five cycles. FSP-1 design has been completed and irradiations are scheduled to begin in late 2021.

## 11.6 A Transmission Electron Microscopy Study of Low Burnup U-7Mo Samples

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Transmission electron microscopy of irradiated U(Mo) revealed that the intragranular fission gas bubbles self-organize into an ordered lattice. Samples from intermediate to high burnups have been widely analyzed in the past showing the morphology of the ordered structure as a function of fission density. However, in these specimens the nanobubble lattice was typically pictured in a later stage of its life, at a time that ordering was already fully completed. In this study, a transmission electron microscopy characterization of very low fission density UMo samples was performed to investigate the first stages of ordering. It was found that: i) the bubble lattice gets formed at fission densities smaller than $0.7 \times 10^{21}$ fissions/cc; ii) nanobubbles starts to align in proximity of grain boundaries; iii) the lattice geometrical characteristics are constant with burnup; iv) bubbles are extremely underpressurized at low burnup. The presented results provide further insights into the development of the xenon bubble lattice in irradiated U(Mo).

# SESSION 12

## Fuel Performance Measurement Analysis and Modeling

Session Chair: Adam Robinson

### 12.1 Projection of Irradiation Behavior of $U_3Si_2$ Dispersion Fuel at High Fuel Loading and High Operating Power

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In view of the recently renewed interest in the use of low enriched silicide dispersion fuel for the conversion of Research Reactors that operate at the high end of the performance range, an assessment of the projected irradiation behavior under the more taxing conditions is required.

To this end the PIE data accumulated during the $U_3Si_2$ irradiation programs in the ORR were reexamined as well as a few more recent results of higher power experiments performed in BR-2 and ATR.

The data show that at LEU burnup of 85% or higher, coarsening of fission gas bubbles occurs in all experiments examined. However, the porosity does not form interconnected gas bubbles and appears to be of acceptable magnitude.

The operating parameters of 4-5 g $cm^{-3}$ $U_3Si_2$ experiments irradiated in ORR, BR2 and ATR ranged at a fuel temperature of 100°C - 165°C and a life-average fission rate of 2 - 7 x $10^{14}$ $cm^{-3}$ $s^{-1}$. The fuel swelling and the fuel-aluminum interaction appear not to be significantly affected by this parameter range.

From the results of tests performed in HFIR, as part of the ANS project, it appears that at fuel temperatures of 200°C and higher the behavior of $U_3Si_2$-Al dispersion fuel is clearly different. The fuel particles do not transform to an amorphous structure but remain crystalline – having its different fission gas behavior. The $U_3Si_2$-Al interdiffusion, being temperature dependent, is quite extensive at these higher temperatures.

If operation at 200°C and higher is contemplated, an accurate and more extensive temperature evaluation is recommended.

### 12.2 Evaluation of the Thickness of PVD-Deposited Mo, Zr Coatings as Diffusion Barriers Between U-Mo and Al Using Heavy Ion Irradiation

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Diffusional interactions between U-Mo and Al significantly degenerate the performance of U-Mo fuels. Using a coating barrier might be an efficient protection against the undesired interdiffusion. Transition metals Zr and Mo have been proven to be promising candidates of diffusion barriers during the previous diffusion couple tests, ion irradiation tests and also in-pile irradiation tests. The focus of this work lies on a systematic evaluation of the coating thickness, which presents a notable cost factor in the industrial fuel production. U-Mo/Zr/Al and U-Mo/Mo/Al trilayers with different coating thicknesses (50 nm, 150 nm, 500 nm and 2 µm for Mo; 200 nm and 5 µm for Zr) were prepared using the PVD technique and then tested under irradiations with 80 MeV I-127 ions at 140°C performed at the MLL tandem accelerator. To study the interdiffusion behaviour at the interface between Al and the coating, we additionally prepared a

Mo/Al bilayer and irradiated it under the same conditions. A maximum burn-up equivalent of ~ $1\times10^{21}$ fissions/cm$^3$ has been reached during each irradiation.

Atomic mixing at interfaces occurred during the ion irradiations. The interdiffusion microstructures and their chemical composition were examined by scanning electron microscopy (SEM) and energy-dispersive X-ray spectroscopy (EDX). Irradiation performance of the fuel samples is discussed in terms of the coating thickness and the irradiation dose. This paper provides a first overview of evaluation and eventually selection of the barrier thickness in U-Mo/Al based fuels via heavy ion irradiation.

## 12.3 Fission Product Release Testing to Support Qualification of an LEU Fuel for the Advanced Test Reactor

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To support the qualification of a low-enriched uranium (LEU) fuel that can meet the mission of Advanced Test Reactor (ATR), fission product release rate data on the LEU fuel (U-10Mo) must be obtained to update the ATR maximum hypothetical accident (MHA) evaluation. The MHA in the ATR safety analysis report (SAR) is currently supported by a detailed severe accident analysis (SAA) that was used to estimate the offsite and onsite consequences of the MHA. In order to support the ATR LEU SAR, this accident analysis must be updated to incorporate models that can predict the degradation and fission product release behavior of the new LEU fuel (U-10Mo) so that offsite and onsite source terms can be estimated. At present, there are limited data available on the behavior of U-10Mo under degraded conditions, particularly on fission product release rates, which are a necessary piece of information to perform a comparable severe accident analysis to what was performed with the existing fuel system. This presentation discusses the framework that will be used to evaluate the MHA for ATR LEU fuel and summarizes an evaluation of options that was performed to determine the best approach for obtaining fission product release data for U-10Mo fuel.

## 12.4 Operational Effects of the BR2 Driver Fuel Transition

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Between August 2017 and December 2018, the Belgian Reactor 2 (BR2) undertook a measurement campaign to validate the COBRA HEU fuel elements. These fuel elements represent an intermediate step in the BR2's mission to convert from HEU to LEU fuel elements, whereby the neutron poison was changed. This measurement campaign was successfully completed at the end of 2018, with the first shipment of validated COBRA fuel elements arriving in December 2018. In January 2019, the BR2 began to transition to COBRA elements as the primary source of reactor fuel.

As more COBRA elements were introduced into the core, variations from historical operating characteristics were noticed. First, the gadolinium (Gd) content in a fresh COBRA element yielded a greater negative reactivity than the previous boron and samarium (B&Sm) doped fuel elements. This meant fresh fuel elements needed to be placed more towards the periphery of the core to ensure required control rod critical height at beginning of cycle. Second, the Gd burned at a higher rate than the B&Sm; thereby offsetting the xenon (Xe) transient. This meant that the evolution of the control rods during the cycle became more linear versus the previous crested curve.

### 12.5 Attaining Uniform Thickness of U-10Mo and Zr in Monolithic Fuel

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Low-enriched uranium with 10 wt.% molybdenum (U-10Mo) is a promising candidate to replace high-enriched uranium fuel due to its ability to meet the neutron flux demands of U.S. high power research reactors. During hot rolling, the roll pack is consisting of U-10Mo, a Zr (Zirconium) interlayer, and a steel can. Experimental observations revealed a thinning of the Zr layer during this fuel fabrication process, which is not desirable from the fuel performance perspective. Coarse U-10Mo grains and non-homogenized molybdenum in the uranium matrix can lead to a nonuniform U-10Mo/Zr interface. The purpose of this work is to investigate the effects of these microstructural parameters on the Zr coating variation. A microstructure-based finite-element model was developed, and a study on the effect of grain size, can material and geometry, and homogenization on the Zr/U-10Mo interface was conducted. The model successfully predicted the experimentally observed thinning of the Zr layer.

### 12.6 Review of the Technical Basis for Properties and Fuel Performance Data Used in USHPRR HEU to LEU Conversion Analysis Compared to the Preliminary UMo Report

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Ongoing efforts to convert research and test reactors (RTR) to operate with low-enriched uranium (LEU) rather than high-enriched uranium (HEU) fuel requires analysis for each reactor to be converted. This work presents the technical basis for several key material properties for uranium-10 wt.% molybdenum (U-10Mo) that is under qualification for conversion of US High Performance Research Reactors (USHPRR) within the DOE/NNSA Material Management and Minimization (M3) Reactor Conversion program, as well as a discussion of fuel performance parameters (swelling and blister temperature). Collection and assessment of the data available in literature was conducted, the results of which were compared to the current values utilized for analysis in each of the USHPRR and to Revision 1 of the Preliminary Report on U-Mo Monolithic Fuel for Research Reactors (INL/EXT-17-40975). There is general agreement amongst the values used in the USHPRR analysis, the Preliminary UMo Report, and literature, although key data gaps were identified. Of particular importance is the collection of density and thermal conductivity data from prototypic samples over the temperature and burnup ranges relevant to the USHPRR.

# SESSION 13

## Conversion and Systems Analyses

Session Chair: Ann Leenaers

### 13.1 Non-parametric Statistical Safety Analysis Tools to Support ATR Conversion to LEU

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To better support the Advanced Test Reactor (ATR) Low-enriched Uranium (LEU) conversion, a new ATR design basis accident safety analysis approach has been proposed which utilizes the Best-Estimate Plus Uncertainty (BEPU) code SASQUATCH. SASQUATCH employs a non-parametric statistical method which allows analysts to calculate a pre-defined number of runs required to reach a specified probability and confidence threshold for a distributionless output, and grants the ability to control the sampling error

using order statistics. This presentation discusses the modernization of the ATR design-basis tools and evaluates the efficacy of the non-parametric statistical approach proposed for the LEU safety basis relative to the current parametric approach outlined in the ATR Safety Analysis Report (SAR).

## 13.2 Safety Analysis of the IVG.1M Reactor with LEU Fuel

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The IVG.1M research reactor is located in Kurchatov, Kazakhstan, operating with high-enriched uranium (HEU) fuel since 1971. In preparation for future conversion of IVG.1M to low-enriched uranium (LEU) fuel, a revised safety analysis report is being prepared. IAE has developed a new model for transient analysis. The model has been applied to analyze both reactivity and coolant flow initiated transients under assumption that emergency protection system operates. ANL has performed analysis in parallel using a different model than used by IAE. The results from the two modeling efforts are similar and show that the maximum cladding and fuel temperatures are well below their melting temperatures.

## 13.3 Computational Codes Developed by RERTR for LEU Conversion and Used at Research and Test Reactors Worldwide

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As a part of the conversion of reactors since the inception of the Reduced Enrichment for Research and Test Reactors (RERTR) in 1978, a suite of computational software codes has been developed and deployed for design of the LEU fuel required for conversions worldwide. Today, a variety of commercial or general-use codes are also used today during reactor conversions from highly enriched uranium (HEU) to low-enriched uranium (LEU). There remain significant advantages to continued use of the software codes developed as a part of RERTR since they have been custom-developed for the wide spectrum of research and test reactor designs including both plate-type and pin-type reactors of various complex geometries. Many of these software codes are in use at research and test reactors after conversion for ongoing fuel calculations, safety analysis, and as a part of the licensing basis for reactors that have been converted.

The analysis domain for the specific software packages that are available includes primarily neutronics (steady-state, fuel depletion and transient), and thermal-hydraulics (steady-state and transient). Reactor safety determinations are made for key safety parameters, such as margins to onset of nucleate boiling (ONB), to onset of flow instability (OFI), and to critical heat flux (CHF). Postulated accident scenarios are analyzed in order to assure that the reactor does not exceed any safety limit at any time in each scenario. Under continued work on verification and validation, many of the analysis software packages used comply with ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," with a focus on supporting the needs of nuclear fuel element design and licensing.

## 13.4 Overview of Beyond Design Studies Performed within the Framework of BR2 Decennial Safety Review

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An extensive set of beyond design studies was performed by BR2 and ANL as one of the final remaining parts of BR2's 2016 decennial safety review. A general overview is provided of the scope of these beyond design studies as pre-defined at the request of the regulator, and the approach followed for the analyzed scenarios is discussed. Scenarios cover, among others, the long term cooling impact of an ordinary loss of forced cooling (part of BR2's design base), malfunction of the bypass valve that allows natural circulation cooling of the primary circuit and loss of heat sink (pool water level drop) in natural circulation cooling mode, even up to the level of air cooling. Furthermore, analyses were made of several primary LOCA scenarios, spent fuel pool cooling behavior under accidental conditions, performance of BR2's side and main pool syphon breakers, while a qualitative review was performed of the risks associated with fuel channel blockage. A summary for one or more of the key results is highlighted to illustrate the work that has been performed. It remains to be assessed to what extent conversion to LEU may impact the outcome of the different scenarios.

## 13.5 Four Design Demonstration Elements (DDEs) to be Irradiated in the Advanced Test Reactor (ATR) and Belgian Reactor 2 (BR2)

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The U.S. High Performance Research Reactor Project Fuel Qualification Pillar is focused on developing and qualifying low enriched uranium (LEU) plate type uranium molybdenum (U-Mo) monolithic and uranium silicide (U3Si4) dispersion fuel for conversion of US high power research reactors. Several irradiations of mini-plates and full-size plate irradiations will demonstrate the fuel's behavior. Along with qualifying fuel, larger tests are planned to support licensing fuel elements. These tests are referred to as Design Demonstration Elements (DDEs).

As a result of a scoping study performed in 2018, a decision was made to irradiate two DDEs in ATR at the Idaho National Laboratory (INL) and two other DDEs in BR2 in Belgium. The objective is to demonstrate successful irradiation of each DDE to support the licensing submittal of that specific reactor.

Neutronics analysis is being performed to match the power density distribution of the represented elements. Test train hardware design are being done to ensure all the target conditions are being met for the DDEs to be irradiated.

## 13.6 Involute Working Group – Progress towards Validation of CFD for Involute-Plate Reactors Safety Analysis

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There are three research reactors having fuel plates curved as circle-involute in the world: The Oak Ridge National Laboratory (ORNL) High Flux Isotope Reactor (HFIR), The Laue-Langevin institute (ILL) High Flux Reactor (RHF) and the Technical University of Munich (TUM) Research Neutron Source Heinz Maier-Leibnitz (FRM II). All three reactors are currently using Highly Enriched Uranium (HEU) as fuel and all three are actively engaged in activities to convert to Low Enriched Uranium (LEU) fuel. For various reasons, these reactors have expressed interest in using computational Fluid Dynamics (CFD) tools to perform their Steady-State Thermal-Hydraulic (SSTH) safety calculations. Using CFD for this type of analysis represents generally a significant move forward from traditional methods and therefore, acceptability by regulators is not straightforward. This is why Argonne National Laboratory (ANL) and the three involute-plate reactors formed an informal group to help each other in this endeavor. This so-called Involute Working Group (IWG) aims at justifying the usefulness and legitimacy of using CFD tools for SSTH safety calculations. Activities include benchmarking, code-to-code comparison, V&V and technical support. The present paper will describe the work currently performed under the IWG and what is planned for the future.