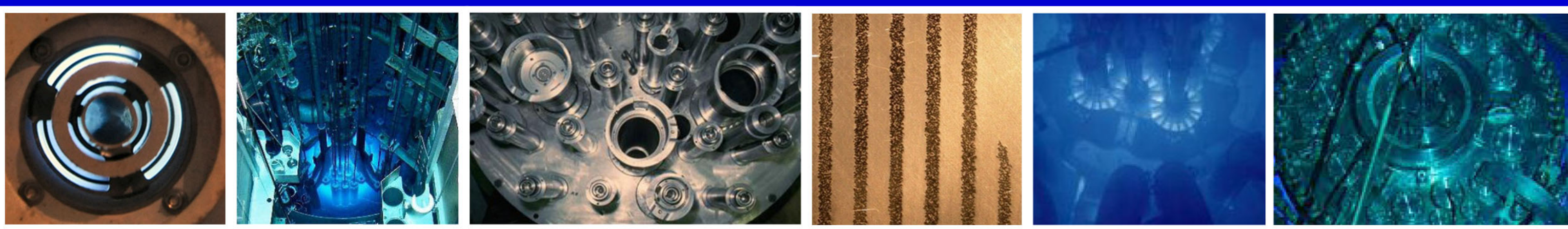
RERTR-2022

42ND International Meeting on Reduced Enrichment for Research and Test Reactors



October 2-5, 2022

IAEA, Vienna, Austria

www.rertr.anl.gov

Sunday, October 2

Registration: 4:30 – 6:00 pm, Vienna Marriott Hotel, Floor 1, Foyer II

Welcome Reception: 6:00 – 8:00 pm, Ballrooms A & B

#	Session Title	Time	Paper Title	Presenter	Organization/ Country
Monday, October 3					
Meeting Room: IAEA C-Building (Fourth Floor) Board Room C					
Opening and Welcome Chaired by Jeff Chamberlin DOE-NNSA		2:00 pm	Welcome to the IAEA and RERTR-2022 International Meeting: A World of Progress and Reopening	Amb. Laura Holgate	UNVIE/US
		2:10 pm		DDG Mikhail Chudakov	IAEA
		2:20 pm		Vyacheslav Gnyrya	NNC/ Kazakhstan
		2:40 pm		Jeff Chamberlin	NNSA/US
1	**HEU Minimization Programs** Chaired by Kyle Sallee DOE-NNSA	3:00 pm	1. IAEA Programs HEU Minimization, Research Reactor Projects, Fuel Return Program	John Dewes	IAEA
		3:20 pm	2. HEU Minimization in Japan	Masafumi Sato	MEXT/Japan
		3:25 pm	3. Fireside Chat: Offices of Conversion and Nuclear Material Removal	Chris Landers, Tiffany Blanchard-Case Facilitated by Kyle Sallee	NNSA/US
		4:00 pm	4. Proliferation Resistance Optimization (PRO-X) Overview	Alex Meehan	NNSA/US
4:20 pm Self-Hosted Refreshment Break					
2	**Proliferation Resistance Now and for the Future** Chaired by Alex Meehan DOE-NNSA	4:40 pm	1. Subcritical Experiment Using U-7Mo LEU Fuel at KUCA Facility	Hironobu Unesaki	Kyoto University/ Japan
		5:00 pm	2. Present Status of UTR-KINKI and Preliminary Feasibility Study on its Future Conversion to Low-Enriched Fuel	Genichiro Wakabayashi	Kindai/Japan
		5:20 pm	3. Nigeria Research Reactor-2 (NIRR-2) Project Perspectives on Proliferation Resistance	Sunday Jonah	CERT/Nigeria
		5:40 pm	4. INVAP Perspectives and Initiatives for Proliferation Resistance as Research Reactors Designer	Diego Ferraro	INVAP/ Argentina
		6:00 pm	5. IAEA Support to New Research Reactor Progammes: Planning for Sustained Utilization	Nuno Barradas	IAEA
6:20 pm Adjourn					
Tuesday, October 4					
Meeting Room: IAEA C-Building (Fourth Floor) Board Room C					
3	**Panel: Qualification and Fabrication of Sustainable Fuel** Chaired by John Stevens ANL	9:00 am	1. Integration of Fuel Testing and Fabrication Efforts to Support Regulatory Qualification of LEU U-10Mo Monolithic Fuel	James I. Cole	INL/US
		9:10 am	2. Overview of Research Reactor Fuel Development at KAERI	Jong Man Park	KAERI/Korea
		9:20 am	3.Framatome CERCA™ sustainability of LEU fuel fabrication	Dominique Geslin	CERCA/France
		9:30 am	Facilitated discussion among the three panelists to inject interplay of fuel fab constraints and QC and relationship to fuel qualification campaigns	Group	
10:00 am Self-Hosted Refreshment Break					
4	**Fuel Qualification and Irradiation Campaigns** Chaired by Bruno Baumeister TUM	10:20 am	1. Licensing Process of a New Fuel Type Element in Poland on an Example of the Experimental Fuel Element for Samples Irradiation in the Fast Neutron Spectrum	Maciej Lipka	NCBJ/Poland
		10:40 am	2. Developments of High-density Atomized U_3Si_2 Fuel Plates in KAERI	Tae Won Cho	KAERI/Korea
		11:00 am	3. Update of Fuel Meat Swelling Determination of Coated-(U-7Mo)/Al Dispersion Fuel from EMPIrE	Bei Ye	ANL/US
		11:20 am	4. New Results from the Scanning Electron Microscopy Characterization of Fuel Plates Irradiated in the EMPIrE Irradiation Experiment	Dennis Keiser	INL/US
		11:40 am	5. Post-irradiation Optical Microscopy, Chemical Burn-up Analysis, and Blister Threshold Testing of the MP-1 Irradiation Experiment	Adam Robinson	INL/US
		12:00 pm	6. Design of the University of Missouri Design Demonstration Element Test for Irradiation in the Advanced Test Reactor	Irina Glagolenko	INL/US

12:20 pm Lunch Break					
5	**Fuel Fabrication Challenges and Advances** Chaired by Yong Jin Jeong KAERI	1:40 pm	1. Status and Plans of U.S. U-10Mo Fuel Fabrication	Curt Lavender	PNNL/US
		2:00 pm	2. USHPRR HFIR Silicide Fabrication Update	Zach Huber	PNNL/US
		2:20 pm	3. USHPRR Critical Characteristics	Paul T Gee	PNNL/US
		2:40 pm	4. U-Mo Bare Foil Rolling Progress for FRM II Conversion	Kevin Buducan	TUM/Germany
3:00 pm — Refreshments, Poster Area					
6	**Poster Session and Refreshments** Chaired by Caryn Warsaw ANL	3:00 – 4:00 pm	Fuel Plate Cladding Thickness Estimation Thanks to Acoustic Microscopy	Abdelhak Megzari	U. Montpellier/ France
			Effect of Heat Treatments on the Irradiation Behavior of Monolithic U-Mo Fuels	Jan-Fong Jue	INL/US
			Development of Technology for Manufacturing of Dispersion Type Targets for Fission Mo-99 Production	Luis Olivares	CCHEN/Chile
			The HANARO Irradiation Test of Coated U-7Mo/Al-5Si Mini-plates	Yong Jin Jeong	KAERI/Korea
			Identification and Assessment of the Hazards in a Nuclear Fuel Fabrication Facility with LEU	Hade Elsayed	EAEA/Egypt
			Recent Developments in PLTEMP/ANL V4.3 Code for Research Reactor Thermal Hydraulics Analysis	Jeremy Licht	ANL/US
			The STAT7 V1.1 Code for Statistical Propagation of Uncertainties in Steady-State Thermal Hydraulics Analysis of Research Reactors	Erik Wilson	ANL/US
			Low Enriched Nuclear Fuel Based on Uranium-Zirconium Carbon-Nitride: Reactor Tests and Post-Reactor Studies	S.N. Sikorin	JIPNR-Sosny/ Belarus
			USHPRR MP-1 Irradiation Test: Assessment of Edge Pitting and Bond Line Corrosion in Vendor Produced Fuel Plates	Jeffrey Giglio	INL/US
			Non-destructive Post-Irradiation Examination and Fuel Swelling Analysis of the MP-1 Irradiation Experiment	Adam Robinson	INL/US
			Assessment of Critical Data for Qualification of U-10Mo Monolithic Fuel	William Hanson	INL/US
			Fabrication Process Research and Development to Support HFIR LEU Silicide	Zach Huber	PNNL/US
			Modeling of Thermal Conductivity in a Uranium Silicide Dispersion Fuel to Support Conversion of HFIR to LEU Fuels	Curt Lavender	PNNL/US
			Modeling Insights in Forming and Rolling Complex Geometries of Highly Loaded Uranium Silicide Dispersion Fuels	Curt Lavender	PNNL/US
			Recent Progress in U-10Mo Mechanical and Thermophysical Property Characterization	Jason Schulthess	INL/US
			Fine Mapping of the Power Density Distribution of MTR Fuel Using Gamma Spectroscopy	Guy Gabrieli	SNRC/Israel
			About the Limits of Optical Microscopy Measurement for Al-Fuel Cladding Thickness	Bertrand Stepnik	Framatome/ France
			PRO-X Auxiliary Capabilities: Balancing Performance & Proliferation for Research Reactor Products	M. Alex Brown	ANL/US
7	**U.S. High Performance Reactor Conversions** Chaired by Andrew Hebden ANL	4:00 pm	1. U.S. High Performance Research Reactor LEU Conversion Design, Testing and Fabrication Progress	Erik Wilson	ANL/US
		4:20 pm	2. Alternative HEU-LEU Mixed Core Transition Strategy for the MIT Research Reactor	Lin-Wen Hu	MIT/US
		4:40 pm	3. A Progress Update on the Highly Enriched Uranium to Low-Enriched Uranium Fuel Conversion at the University of Missouri Research Reactor	Maria Pinilla	MU/US
		5:00 pm	4. High Flux Isotope Reactor Low-Enriched Uranium Conversion Activities – 2022 Status Update	Carol Sizemore	ORNL/US
		5:20 pm	5. Analysis Methods for Lead Test Assemblies in the Advanced Test Reactor	Coilin Clark	INL/US
		5:40 pm	6. NIST Neutron Source Preconceptual Design	Dagistan Sahin	NIST/US
6:00 pm Adjourn					
Wednesday, October 5 Meeting Room: IAEA C-Building (Fourth Floor) Board Room C					
8	**HEU Removal Operations and Fuel Transportation** Chaired by Jeff England NAC Intl.	9:00 am	1. Packaging of Critical Assembly Fuel Materials for Shipment in the ES-3100	Trent Andes	CNS Y-12/US
		9:20 am	2. The Role of Nuclear Criticality Safety in Enabling the Transport of Highly Enriched Uranium (HEU) (and Other Fissile Materials) to Support Global Strategic Removal Projects	Charlotte Davis	NTS/UK
		9:40 am	3. Mobile Packaging Program Overview	Joshua Smith	NNSA/US
		10:00 am	4. Nigerian Nuclear Regulatory Authority Experience on NIRR-1 Core Conversion from HEU to LEU Fuel	Godwin Omeje	NNRA/Nigeria

10:20 am Self-Hosted Refreshment Break					
9	**International Reactor Conversion Progress and Partnerships** Chaired by Sunday Jonah CERT	10:40 am	1. Acceptance Test of WCTC with LEU Fuel at the IVG.1M Research Reactor Site in Kazakhstan	Igor Bolshinsky	INL/US
		11:00 am	2.First Steps for the Optimization of Experimental Facilities at FRM II during Conversion	Daniel Bonete-Wiese	TUM/Germany
		11:20 am	3. RELAP5 Safety Analyses in Support of the BR2 COBRA Lead Test Assembly Irradiation	Frank Wols	SCK-CEN/Belgium
		11:40 am	4. Benchmark between the MAIA and DART Fuel Performance Codes on the E-FUTURE U-Mo/Al Dispersion Fuel Test	Stéphane Valance	CEA/France
		12:00 pm	5. Water Channel Thickness Estimation through High Frequency Ultrasonic Measurements	Rhofrane Mrabti	U. Montpellier/ France
12:20 pm Lunch Break					
10	**Design and Analysis Methods** Chaired by Diego Ferraro INVAP	1:40 pm	1. Identification of Relevant Parameters for the Structural Analysis of an Involute LEU Fuel Plate	Aurelien Bergeron	ANL/US
		2:00 pm	2. First Steps Towards the Development of a Tool for Sensitivity Analysis and Uncertainty Propagation Studies for Steady-State Thermal-Hydraulic Simulations of Research Reactors	Ronja Schönecker	TUM/Germany
		2:20 pm	3. Improvements to Thermal-Hydraulics Models and Methods for MTR-Type Reactors	Mauro Nasso	INVAP/ Argentina
		2:40 pm	4. Neutronic Simulation of Curved Fuel Plate with Flat Plate Geometry	Lap-Yan Cheng	BNL/US
		3:00 pm	5. Process Modeling of U-10Mo and U_3Si_2 Using Integrated Computational Materials Engineering	Curt Lavender	PNNL/US
3:20 pm Self-Hosted Refreshment Break					
11	**Licensing and Conversion Reactor Experience** Chaired by Dennis Vinson SRNL	3:40 pm	1. Physical Start-up of IVG.1M Reactor with Low-Enriched Uranium Fuel	I.V. Prozorova	NNC/ Kazakhstan
		4:00 pm	2. Six-Year Experience of the WWR-K Reactor Operation with LEU Fuel	Asset Shaimerdenov	INP/ Kazakhstan
		4:20 pm	3. Practical Application of LEU Fuel for NIRR-1 Safe Operation	Kayode James Adedoyin	NNRA/Nigeria
		4:40 pm	4. Utilization and Operation of the Dalat Nuclear Research Reactor after Full Core Conversion	Kien Cuong Nguyen	VINATOM/ Vietnam
		5:00 pm	5. Five Years of Operating GHARR-1 on LEU Fuel: Successes and Challenges	Henry Cecil Odoi	GAEC/Ghana
5:20 pm Summary and Closure Jeff Chamberlin (DOE-NNSA) and John Stevens (ANL)					
5:40 pm Adjourn					

SESSION ABSTRACTS

Opening and Welcome

Welcome to the IAEA and RERTR-2022 International Meeting:

A World of Progress and Reopening

Session Chair: Jeff Chamberlin

Ambassador Laura S.H. Holgate
Boltzmanngasse 16, 1090 Vienna - Austria

Mikhail Chudakov
Deputy Director General, International Atomic Energy Agency, Vienna International Centre, PO Box 100 A-1400, Vienna -Austria

Vyacheslav Gnyrya (see abstract below)
Deputy Director for Tests, Institute of Atomic Energy branch of National Nuclear Center, 2B Beibyt Atom Street, Kurchatov, 071100 - Republic of Kazakhstan

Jeff Chamberlin
Assistant Deputy Administrator for Material Management & Minimization, US Department of Energy's National Nuclear Security Administration, 1000 Independence Ave., Washington DC 20585 – USA

The National Nuclear Center 30 Years' Experience in the Peaceful Use of Atomic Energy and WMD Non-Proliferation

E.G. Batyrbekov[1], **Gnyrya V.S**. [2], Baklanov V.V. [2], Koyanbayev Ye.T. [2]., Korovikov A.G. [2].
[1] National Nuclear Center of the Republic of Kazakhstan
[2] Institute of Atomic Energy, National Nuclear Center of the Republic of Kazakhstan

The National Nuclear Center was established on May 15th 1992 by the Decree of the First Kazakh President right after Kazakhstan became independent. Over the 30 years the National Nuclear Center has been successfully addressing the challenges stated by the Government and the Scientific Community.

One of the key missions assigned by the Government of Kazakhstan is to ensure the Non-Proliferation regime. The National Nuclear Center RK (NNC) has implemented three big projects on the elimination of the infrastructure engaged in the nuclear tests conduction and on the conversion of the military complex of the former Semipalatinsk Test Site (STS). In support of the Comprehensive Nuclear test Ban Treaty the NNC has realized several unique experiments aimed at the calibration of the global seismic stations network.

In 1995 the NNC in cooperation with Japan Atomic Power Company and Japan Nuclear Cycle Development Institute launched a big EAGLE Project.

In 2010 NNC successfully accomplished a big Project of the BN-350 spent fuel transportation from Aktau to the STS. In September 2012 longstanding activities on the elimination of the infrastructure at the Degelen site were completed as part of the Cooperative Threat Reduction Program.

Another significant mission of the NNC is research and investigations jointly implemented with international partners from TOSHIBA and Marubeni Utility Services. In cooperation with Japanese colleagues NNC has completed two-years project Fukushima Debris and CORMIT project. In 2019, the NNC concluded the seven-years contract with French Commissariat on atomic energy and alternative energy sources on SAIGA Project focused on the preparation of a big reactor experiment scheduled for conduction in 2024.

NNC has also made much progress towards the Conversion of two Kazakh research reactors IVG.1M and IGR to Low-Enriched Uranium. The Conversion Program was realized under support of the U.S. Department of Energy, Idaho National Laboratory and Argonne National Laboratory.

NNC specialists have realized remarkable physical startup of the IVG.1M research reactor with low-enriched core following the long-years preparation work. Our U.S. colleagues gave valuable support to NNC in preparation of the Feasibility Study for the Conversion of the IVG.1M research reactor to LEU-fuel, in choice of appropriate type of LEU fuel and Fuel Assemblies, drawing of the requirements for LEU fuel, testing of the LEU fuel and Safety analysis.

Today the NNC team has an increased focus on the development of the IGR-HEU fuel downblending technology by dry mixing and its utilization. The technology concept was endorsed by the IAEA and British Sellafield Company with high qualification in this field. NNC specialist are making calculations in support to the Conversion of the IGR research reactor to the LEU fuel.

NNC is additionally developing one more new enough strategy mission like controlled thermonuclear fusion. In 2019 NNC completed the final stage of the physical startup of the KTM site, right after that the KTM Tokamak was put into operation.

In conclusion it should be highlighted, that through the years the NNC take actions to provide the radiation safety everywhere in Kazakhstan including the former Semipalatinsk Test Site.

SESSION 1 HEU Minimization Programs

Session Chair: Kyle Sallee

1.1 IAEA Programs: HEU Minimization Research Reactor Projects, Fuel Return Program

John Dewes

International Atomic Energy Agency, Vienna International Centre, PO Box 100 A-1400, Vienna - Austria

1.2 HEU Minimization in Japan

Masafumi Sato

Ministry of Education, Culture, Sports, Science and Technology (MEXT) 3-2-2 Kasumigaseki, Chiyoda-ku, Tokyo 100-8959 – Japan

1.3 Fireside Chat: Offices of Conversion and Nuclear Material Removal

Chris Landers, Tiffany Blanchard-Case Facilitated by Kyle Sallee

US Department of Energy’s National Nuclear Security Administration, 1000 Independence Ave., Washington DC 20585 – USA

1.4 Proliferation Resistance Optimization (PRO-X) Overview

Alex Meehan

US Department of Energy’s National Nuclear Security Administration, 1000 Independence Ave., Washington DC 20585 – USA

SESSION 2 Proliferation Resistance Now and for the Future

Session Chair: Alex Meehan

2.1 Subcritical Experiment Using U-7Mo LEU Fuel at KUCA Facility

Y. Takahashi, Y. Kitamura, T. Misawa and **H. Unesaki**

Institute for Integrated Radiation and Nuclear Science, Kyoto University, Asashiro-Nishi-2, Kumatori-cho, Sennan-gun, Osaka 590-0494, Japan

This paper summarizes the result of the first subcritical experiment using U-7Mo LEU fuel at the KUCA facility of Institute for Integrated Nuclear and Radiation Science, Kyoto University. The U-7Mo Al dispersion type LEU fuel is in form of 2” x 2” square coupon, fabricated at CERCA Romans plant as the sample coupons for the forthcoming KUCA conversion and utilized the identical design specification for the KUCA LEU coupon fuel. The U-7Mo LEU fuel was loaded into the natural uranium subcritical pile constructed in the KUCA facility (reactor room), and neutronic characteristics including neutron flux distribution and subcriticality were measured. This experiment is the first series of the reactor physics experiment conducted using U-7Mo LEU fuel.

2.2 Present Status of UTR-KINKI and Preliminary Feasibility Study on its Future Conversion to Low-Enriched Fuel

C. D. Stratton, J. A. Morman and J. G. Stevens

Argonne National Laboratory, 9700 S. Cass Ave., Lemont, IL 60439 – USA

G. Wakabayashi, T. Sano, S. Hohara, and H. Yamanishi

Atomic Energy Research Institute Kindai University, 3-4-1 Kowakae Higashiosaka-shi, 577-8502 Osaka – Japan

The Kindai University Reactor (UTR-KINKI) is an educational and training reactor with a thermal power of 1 W, that started operation in 1961. The reactor is currently utilized for nuclear education from basic to advanced levels, including practical training for nuclear engineering students as well as for science teachers and foreign engineers. The reactor is currently fueled with HEU, and with the recent progress in the removal of HEU from other facilities in Japan, it is the only reactor left in Japan that is fueled with HEU. Considering its important role in Japan's nuclear education program, it is necessary to continue its operation by converting the reactor to LEU fuel. The results of a preliminary technical feasibility study show that LEU fuel plates can be configured to provide an appropriate value of excess reactivity with only a slight reduction in peak neutron flux while maintaining control rod worths and shutdown margin.

2.3 Nigeria Research Reactor-2 (NIRR-2) Project Perspectives on Proliferation Resistance

S. A. Jonah

Centre for Energy Research and Training, (Nigeria Atomic Energy Commission) Ahmadu Bello University, Zaria – Nigeria

Nigeria, with a population of over 200 million people, is Africa’s most populous nation and has a lot of thriving industries, tertiary institutions and numerous Research Centres. The only Research Reactor in the country, NIRR-1 is operated by the Centre for Energy Research and Training in Zaria. NIRR-1 was converted to LEU under the aegis of the DOE-NNSA-M3 initiatives and the IAEA in 2018. In order to expand the capacity for Human Resources Development (HRD) towards the generation of electricity based on nuclear energy, a Multi-Purpose Research Reactor (MPRR) of nominal power of 10 MWth is proposed.

Consequently, this paper outlines preliminary work performed under the IAEA TC project for the MPRR and perspectives for integrating Proliferation Resistance into design of the reactor and its auxiliary systems

2.4 INVAP Perspectives and Initiatives for Proliferation Resistance as Research Reactors Designers

Alicia Doval, **Diego Ferraro**, Daniel Hergenrede, Eduardo Villarino

INVAP S.E., Av Cmte. Piedrabuena 4950, R8404, Bariloche - Argentina

INVAP S.E., Esmeralda 356, C1035 Buenos Aires – Argentina

INVAP is an Argentine tech-company recognized as one of the leaders within the research reactor industry, with more than four decades in the field and several on-going projects worldwide. INVAP's projects portfolio covers not only the Research Reactors but also the associated facilities such as Radioisotope Production Facilities, neutron science Halls and Fuel Fabrication Plants. In this line, the consideration of good practices from non-proliferation point of view represents a key factor within the design.

As a nuclear Vendor INVAP has been developing special skills and capabilities to deal with diverse customers and regulatory bodies, even with different cultural features. These capabilities allow the customer to be integrated within the project, to develop core capabilities and to share a smooth but effective knowledge transfer in high-visibility projects.

This paper summarizes INVAP´s ongoing Research reactor projects and its partnership to tailor the reactor design according to the customer´s objectives and capabilities, making emphasis in their key characteristics, from the designer point of view, that enhance the proliferation resistance.

2.5 IAEA Support to New Research Reactor Programmes: Planning for Sustained Utilization

Nuno Barradas

International Atomic Energy Agency, Vienna International Centre, PO Box 100 A-1400, Vienna - Austria

As research reactors age and in some cases no longer fulfill the purpose for which they were built and do not find a renewed justification for their continued operation, national authorities and operating organizations take the decision to permanently shut down and then decommission facilities. Correspondingly, the number of operational research reactors has been slowly but steadily decreasing since the mid 1970s. However, this reduction is not felt the same way in all the regions, and the number of operational research reactors in developing countries has increased from 82 in 2000 to 89 in 2020. At the same time, in recent years there is a renewed interest in developing new research reactor projects. According to the 2021 edition of the IAEA Nuclear Technology Review, there are 11 research reactors in construction in eight countries, and more than 20 other countries have formal plans or are considering building a new research reactor. In many cases this would be the first nuclear installation in the country.

The IAEA has developed the Milestones Approach, providing guidance on the timely preparation of a research reactor project through a sequential development process. The research reactor programme starts with a justification for the research reactor based on the national or regional needs for research reactor services, the availability of alternatives, and the availability of sufficient financial, technical and human resources. Developing the justification and stakeholder base is an essential step to ensure that it does not become underutilized and thereby a potential cause for concern on safety or security grounds and a burden on the country's resources. This presentation addresses the support of the IAEA to new research reactor programmes, focusing on strategic planning, aimed at a sustained utilization during the reactor's operational lifetime. The Agency support and the tools it has developed for this purpose are presented discussed.

SESSION 3 Panel: Qualification and Fabrication of Sustainable Fuel

Session Chair: John Stevens

3.1 Integration of Fuel Testing and Fabrication Efforts to Support Regulatory Qualification of LEU U-10Mo Monolithic Fuel

James Cole

Idaho National Laboratory, P.O. Box 1625, Idaho Falls, ID 83415 – USA

Within the US, the High-Performance Research Reactor Project has been tasked with the development and qualification of high density low-enriched uranium (LEU) fuel systems and establishing commercial scale fabrication capability leading to HPRR conversions. For the LEU U-10Mo monolithic fuel system, the project has entered a regulatory qualification phase in parallel with commercial scale fabrication development and optimization. These two activities are intimately linked as irradiation testing data must confirm that changes to the fabrication process to improve commercial viability do not negatively impact demonstrated fuel performance attributes. Efforts are coordinated through a centralized Design Authority for the generic U-10Mo monolithic fuel system. The Design Authority maintains ownership of the fuel product specification that serves as the basis for the current fuel experiment fabrication specifications and will inform the conversion element specifications. The fuel product specification is periodically updated to capture the most current information on fabrication process limits and irradiation performance data that serves as the technical basis for specification requirements and will ultimately be provided as supporting information for regulatory review.

3.2 Overview of Research Reactor Fuel Development at KAERI

J.M. Park, Y.J. Jeong, S.H. Kim, T.W. Cho, J.S. Yim, Y.W. Tahk, H.J. Kim

Korea Atomic Energy Research Institute, 111 Daedeok-daero, 989 Beon-gil, Yuseong, Daejeon, 34057 - Republic of Korea

KAERI has developed the atomization technology to be applied for research reactor fuel by overcoming disadvantages of conventional powdering method. Atomization technology is capable of manufacturing spherical U-Si, U-Mo, and U-Al alloy powders in commercial scale and has been proven as a key technology for HEU minimization in research reactors. In addition, KAERI has developed diverse fuel fabrication technology and its facility based on atomized process, those are pin-type U_3Si dispersion fuel for HANARO, U-Mo plate fuel for KJRR and HPRR, U_3Si_2 plate fuel, and fission Mo-99 target with UAl_x. Qualification for the KJRR U-Mo fuel, including LTA and HAMP tests, has been carried out successfully so far, and KAERI has recently launched a R&D program to develop and qualify plate-type U_3Si_2 fuel, in which U-density is up to 5.3 g-U/cc. This paper overviews how fuel qualification campaigns and fuel fabrication innovations at KAERI have been coordinated in a timely manner.

3.3 Framatome CERCA™ Sustainability of LEU Fuel Fabrication

Cyrille Rontards, **Dominique Geslin**

Framatome CERCA, 2 rue Professeur Jean Bernard, F-69007 LYON – France

Framatome CERCA, established in France in 1957, supplies research-reactors all over the world with nuclear fuel and Mo-99 medical targets. It is a major player in metallic fuel manufacturing and transport, with more than 1,000,000 fuel plates and 6,500 fuel assemblies delivered to its customers.

Framatome CERCA™ was involved in many conversion projects (pool types reactors, TRIGA reactors, critical facilities, Mo-99 production) in partnership with European Union, the US DoE or the IAEA.

Sustaining a nuclear fuel production capability requires regular investments and permanent innovation.

This paper focuses on the recent actions made by Framatome CERCA™ : improvement of the industrial footprint (upgrade, investment), R&D, people ; all of them are required to face today supply requests and to prepare tomorrow's challenges.

3.4 Facilitated Discussion Among the Three Panelists to Inject Interplay of Fuel Fabrication Constraints and QC and Relationship to Fuel Qualification Campaigns

Group Discussion

SESSION 4 Fuel Qualification and Irradiation Campaigns

Session Chair: **Bruno Baumeister**

4.1 Licensing Process of a New Fuel Type Element in Poland on an Example of the Experimental Fuel Element for Samples Irradiation in the Fast Neutron Spectrum

Maciej Lipka, Rafał Prokopowicz, Anna Talarowska, Michał Dorosz, Zuzanna Marcinkowska, Tomasz Machtyl

National Centre for Nuclear Research, ul. Andrzeja Sołtana 7, 05-400 Otwock-Świerk – Poland

Operating new type of nuclear fuel in the research reactor requires walking through the licensing process and obtaining the license from the national nuclear regulatory body. MARIA reactor currently have two types of fuel element in the routine element, but in the last months the third have obtained the license and will start operation in Q3 2022. The article reviews the licensing process, both from the perspective of the formal requirements, and measurements and calculations performer by the reactor's operational and engineering team. They include but are not limited to: thermal-hydraulic, source term, reactivity, neutron spectrum calculation, and hydraulic and neutronic measurements. Recent completion of the fuel element licensing process means that the MARIA reactor's team is ready to repeat the process for the potential new fuel elements from new suppliers.

4.2 Developments of High-density Atomized U_3Si_2 Fuel Plates in KAERI

T.W. Cho, S.H. Kim, J.M. Park, Y.J. Jeong

Korea Atomic Energy Research Institute (KAERI), Daedeok-daero 989beon-gil, Daejeon, 34057 -Republic of South Korea

Jared Wight

SCK CEN, Belgian Nuclear Research Centre, Boeretang 200, BE-2400 MOL – Belgium

KAERI initiated the KAERI high-density atomized silicide fuel Qualification Irradiation (KIMQI) project with SCK CEN. This project aims at demonstration of the irradiation behaviors of the high-density atomized U_3Si_2 dispersion fuel. As its first step, KAERI established a fabrication system of the flat high-density atomized U_3Si_2 dispersion fuel and fabricated 5 full-size fuel plates for the irradiation test. The irradiation test was designed to meet general fuel qualification conditions required for high-power MTRs. Two irradiation cycles were completed in SCK CEN BR2 reactor and the experiment achieved the targeted heat flux and burnup. Underwater visual observations and tests showed stable irradiation behavior and visual inspections revealed only typical discoloration in the high heat flux zone. The fuel plates were extracted from the basket easily, indicating that the plates are flat or that the warping of the plates should be very minor. In addition, a wet shipping inspection indicated there were no signs of abnormal fission product release after either cycle.

4.3 Update of Fuel Meat Swelling Determination of Coated-(U-7Mo)/Al Dispersion Fuel from EMPIrE

Yeon Soo Kim, Z.-G. Mei, **B. Ye,** L.M. Jamison, G.L. Hofman, A.M. Yacout

Argonne National Laboratory, 9700 S. Cass Ave., Lemont, IL 60439 – USA

A.B. Robinson, W.A. Hanson

Idaho National Laboratory, P.O. Box 1625, Idaho Falls, ID 83415 – USA

Several methods of deriving fuel meat swelling of the diffusion-barrier coated U-7Mo/Al dispersion fuel plates irradiated in the EMPIrE (European Mini Plate Irradiation Experiment) test were explored and compared. The available irradiation data of the EMPIrE plates include plate thickness data and fuel plate cross-section optical microscopy images. The typical method to estimate fuel meat swelling is to divide plate thickness change caused by irradiation with the pre-irradiation fuel meat thickness. Because the pre-irradiation meat thickness cannot be measured directly for the plates that underwent irradiation, an approximate method is to use the nominal value defined by the fuel plate specification. For the EMPIrE plates, the pre-irradiation meat thicknesses of the archive plates measured using optical microscopy images showed that the measurements varied relative to the nominal value to a degree significant to the interpretation of swelling. This necessitated the reassessment of the pre-irradiation meat thickness of the irradiated plates using post-irradiation optical microscopy images. Using the reassessed pre-irradiation meat thicknesses, the fuel meat swelling of the plates selected for destructive examination from EMPIrE was determined.

4.4 New Results from the Scanning Electron Microscopy Characterization of Fuel Plates Irradiated in the EMPIrE Irradiation Experiment

Dennis Keiser Jr., Daniele Salvato, Charlyne Smith, Tammy Trowbridge, William Hanson, Adam Robinson, and Irina Y. Glagolenko

Idaho National Laboratory, P. O. Box 1625, Idaho Falls, ID 83415-6188 – USA

Bei Ye, Laura M. Jamison, and Gerard L. Hofman

Argonne National Laboratory, 9700 S. Cass Ave., Lemont, IL 60439 – USA

Irradiation of fuel plates in the European Mini-Plate Irradiation Experiment (EMPIrE) has been completed. The mini-plates (2.54 cm in width × 10.16 cm in length) were irradiated in the Advanced Test Reactor (ATR) located at the Idaho National Laboratory (INL). The dispersion fuel plates were comprised of zirconium nitride (ZrN)-coated U-Mo fuel particles in an aluminum (Al) matrix, clad in Al-alloy. Several fabrication parameters were varied as a part of the experiment design to evaluate their influence on the fuel performance, including fuel powder heat treatment, coating methodology, coating thickness, fuel particle size distribution, Mo-alloying content, and fuel powder source. Of interest is the swelling behavior of the fuel plate, the effectiveness of the ZrN coating in mitigating fuel/matrix interaction during irradiation, and other microstructural developments in the fuel system. Samples from a selection of EMPIRE fuel plates have been microstructurally characterized along transverse cross-sections taken near the midplane using scanning electron microscopy (SEM). The results of the SEM characterization are presented here.

4.5 Post-irradiation Optical Microscopy, Chemical Burn-up Analysis, and Blister Threshold Testing of the MP-1 Irradiation Experiment

A.B. Robinson, W.A. Hanson, N.J. Lybeck, M.A. Plummer, John W. Merickel, A.J. Winston, J.J. Giglio, M.A. Marshall, J.I. Cole

Idaho National Laboratory, 1955 N. Fremont Ave, Idaho Falls, ID 83415 – USA

Destructive post-irradiation examination (DE-PIE), including optical microscopy (OM), chemical burn-up analysis, and blister threshold testing, was completed for the MP-1 irradiation experiment. The experiment is the most recent monolithic, uranium-10wt%molybdenum (U-10Mo) plate-type fuel test to support the conversion of United States High Performance Research Reactors (USHPRRs) to utilize low-enriched uranium (LEU) fuel. Specifically, MP-1 evaluated 62 plates fabricated at a commercial scale against lab scale fabrications. Following non-destructive PIE, plates were strategically allocated to DE-PIE characterization techniques: 22 mini-plates were sectioned for OM, with a sub-set of 9 also characterized by chemical burn-up analysis, and 19 mini-plates were annealed for blister-threshold testing. The microstructural evolution of the U-10Mo foils appeared consistent with historic results and highlights will be presented along with a comparison between the chemical burn-up analysis and as-run neutronics analysis, and blister testing results were used to develop an updated blister threshold testing model.

4.6 Design of the University of Missouri Design Demonstration Element Test for Irradiation in the Advanced Test Reactor

I. Glagolenko, G. Housley, K. Anderson, J. Lower, H. Hiruta, J. Nielsen, G. Hawkes, C. Jesse, K. Gagnon, J. Fishler, T. Howell

Idaho National Laboratory P.O. Box 1625 Idaho Falls, ID 83415 – USA

The team at Idaho National Laboratory is tasked with the design of the University of Missouri Design Demonstration Element (MURR DDE) test in the Advanced Test Reactor (ATR). MURR DDE is the last in the series of irradiation tests of monolithic U-10Mo fuel in support of conversion of the MURR reactor from Highly Enriched Uranium (HEU) to Low Enriched Uranium (LEU). The main goal of the test, which is sponsored by the Office of Materials Management and Minimization (M^3), DOE-NNSA is to demonstrate acceptable performance of the LEU U-10Mo fuel in an element configuration and at operating conditions prototypic of MURR LEU conversion element. The paper describes details of the experiment design and the various challenges that exist in the design space.

SESSION 5 Fuel Fabrication Challenges and Advances

Session Chair: Yong Jin Jeong

5.1 Status and Plans of U.S. U-10Mo Fuel Fabrication

C.A. Lavender, M.R. Rossiter, C.K. Clayton

Pacific Northwest National Laboratory, 902 Battelle Blvd., Richland, WA 99354 – USA

R.B. Nielson

Idaho National Laboratory, P.O. Box 1625, Idaho Falls, ID 83415 – USA

G. Argon

BWX Technologies, Inc., P.O. Box 785, Lynchburg, VA 24505 – USA

D. Turpin

Y-12 National Security Complex, P.O. Box 2009, Oak Ridge, TN 37831-8112 – USA

5.2 USHPRR HFIR Silicide Fabrication Update

Z. Huber, E. Conte, M. Rossiter, K. Brooks, C. Painter, V. Joshi, C. Lavender

Pacific Northwest National Laboratory, 902 Battelle Blvd, Richland, WA 99354 – USA

High volume fractions of U_3Si_2 fuels create unique fabrication and inspection challenges. The High Flux Isotope Reactor (HFIR) creates even more unique challenges because of the complex contoured fuel shape and the use of a discrete burnable absorber layer. In fabrication trials of rectangular fuel zones, the typical challenges observed were dog bone, thin clad areas, heterogeneous fuel zones, and inspection uncertainties. An attempt to resolve the issues was then undertaken at Pacific Northwest National Laboratory. To this end, modeling and experimental work was pursued with surrogate materials to determine if an alternative composite processing approach would help to resolve challenges. The correlations to fabrication and simulation were excellent and could be used widely for highly loaded composites. This presentation will discuss timelines relating to fabrication, challenges with highly loaded dispersion fuels, and how the Fuel Fabrication Pillar will address many of the challenges with the alternative composite processing approach.

5.3 USHPRR Critical Characteristics

Paul Gee

Pacific Northwest National Laboratory, 902 Battelle Blvd, Richland, WA 99354 – USA

This presentation will be given at Reduced Enrichment Research and Test Reactor meeting in Vienna, Austria, October 3, 4 & 5, 2022. This power point slide show presentation on Critical Fuel Characteristics explains their basis and value when creating a productive, manufacturing quality system based on the end user requirements and expectations of Safety, Function, Performance and Durability. This is intended to be explanatory regarding what they are, how to utilize them and the potential benefits they can provide within the research reactor mission scope. As a part of this, the benefits of utilizing a database such as a system in development by PNNL specifically for this application will be included.

5.4 U-Mo Bare Foil Rolling Progress for FRM II Conversion

Buducan K.[1,2], Lorand S.[1], Stepnik B.[1], Rontard C.[1], Gauche F.[1], Baumeister B.[2], Schwarz C.[2], Chemnitz T.[2], Petry W.[2]

[1]Framatome, 2 Rue Professeur Jean Bernard, 69007 Lyon – France

[2]Forschungs-Neutronenquelle Heinz Maier-Leibnitz (FRM II) Technische Universität München, Lichtenbergstr. 1, 85747 Garching – Germany

CERCA division and FRM II are working together to develop a European pilot line for uranium-molybdenum (U-Mo) monolithic fuel plate manufacturing. This line aims to supply the FRM II reactor with lower enriched fuel for its conversion from actual dispersed U3Si2 highly enriched fuel (HEU). Many processes are involved, from U-Mo casting for alloying and ingot shaping, to zirconium coating by physical vapor deposition (PVD) and aluminum cladding. The main process is a flat rolling process, both hot & cold, to produce U-Mo bare foils. Specific geometry, surface condition and material properties have to be assured in order to obtain high quality bare foils. These requirements are achievable thanks to a precise control of rolling parameters process such as roll speed, working temperature and reduction ratio. This paper will focus on the flat rolling progress for U-Mo bare foil production in CERCA division. The global foil quality such as thickness distribution, surface quality and material properties will be investigated for further improvement.

SESSION 6 Poster Session

Session Chair: Caryn Warsaw

6.1 Fuel Plate Cladding Thickness Estimation Thanks to Acoustic Microscopy

A. Megzari[1,2], E. Le Clézio[2], B. Stepnik[1], and G. Despaux[2]

[1] Framatome, 2 Rue Professeur Jean Bernard, 69007 Lyon – France

[2] IES, Univ Montpellier, 641 Av. du Doyen Gaston Giraud, 34000 Montpellier - France

When the nuclear fuel is in the form of plates, the resulting structures can be seen as heterogeneous tri-layers made of a fissionable core and an aluminum cladding. For safety reasons, the cladding thickness of these plates must be measured.

Currently, this measurement is done by a destructive method. The purpose of this project is to replace this procedure by a non-destructive measurement thanks to acoustic microscopy allowing two dimensional measurements to be performed and the preservation of the controlled plates.

The presentation will focus on the specifications of the acoustic microscope to allow interaction between the ultrasonic waves and the granular interface. The signal processing allowing the time-of-flight estimation will then be presented. Finally, a comparison between the current process results and the thickness estimations obtained through acoustic microscopy will be made, proving the relevance of a non-destructive control of the fuel plates to image the cladding thickness.

6.2 Effect of Heat Treatments on the Irradiation Behavior of Monolithic U-Mo Fuels

J.F. Jue, T.L. Trowbridge, F.G. Di Lemma, A. J. Winston, D.D. Keiser, J.J. Giglio, and J.I. Cole
Idaho National Laboratory, 1955 N. Fremont Avenue, Idaho Falls, ID 83415 – USA

In the MP-1 irradiation experiment, new heat treatments were introduced to improve Mo homogeneity and pre-irradiation U-Mo microstructure. The new heat treatments are expected to have positive impacts on the irradiation performance of monolithic U-Mo fuels. In this study, the microstructural evolution of irradiated MP-1 U-Mo fuel plates produced with and without heat treatments was documented. Grain refining in U-Mo was observed in regions where pre-irradiation γ-U decomposition was present and was especially abundant in the fuel plates without heat treatments. Strong evidence of reverse transformation under irradiation (α + γ'-$U_2Mo \rightarrow$bcc γ-U) at very low burnup was also observed. The larger grain size observed in the heat-treated fuel plates before irradiation led to fewer intergranular porosity/precipitates after irradiation. In this presentation, the significant microstructural features of the irradiated MP-1 fuel plates will be provided and their impacts on U-Mo irradiation performance will be discussed.

6.3 Development of Technology for Manufacturing of Dispersion Type Targets for Fission Mo-99 Production

A. Luis Olivares, Jaime Lisboa, Mario Barrera
Chilean Nuclear Energy Commission – CCHEN, Nueva Bilbao 12501, Santiago 7600713 – Chile
B. Allison Castro, Gisel Carter, Jorge Marin,
University of Santiago of Chile – USACH, Av. Lib. B. O'Higgins 3363, Santiago 9170022 – Chile

Nowadays, the most promissory method for producing Mo-99 on an industrial scale is by means of fission of ^{235}U through neutron irradiation of targets containing low enriched uranium in research reactors. The current methodological trend is manufacturing of small plates containing a core or "meat" comprised by UAl_3 or UAl_4 compounds, dispersed in an aluminum matrix, using similar methods to the manufacture and inspection of full size plates applied as nuclear fuel for research reactors.

Currently at CCHEN is being working out an R&D project aiming to develop the methodology to obtain dispersion-type targets, based on the experience and available facilities. This was a result of its previously developed capacity for production of fuel assemblies for research reactors.

A proposal for improvement productivity and process efficiency of this technology is to increase the uranium load inside the targets. Using UAl_2 stoichiometric composition, this parameter is limited to less than 3 gU/cm^3, value that could be significantly increased up to 4.9 gU/cm^3 using particles made of hypostoichiometric U-Al alloy (U+10wt% Al).

This paper summarizes the results of targets manufactured with standard stoichiometric of U-18.5wt% Al alloy particles (UAl2), dispersed in an aluminum matrix, verifying the phase transformations to UAl4 during the fabrication process through XRD, SEM and microanalysis. For the hypo-stoichiometric U-Al alloy, it was possible to observe the effect of homogenization annealing, characterizing its phase composition and confirming the ability to transform ingots into particles through conventional grinding methods. The next activity will be to prepare a set of plate type targets using this hypo-stoichiometric alloy, with higher uranium load, as a way to increase the uranium density in dispersion type LEU targets.

6.4 The HANARO Irradiation Test of Coated U-7Mo/Al-5Si Mini-plates

Sunghwan Kim and **Yong Jin Jeong**
Korea Atomic Energy Research Institute, 111 Daedeok-daero, 989 Beon-gil, Yuseong-gu, Daejeon, 34057 - Republic of Korea

HEU to LEU conversion in civilian facilities to minimize the amount of nuclear material available that could be used for nuclear weapons has been worked under the GTRI and RERTR. Research reactor fuel society continues to support them by developing fuel fabrication technology necessary to enable the conversion of the high performance research reactors in Europe and the USA. High-density LEU U-Mo dispersion fuels have been chosen for the promising candidate thanks to the higher uranium loading than that of conventional U3Si2/Al dispersion fuel. In this study, KAERI performed coating on the surface of atomized U-7Mo powders with Zr or Mo, respectively by PVD to suppress interaction between U-Mo and Al-5Si matrix. Subsequently, each coated U-7Mo was fabricated into the mini-plate fuels with a U loading of 8.0 g-U/cm3 for the irradiation test at HANARO. From the design of fuels, fuel fabrication process, inspection results, and as-run analysis were described.

6.5 Identification and Assessment of the Hazards in a Nuclear Fuel Fabrication Facility with LEU

H. Elsayed
Egyptian Atomic Energy Authority, 3 Ahamed Elzomor St., P.O.Box 9621, Cairo 11787 – Egypt

In uranium fuel fabrication facilities with enriched uranium 20%, large amounts of radioactive material are present in a dispersible form. This is particularly so in the early stages of the fuel fabrication process. In addition, the radioactive material encountered exists in diverse chemical and physical forms and is used in conjunction with flammable or chemically reactive substances as part of the process. Thus, in these facilities, the main hazards are potential criticality and releases of uranium hexafluoride (UF_6) and (U_3O_8), from which workers, public and the environment should be protected. In nuclear fuel fabrication facility, the process for the obtainment of U_3O_8 for fuel elements fabrication for research reactors starting from UF_6 comprises two well defined stages characterized by the risks involved in the raw materials and intermediate products. The first stage is the wet process (conversion process) includes the hydrolysis of UF_6 to UO_2F_2 and posterior precipitation to ammonium diuranate (ADU); the second stage is dry process to obtain the U_3O_8 powder from ADU at high temperature.

This work will be show the analysis of the hazards and events in nuclear fuel fabrication facility that have as a consequence the stated risks, their detection and prevention to protect the workers, public and environment from the radiological and chemical hazards.

6.6 Recent Developments in PLTEMP/ANL V4.4 Code for Research Reactor Thermal Hydraulics Analysis

M. Kalimullah, A. P. Olson, E. E. Feldman, B. Ozar, D. S. Yoon, S. Yang, and **J. R. Licht**
Argonne National Laboratory, 9700 South Cass Avenue, Argonne, IL 60439 – USA

PLTEMP/ANL is a steady-state single-phase (liquid H_2O or D_2O) thermal hydraulics code. The code has been used by Argonne National Laboratory and the research reactor community since 1980 to calculate temperature distribution and safety margins in research reactors whose cores contain fuel assemblies made of multiple fuel plates, nested fuel tubes, or fuel rods. Its capabilities include calculating channel flow rates in forced or natural circulation flows, the temperature distribution in the coolant, cladding, and fuel, and the calculation of margins to the onset of nucleate boiling, to the onset of flow instability, and to critical heat flux. Over the last decade, the verification of 16 capabilities frequently used by research reactor analysts has been performed and documented. The latest version of the code, PLTEMP/ANL V4.4 includes new capabilities: 1) three-dimensional heat conduction in the fueled as well as the unfueled lengths of all fuel plates of MTR-type fuel assemblies, 2) eddy-induced coolant mixing in the coolant channels, and 3) the implementation of H_2O properties based on the IAPWS 1997 Industrial Formulation. The heat conduction calculation uses an iterative scheme to solve a three-dimensional finite-volume representation of the assembly.

6.7 The STAT7 V1.1 Code for Statistical Propagation of Uncertainties in Steady-State Thermal Hydraulics Analysis of Research Reactors

S. Yang, D. S. Yoon, J. Rudolph, J. Thomas, A. P. Olson, and **E. H. Wilson**
Argonne National Laboratory, 9700 South Cass Avenue, Lemont, IL 60439 – USA

L.-W. Hu and Y. Zhao
MIT Nuclear Reactor Laboratory, 138 Albany Street, MA 02139 – USA

STAT7 is a steady-state, single-phase thermal hydraulics (TH) software for plate-fueled research and test reactors that supports statistical propagation of uncertainties of fabrication tolerances and other key reactor parameters in analysis using Monte Carlo method. Application of the software includes the reactor conversion from highly enriched to low-enriched uranium fuel of U.S. High-Performance Research Reactors such as MITR-II. STAT7 can calculate multiple histories of datasets that includes flow rate of coolant and bypass channels and temperature distribution of the coolant and fuel plate with or without fins. Furthermore, it can predict operational safety margins for the onset of nucleate boiling or onset of flow instability, including statistical uncertainties. The statistical sampling and TH capabilities of STAT7 have been validated and verified against hand calculations, other codes, and experimental data. STAT7 has been actively employed in various TH analyses to support MITR-II LEU conversion, such as TH impact assessment of fuel specification and mixed HEU-LEU transition analysis.

6.8 Low Enriched Nuclear Fuel Based on Uranium-Zirconium Carbon-Nitride: Reactor Tests and Post-Reactor Studies

S.N. Sikorin, S.G. Mandzik, S.A. Polazau, A.V. Kuzmin, T.K. Hryharovich
State Scientific Institution “JIPNR-Sosny”, PO BOX 119, Minsk, 220109 - Belarus

Sh.T. Tukhvatulin, A.N. Bakhin, I.E. Galev, A.A. Mokruyshin
LUCH JSC, 24, Zheleznodorozhnaya St., Podolsk, Moscow region, 142103 – Russia

A.L. Izhutov, N.K. Kalinina
JSC “SSC RIAR”, 9, Zapadnoye Shosse, Dimitrovgrad, Ulyanovsk region, 433510 - Russia

The creation of energy-intensive nuclear power plants requires the use of nuclear fuel capable of withstanding various effects of neutron fields, high temperatures and thermal stresses during operation. At present, such a fuel can be the uranium-zirconium carbonitride (UZrCN) with a low oxygen content (less than 0.1 wt. %), developed at the LUCH JSC, State Atomic Energy Corporation “Rosatom”. Uranium-zirconium carbonitride combines the positive qualities of fuels based on uranium carbides and nitrides, namely: the heat conduction of UZrCN – based fuel is almost 10 times higher, the ultimate stress is almost 3 times higher, and the volumetric swelling is 3 times lower than that of UO2; compared to UN, it has a reduced yield of gaseous fission products, less tendency to swell and a significantly higher operating temperature; compared to UC it has increased compatibility with structural materials. Due to its optimal characteristics, UZrCN is an attractive fuel material for use in various types of reactors. The limited use of this type of fuel is due to the small amount of information about the behavior and characteristics of UZrCN under irradiation, especially at deep burnup levels. To eliminate this problem, in the framework of RRRFR programme an international project is currently being implemented to study the properties and characteristics of UZrCN in conditions of irradiation to reach 5, 15, 40% fifa burnup with the energy release up to 600 W/cm3. The project have been implemented step-by-step through the solution of a series of research and computational and experimental tasks. Starting from 2016, the tasks of developing the design of the experimental capsule and the irradiation device have been solved; for this, neutron-physical, thermophysical and strength computations of the experimental capsule and the irradiation device have been carried out, and a program of out-of-pile experiments has been implemented. In 2019, a methodical experiment has been carried out in the SM-3 research reactor (JSC “SSC RIAR”, State Atomic Energy Corporation “Rosatom”) with a duration of 23.3 eff. days with the 0.63% fifa burnup reached and post-reactor studies of the irradiated fuel have been performed. Starting from July 2021, systematic preparations have been made for a reactor experiment to study the properties and characteristics of UZrCN in conditions of irradiation until the 5% fifa burnup has been reached. As a result of the work carried out, an experimental capsule, an irradiation device have been manufactured, and on December 19, 2021, a reactor experiment with a duration of 200 eff. days has started.

The critical facilities “Giacint” and “Kristal”, located in the scientific institution “JIPNR – Sosny”, National Academy of Science of Belarus are used for the preparation of critical facility experiments on the multiplication units, simulating physical characteristics of the cores using low-enriched UZrCN-based nuclear fuel (19,75% U-235) so as to use them in the reactors on fast and intermediate neutrons with gas or liquid-metal coolants. It is planned to investigate the fast-neutron critical assemblies with three types of fuel cassettes and different matrix materials (air, aluminum and lead). Also, it is planned to investigate the intermediate-neutron critical assemblies with a hydride-zirconium moderator. The side reflectors of these critical assemblies are beryllium (internal layer) and stainless steel (external layer).

The report describes in detail the results of the work carried out on irradiation of UZrCN in the SM-3 research reactor, as well as a description of the design and composition of critical assemblies with UZrCN- based fuel, and the results of calculations.

6.9 USHPRR MP-1 Irradiation Test: Assessment of Edge Pitting and Bond Line Corrosion in Vendor Produced Fuel Plates

J. J. Giglio, J. F. Jue, M. P. Johnson, J.I. Cole
Idaho National Laboratory, 1955 N. Fremont Ave, Idaho Falls, ID – USA

This presentation will summarize an effort to identify the source of anomalous localized edge pitting and bond line corrosive attack of experimental mini fuel plates manufactured at the commercial fuel vendor and tested in the Advanced Test Reactor (ATR) at Idaho National Laboratory (INL) as part of the Miniplate 1 (MP-1) irradiation test. The MP-1 test was conducted by the US High Performance Research Reactor Project (USHPRR) and overseen by the Fuel Qualification Pillar (FQ). The test involved evaluation of a mix of 62 “vendor” fuel plates and 12 “laboratory” fuel plates fabricated at Idaho National Laboratory using commercial-scale and lab-scale fabrication processes, respectively. The major objective of the MP-1 experiment was to test, for the first time, miniplates fabricated by a commercial fuel vendor over the range of irradiation conditions relevant to conversion of NRC regulated High Performance Research Reactors (HPRRs) from high-enriched uranium to LEU. The focus of the efforts described in this presentation are the vendor fuel plates to inform changes in the fabrication process prior to fabrication of the next miniplate experiment, MP-2

6.10 Non-destructive Post-Irradiation Examination and Fuel Swelling Analysis of the MP-1 Irradiation Experiment

W.A. Hanson, **A.B. Robinson,** N.J. Lybeck, J.J. Giglio, M.A. Marshall, J.I. Cole
Idaho National Laboratory, 1955 N. Fremont Ave, Idaho Falls, ID – USA

Non-destructive post-irradiation examination (NDE-PIE) and a fuel-foil swelling analysis were completed for the Mini-plate 1 (MP-1) irradiation experiment. This is the most recent test to support the generic qualification of low-enriched uranium (LEU), U-10Mo monolithic, plate-type fuel. The irradiation performance of 62 plates fabricated at a commercial scale was compared against 12 INL fabricated plates. Two fuel geometries from each source were irradiated at the INL Advanced Test Reactor, with “thick-fuel” plates irradiated at a low-power (5-10 kW/cm^3) to a moderate fission density (0.76-2.82 × 10^{21} fission/cm^3) and “thin-fuel” plates irradiated at a medium-power (~20-35 kW/cm^3) to a high fission density (2.85-5.56 × 10^{21} fissions/cm^3). Highlights of the experiment NDE-PIE, including visual examination, neutron radiography, gamma spectrometry, and plate profilometry, will be presented. Additionally, the MP-1 fuel-foil swelling behavior was characterized and will be compared against the USHPRR-FQ currently recommended model of the U-10Mo swelling.

6.11 Assessment of Critical Data for Qualification of U-10Mo Monolithic Fuel

M.A. Marshall, **W.A. Hanson,** M.A. Plummer, J.A. Evans, N.J. Lybeck, J.I. Cole
Idaho National Laboratory, 1955 N. Fremont Ave, Idaho Falls, ID, USA

In the coming years, the United States High Performance Research Reactor Fuel Qualification (USHPRR-FQ) Project will conduct three unique irradiation tests in support of generic qualification of U-10Mo monolithic fuel through the U.S. Nuclear Regulatory Commission (NRC). Data gathered from these tests will be used to compile a topical report that establishes fuel properties and limiting performance behavior of U-10Mo monolithic fuel under normal and off-normal operating conditions. Fuel qualification requirements have previously been documented. An assessment has been performed to identify critical post-irradiation data to be collected and how it will support qualification. A minimum number of samples is recommended for post-irradiation measurements that are used to define fuel performance or properties. Experiment designs are evaluated to confirm that reactor operational ranges are covered and analysis can be performed to evaluate the impact of variables such as fission density, power density, fuel thickness, and geometry on performance and properties.

6.12 Fabrication Process Research and Development to Support HFIR LEU Silicide

E. Conte, M. Rossiter, K. Brooks, C. Painter, V. Joshi, C. Lavender, **Z. Huber**
Pacific Northwest National Laboratory, 902 Battelle Blvd, 99354, Richland, WA – USA

This poster will present a process flow diagram for fabrication of a highly loaded uranium silicide dispersion fuel. The fuel is needed for converting The High Flux Isotope Reactor (HFIR) to a low enriched fuel. Along each of the process steps, examples of fabrication research and development, both modeling and experimental, carried out at Pacific Northwest National Laboratory (PNNL), will be shown. The fabrication process presented aims to solve the unique challenges presented by high volume fractions of fuel such as dog bone, thin cladding sections, and heterogeneous fuel zones.

6.13 Modeling of Thermal Conductivity in a Uranium Silicide Dispersion Fuel to Support Conversion of HFIR to LEU Fuels

L. Li, A. Soulami, V. Joshi, K. Choi, K. Brooks, **C. Lavender**, Z. Huber
Pacific Northwest National Laboratory, 902 Battelle Blvd, 99354, Richland, WA – USA

This poster will present the methodology and findings of a microstructure-based finite element method (FEM) to study the effects of particle size distribution and particle morphology on thermal conductivity and heat fluxes at the surface of a uranium silicide (U_3Si_2) dispersion fuel. A parametric study using the FEM model is carried out to investigate the sensitivities of U_3Si_2 volume fraction, average particle size, heat generation rate, and surrogate particle shapes on the thermophysical properties of the unirradiated, cladded U3Si2-Al fuel plate.

6.14 Modeling Insights in Forming and Rolling Complex Geometries of Highly Loaded Uranium Silicide Dispersion Fuels

K. Choi, A. Soulami, V. Joshi, K. Brooks, **C. Lavender**, Z. Huber
Pacific Northwest National Laboratory, 902 Battelle Blvd, 99354, Richland, WA – USA

This poster presents a finite element method (FEM) investigation of forming and rolling processes to support fabrication of complex geometries for conversion of The High Flux Reactor (HFIR) to a LEU silicide fuel with a discrete burnable absorber layer. This work has driven multiple fabrication process changes to overcome the challenges associated with highly loaded metal matrix composite processing. Of particular importance, this work has driven changes in geometries and matrix strengths that reduce dog bone and cladding thickness issues while increasing fabricability of the complex fuel shape.

6.15 Recent Progress in U-10Mo Mechanical and Thermophysical Property Characterization

J. Schulthess, T.R. Pavlov, M. Heighes, C. Adkins, C. Tan, D. Salvato, M. Marshall, A. Hanson, J. J. Giglio, J. I. Cole

Idaho National Laboratory P.O. Box 1625, Idaho Falls, ID 83415 – USA

Results are presented on the mechanical and thermophysical property characterization of U-10Mo as part of the MP-1 qualification campaign. The thermal diffusivity, specific heat and thermal conductivity of as fabricated U-10Mo fuel foils and plates were measured using the thermal conductivity microscope (TCM), differential scanning calorimeter (DSC) and laser flash analyzer (LFA) all of which located in a shielded glove box – thermal property cell (TPC). A novel inverse method based on finite element analysis was developed to evaluate the thermal diffusivity of composite/layered materials via the LFA technique such as the 5-lyered U-10Mo monolithic fuel mini-plates. (INL/CON-22-69019)

6.16 Fine Mapping of the Power Density Distribution of MTR Fuel Using Gamma Spectroscopy

G. Gabrieli, U. Steinitz, Z. Yungrais, O. Aviv, L. Danon, A. Krakovich, T. Makmal, I. Neder

Soreq Nuclear Research Centre, rd. 4111, Yavne 8180000 – Israel

An accurate power-density distribution estimation is essential in planning, commissioning, and maintaining the operation of nuclear reactors. A flux trap such as a water volume in the core increases the thermal flux around it, and thus the power peaking factor (PPF) may rise, reducing safety margins. To predict the effect, high-resolution simulations are required, including local burnup calculations. Thus, experimental validation of computed PPF allows a better estimation of the calculation errors. Here we report measurements allowing fine 2D mapping of the surface power density in MTR fuel assemblies, analyzing fission products by gamma spectroscopy. We probed a fuel assembly positioned near a flux trap in the Israeli Research Reactor - 1 (IRR-1). We found a steep power density profile, indicating a need for a high tally resolution for correct prediction of this effect. However, the resulting PPF agreed well with full-core 3D Monte-Carlo simulation with homogeneous fuel-plate burnup assumption.

6.17 About the Limits of Optical Microscopy Measurement for Al-Fuel Cladding Thickness

N. Hibert[1], A. Megzari [2,3], G. Despaux[2], E. Le Clézio[2], O. Tougait[1], **B. Stepnik[3]**,

[1] Unité de Catalyse et Chimie du Solide, Université de Lille, Cité Scientifique, F-59650 Villeneuve d'Ascq – France

[2] IES, Univ Montpellier, CNRS, Montpellier - France

[3] Framatome, CERCA, SPL, ZI Les Bérauds, 54 avenue de la déportation, BP 114, F-26104 Romans-sur-Isère – France

The Al-based cladding is the only confinement barrier for research reactor fuels with respect to the water in the pools, so its integrity must be scrupulously preserved until the core assemblies are reprocessed. At each stage of production, stringent controls are carried out, such as dimensional examinations, visual inspections and a blister test carried out at 450°C. In addition to these non-destructive tests, fuel plates are also randomly selected for direct measurement of the cladding thickness. This additional control consists in taking cuts from specific areas of the fuel-plate which, once polished, are subject to a metrological measurement.

This destructive examination which is time and cost-consuming would need to be replaced, especially since optical measurements also have their own limits. In this work, we have attempted to stretch the limits of these optical measurements to find the best compromise between high level of accuracy and effectiveness of the process.

6.18 PRO-X Auxilliary Capabilities: Balancing Performance & Proliferation for Research Reactor Product

M. Alex Brown, Anna Servis, Katherine Hepler
Argonne National Laboratory, 9700 S Cass Ave, Lemont, IL, 60439 – USA

Shan Peters
Savannah River National Laboratory, Aiken, SC 29808 – USA

SESSION 7 U.S. High Performance Reactor Conversions

Session Chair: Andrew Hebden

7.1 U.S. High Performance Research Reactor LEU Conversion Design, Testing and Fabrication Progress

E.H. Wilson, D. Jaluvka, A. Hebden, W. Cowherd, D. Yoon, K. Anderson, L. Jamison, J. Stillman
Argonne National Laboratory, 9700 South Cass Ave, Lemont, IL 60439 – USA

U.S. 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 U.S. High Performance Research Reactor (USHPRR) portion of the DOE/NNSA Material Management and Minimization (M^3) Reactor Conversion program. The effort to convert the USHPRR is conducted by eleven U.S. 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 2018, all five USHPRR have been completed with preliminary LEU U-10Mo designs, including the reactor at Idaho National Laboratory (ATR), and including preliminary Safety Analysis Report documentation for conversions of the reactors at the University of Missouri (MURR), Massachusetts Institute of Technology (MITR), and the National Institute of Standards and Technology (NBSR).

The High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory has more recently completed preliminary designs based on an existing uranium silicide–aluminum dispersion (U_3Si_2-Al) fuel due to a greater similarity to the HEU U_3O_8-Al dispersion fuel presently used in HFIR. Pursuing plans for HFIR to convert on silicide fuel, the reactor conversion program plans irradiation testing of HFIR LEU plate designs, including those containing boron. These tests are required since design parameters exceed existing silicide fuel qualification limits (e.g., power density).

Presently, USHPRR Reactor Conversion Pillar efforts include analysis of fabrication specification impacts on reactor operations. These analyses are being studied concurrent with a campaign for plate fabrication demonstrations being carried out by the Fuel Fabrication Pillar. Data from these evaluations, including material compositions and information related to specification limits, are informing design decisions and future fabrication specification development.

Alongside ongoing plate-level irradiation testing specific to each reactor's fuel element design, fuel element flow tests, where required by design changes, are in preparation in order to finalize the information required in the Safety Analysis Reports prior to conversion to LEU fuel.

7.2 Alternative HEU-LEU Mixed Core Transition Strategy for the MIT Research Reactor

L. Hu and Y. Zhao
MIT Nuclear Reactor Laboratory, 138 Albany St, Cambridge, MA 02139 – USA

E. H. Wilson, S. Yang, D. Yoon, K. Anderson, D. Jaluvka
Argonne National Laboratory, 9700 South Cass Ave, Lemont, IL 60439 – USA

The Massachusetts Institute of Technology Reactor (MITR) is a 6 MW research reactor operating with highly enriched uranium (HEU) finned plate-type fuel. Its compact core design produces neutron flux and energy spectrum comparable to light water power reactors. The primary changes for the low-enriched uranium (LEU) conversion are the fuel element design from 15-plate finned HEU fuel to 19-plate unfinned "FYT" LEU fuel, reactor operating power increases from 6 MW to 7 MW, and a higher primary coolant flow rate from 2000 gpm to 2400 gpm. Conversion analyses conclude that the proposed LEU fuel design with power increase will maintain sufficient safety margins during steady-state operation and transients, fuel cycle lengths, and reactor performance to support its experimental program. Detailed analyses were completed for the transition cycles from the all-fresh LEU core starting with 22 elements, and demonstrated both neutronic and thermal hydraulic safety requirements are met throughout the transition cycles to equilibrium LEU operation. An alternative conversion strategy is proposed which involves a gradual transition from an all-HEU equilibrium core to an all-LEU equilibrium core by replacing three HEU fuel elements with fresh LEU fuel elements during each fuel cycle while maintaining the operating power and primary coolant flow rate at 6 MW and 2400 gpm, respectively. Thermal hydraulic analyses consist of (1) statistical steady-state analysis using STAT7, which takes into account fuel fabrication tolerances, measurement and modeling uncertainties; and (2) best-estimate steady-state and loss of primary flow transient analyses using RELAP5/MOD3.3. The results show that flow rate in HEU and LEU coolant channels decreases as the transition cycles progress because there are more coolant channels in the core. HEU coolant channels have lower flow rates than LEU, due to higher hydraulic resistance in HEU finned coolant channels. The thermal hydraulic safety margin evaluated by STAT7 is higher in LEU fuel where onset of nucleate boiling (ONB) power (at 3-σ confidence level) at LSSS coolant outlet condition ranges from 11.43 MW (mixed-7 core) to 14.22 MW (mixed-1 core), and HEU fuel from 8.51 MW (mixed-2 core) to 10.22 MW (mixed-6 core). RELAP5 predicted cladding surface temperatures in HEU and LEU hot channels to have >10°C margin to ONB temperature. During a loss of primary flow transient, boiling is predicted to occur briefly (for about 6 s) in a HEU fuel element. The maximum fuel temperatures are <110°C for both HEU and LEU, significantly lower than their limits, 450°C and 400°C, respectively. In addition to the analysis completed using steady-state MCNP mixed LEU/HEU core peaking factors, next steps include updating software model capabilities and core loadings to deplete with both HEU and LEU loaded in the core. Facility modifications started with the procurement of new pumps and additional heat exchanger plates to allow 2400 gpm primary flow and 7 MW operation. This work is ongoing with additional equipment instrumentation procurement, and installation in the next few years.

7.3 A Progress Update on the Highly Enriched Uranium to Low-Enriched Uranium Fuel Conversion at the University of Missouri Research Reactor

Wilson M. Cowherd, John A. Stillman, Dhongik S. Yoon, Valerio Mascolino, Guanyi Wang, Cezary Bojanowski, Erik H. Wilson

Argonne National Laboratory, 9700 S Cass Ave, 60439 Lemont – USA

Kiratadas Kutikkad, Leslie P. Foyto, Ronald Astrino, **Maria Pinilla**, Nickie Peters

University of Missouri Research Reactor, 1513 Research Park Drive, 65211 Columbia – USA

The University of Missouri Research Reactor (MURR®) is a 10 MW research reactor on the campus of the University of Missouri in Columbia, Missouri. Analyses are underway to convert the reactor from highly enriched uranium (HEU) U-Al_x dispersion fuel to low-enriched uranium (LEU) monolithic fuel consisting of uranium-10 wt% molybdenum (U-10Mo). Recent highlights include a release of the revision to the LEU fuel element specification and drawings for fabrication, progress on the impact study of the fuel element specification and drawing tolerances on the neutronics and thermal hydraulic performance of the reactor, and the completion of the MURR LEU element and MURR Design Demonstration Element (DDE) structural end fitting rigidity analysis. These three activities will contribute to further refinement of the MURR specifications and drawings and fabrication of the LEU element.

7.4 High Flux Isotope Reactor Low-Enriched Uranium Conversion Activities – 2022 Status Update

C. W. Sizemore, Z. A. Bacon, B. R. Betzler, J. W. Bae, K. Borowiec, D. Chandler, D. H. Cook, V. D. Fudurich, C.J. Hurt, P. K. Jain, W. C. Lowe, E. Popov, T. B. Smith

Oak Ridge National Laboratory, 1 Bethel Valley Rd, Oak Ridge, TN 37830-6399 – USA

The Oak Ridge National Laboratory (ORNL) High Flux Isotope Reactor (HFIR) is a high power density research reactor currently operating at 85 MWth. HFIR supports a variety of scientific missions, including cold and thermal neutron scattering science, isotope production, and materials irradiation. Ongoing efforts to convert HFIR from highly enriched uranium (HEU) to low-enriched uranium (LEU) are focusing on uranium silicide dispersion fuel (U_3Si_2-Al) design. Developments for 2021-2022 include fuel drawings and specifications, and evaluation of reactor physics metrics and thermal-hydraulics preliminary accident analysis for the optimized High Density (5.3 gU/cc) and Low Density (4.8 gU/cc) uranium silicide fuel. Additionally, recent progress using modeling tools for direct numerical simulation (DNS) of HFIR central subchannel and multi-physics modeling to benchmark Cheverton-Kelley and Gambill-Bundy experiments will be discussed.

7.5 Analysis Methods for Lead Test Assemblies in the Advanced Test Reactor

Thomas Maddock and Eric Woolstenhulme

Idaho National Laboratory, 1955 North Fremont Avenue, Idaho Falls – USA

In preparation for high assay low enriched (HALEU) lead test assembly insertion at the Advanced Test Reactor (ATR), numerous analyses have been developed to support the licensing safety basis. The methodology and tools utilized to perform these new analyses are modernized relative to the legacy ATR safety basis. In particular, major updates were made to the neutronics analysis tools used to design core reload patterns. Updates were also made to the fuel plate and coolant channel thermal-hydraulic analysis tools. Monte Carlo sampling methods and analysis automation using python are key components of the new analysis processes. Benefits of using these updated tools are that the ATR will have a clearer understanding of what critical shim positions to expect for a given cycle design, which design features are the most critical to safety and operations, and which accident scenarios are the most limiting to their operational window.

7.6 NIST Neutron Source Preconceptual Design

Dagistan Sahin, Osman S. Celikten, Jeremy C. Cook, Abdullah G. Weiss, Thomas H. Newton

NIST Center for Neutron Research, 100 Bureau Drive, Gaithersburg, 20899 – USA

The National Bureau of Standards Reactor (NBSR) at the NIST Center for Neutron Research (NCNR) provides a safe and reliable neutron source for the thousands of visiting the U.S. and international researchers annually. A new state-of-the-art research reactor utilizing U-10Mo (19.75% enriched Y-12 LEU) is being designed for the NCNR to replace the aging NBSR. The new reactor (NIST Neutron Source, or NNS) will be tailored primarily for neutron science involving thermal and cold neutron beams. This work provides preconceptual design characteristics for the proposed NIST neutron source, highlights of reactor core neutronic and thermal-hydraulic analysis results, and detailed description for user facilities, thermal neutron guides, cold neutron sources and cold neutron guide network. The initial results imply a total cold neutron current gain at the guide entrances ranging between 4.8 and 6.4, and at least a factor 2 increase in the thermal neutron Maxwellian brightness with respect to the current NBSR.

SESSION 8 HEU Removal Operations and Fuel Transportation

Session Chair: Jeff England

8.1 Packaging of Critical Assembly Fuel Materials for Shipment in the ES-3100

T. Andes and B. Martin
Y-12 National Security Complex, P.O. Box 2009, Oak Ridge, TN 37831-8112 – USA

H. Unesaki and Y. Takahashi
Kyoto University, Asashiro-Nishi-2, Kumatori-cho, Sennan-gun, Osaka 590-0494 – Japan

This paper summarizes the packaging activities of fuel materials at the KUCA facility during the HEU return campaign, of which its completion was formally announced on August 2022. Characterization of the materials (historical records, identification, weighing, radiological measurements, ORIGEN models) and actual packing procedures as well as the major "good practice" is described. Our experience would be an exercise on how we can accomplish a challenging effort such as packaging so many fuel materials with efficiency.

8.2 The Role of Nuclear Criticality Safety in Enabling the Transport of Highly Enriched Uranium (HEU) (and Other Fissile Materials) to Support Global Strategic Removal Projects

Michelle Nuttall, **Charlotte Davis**
Nuclear Transport Solutions, Hinton House, Risley, Warrington WA3 6GR - United Kingdom

Nuclear Transport Solutions (NTS) has for many decades undertaken all Categories of marine transports in support of critical removal operations in support of the US DOE and other Government stakeholders around the world. In addition to the external shipping operations, NTS also provides a lot of essential expertise and capability to support these operations – such expertise which can easily go unnoticed to the external stakeholders and end customers. NTS' in-house capability provides nuclear shielding and criticality safety analyses for the transport of fissile materials across the globe. The authors have over 45 years of combined experience within the nuclear industry; primarily in the area of criticality safety. This paper will provide insight into the complexity and strategic importance associated with the criticality modelling of fissile nuclear materials proposed for transportation. Transport criticality safety assessments are completed in compliance with international regulatory frameworks (which govern and enable the multitude of global initiatives to remove and transport fissile nuclear materials) in support of security and other associated national prioritisation programmes.

8.3 Mobile Packaging Program Overview

J.H. Smith and I.G. Kapuza
US Department of Energy's National Nuclear Security Administration, 1000 Independence Ave., Washington DC 20585 – USA

S.L. Cleveland
Oak Ridge National Laboratory, 1 Bethel Valley Rd, Oak Ridge, TN 37830-6399 – USA

D.S. Lowry
Savannah River National Laboratory, Aiken, SC 29808 – USA

S.S. Mattmann
Y-12 National Security Complex, P.O. Box 2009, Oak Ridge, TN 37831-8112 – USA

The U.S. Department of Energy's National Nuclear Security Administration (DOE/NNSA) plays a critical role in the nation's ability to prevent, counter, and respond to nuclear threats around the world. Within DOE/NNSA, the Office of Material Management and Minimization's Office of Nuclear Material Removal executes some of these functions via the Mobile Packaging (MP) program. MP maintains

capabilities to rapidly deploy technical experts and specialized equipment anywhere in the world to recover nuclear materials on short notice. These activities are carried out using the Mobile Uranium Facility (MUF) and Mobile Plutonium Facility (MPF), which provide the United States Government a capability to quickly and safely characterize, stabilize, package, and remove nuclear materials from austere environments in a variety of circumstances. DOE/NNSA performs periodic exercises to ensure that the personnel and equipment are ready when called upon. An overview of the recent training exercises will be discussed, as well as the future plans for MUF and MPF.

8.4 Nigerian Nuclear Regulatory Authority Experience on NIRR-1 Core Conversion from HEU to LEU Fuel

Godwin Okewu Omeje

Nigerian Nuclear Regulatory Authority, Plot 564/565, Independence Avenue, Central Business District, Abuja – Nigeria

NIRR-1, is a Miniature Neutron Source Reactor (MNSR) designed and supplied by the China National Nuclear Corporation (CNNC) under a project supply agreement (PSA) and used for scientific research, neutron activation analysis, education and training, operating since February 2004. The project to convert the reactor from HEU to LEU was re-initiated by the Nigeria Atomic Energy Commission in 2016 after initial pre-conversion activities that started in 2011 and was supported by China, Norway, the United Kingdom, the United States and the IAEA.

This paper shares the Nigerian Nuclear Regulatory Authority experience with lessons learned ranging from the verification and validation of conversion safety analysis, authorizations issued (including permits, certifications and licenses with relevant hold points as well as licensing conditions) safeguards verification exercise, safe and secure transportation of the HEU and LEU cores, discharge of the HEU core from the reactor vessel, loading of LEU core into the reactor vessel, detectors installation and instrumentation connection and criticality, reactivity adjustment, power calibration, low and full power operation and dose measurements.

SESSION 9 International Reactor Conversion Progress and Partnerships

Session Chair: Sunday Jonah

9.1 Acceptance Test of WCTC with LEU Fuel at the IVG.1M Research Reactor Site in Kazakhstan

Igor Bolshinsky, Sandor Tozser, Dennis Keiser

Idaho National Laboratory, 1955 N. Fremont Ave, Idaho Falls, ID, USA

Erlan Batyrkokov, Victor Baklanov, Vyacheslav Gnyrya, Yerbolat Koyanbayev

National Nuclear Center of the Republic of Kazakhstan, 10, Beybit Atom St., Kurchatov, AO, 071100 - Republic of Kazakhstan

The water-cooled technological channels (WCTC) with low-enriched uranium (LEU) fuel, in the amount required for conversion, were delivered to the IVG.1M site by the manufacturer in February 2021 and thereafter the site acceptance test (SAT) of the WCTCs and the fuel elements started immediately. The paper provides an overview of the SAT conducted between March and November 2021 by the designated experts of the reactor operator and the manufacturer. It includes the introduction of the IVG.1M reactor and its unique WCTCs, and the inspection methodology to verify the conformity of the quality of the LEU fuel with the *Technical Design* (reference document). The paper presents results of the non-destructive and destructive tests, the outcomes of the thermohydraulic measurements, as well as the evaluation and corrective actions (if any), including the amendments (modifications) of the *Technical Design* initiated by the manufacturer based on the test results. Finally, the paper draws conclusions on the effectiveness of the SAT method used, captures consolidated experiential knowledge and shares lessons learned that can be used in general when planning and performing fuel verification.

9.2 First Steps for the Optimization of Experimental Facilities at FRM II During Conversion

D. Bonete Wiese, C. Reiter
Forschungs-Neutronenquelle Heinz Maier Leibnitz, Technische Universität München, Lichtenbergstr. 1, Garching, 85748, Bavaria – Germany

The conversion of the FRM II will impact the scientific performance. To get a better picture of the effects the instruments have to deal with, one needs information about the reactor, neutron guide and instrument setup. To achieve this, two well-established Monte Carlo codes will be coupled: Serpent, used for full-core neutronics, and McStas, used for neutron ray-tracing through the instrument's optic elements. Also the results will open up ideas for improvements outside the core to minimize the losses of neutron flux due to conversion. The script-based coupling will help instrument scientists optimize their equipment after conversion, but it will also allow predicting flux changes along the irradiation cycles and whenever any short-term modifications might occur in the moderator tank. In this work, a general review of the instruments is presented, and then a detailed scheme of a one-way coupling is given, including some examples of calculations the script will automate. Finally, potential future applications are discussed.

9.3 RELAP5 Safety Analyses in Support of the BR2 COBRA Lead Test Assembly Irradiation

H. Lin, P. Garner, J. Licht
Argonne National Laboratory, 9700 S Cass Ave, 60439 Lemont – USA

F. Wols
SCK CEN, Belgian Nuclear Research Centre, Boeretang 200, BE-2400 MOL – Belgium

A selection of relevant design base transients was modelled with RELAP5-3D in support of the safety file for the COBRA Lead Test Assembly (LTA) Irradiation inside BR2. HEU and LEU COBRA fuel elements at the limiting heat flux were added in parallel branches next to the limiting former BR2 standard fuel element implemented in the current BR2 RELAP5-3D model to compare transient behavior of the different fuel element types. Simulations were performed for a loss of flow with natural circulation bypass valve opening (pump trip, historical Test A) and loss of pressure with bypass valve opening or not (Test F or G). No nucleate boiling is found in Test A for all three fuel elements. In Test F and Test G, nucleate boiling occurs during flow reversal and stops after natural circulation flow becomes stable. Although fuel temperatures are higher for LEU COBRA during steady-state conditions, the trends of fuel, cladding and coolant temperatures are similar for the three fuel elements as the transient unfolds, meaning that the COBRA LEU LTAs can be expected to behave similarly in terms of safety following BR2's most relevant design base core cooling perturbations.

9.4 Benchmark Between the MAIA and DART Fuel Performance Codes on the E-Future U-Mo/Al Dispersion Fuel Test

Denis Lorenzo, Gentien Marois, Hervé Palancher, **Stéphane Valance**
CEA, IRESNE/DEC/SESC, Centre de Cadarache, 13108 St Paul Lez Durance cedex – France

Bei Ye**,** Shipeng Shu, Abdellatif Yacout Program has Bei presenting
Argonne National Laboratory, 9700 S. Cass Ave, Lemont, IL 60439 – USA

Argonne and CEA have collaborated to verify the predictive characters of their respective MTR fuel plate modeling codes (DART and PLEIADES/MAIA) since 2018. Code-to-code comparisons on many important parameters were performed, such as the evolution of fuel temperature, fuel meat thermal conductivity, swelling, oxide growth, and fuel meat constituent volume fractions. Some of the calculated results were also compared to post-irradiation examination (PIE) data. The benchmark efforts were performed using the data of plate 4202 irradiated in the E-FUTURE test, which has a fuel meat composition of U-7Mo particles embedded in an Al matrix with 4 wt% Si addition.

The results comparison showed a good overall match between the two codes on all the values of interest and demonstrated a reasonable agreement between both codes and the experimental results. Although slight differences were observed, they are understandable. This benchmark effort was a reassuring exercise on the predictive capacities of both DART and PLEIADES/MAIA codes for dispersed U(Mo) fuel. This comparison also allowed the adjustment of some parameters and the identification of some improvement paths.

9.5 Water Channel Thickness Estimation through High Frequency Ultrasonic Measurements

Rhofrane Mrabti, Emmanuel Le Clézio, Gilles Despaux
IES, Univ Montpellier, CNRS, Montpellier – France

Yoann Calzavara
Laue Langevin Institute – ILL, 71 Avenue Des Martyrs, Grenoble F-38000 – France

As part of the conversion of the ILL fuel element to low-enriched uranium, a non-destructive PIE device has been developed to measure the swelling of the plates after irradiation. This device integrates two ultrasonic transducers inserted in a stainless-steel blade and resonating around 100 MHz. An ultrasonic wave propagates through the structure, producing series of acoustic echoes leading to the water channel thickness by a time-of-flight measurement. For the optimization of the device, it is necessary to understand the propagation behavior in the structure. We will present the latest Finite Different Time Domain simulation of the transducer-water channel-fuel plates system. The reflection of the ultrasonic waves on the fuel plate surfaces will be analyzed to estimate the channel thickness. Therefore, by simulating the propagation wave in the structure, we aim to analyze the physics of the generated echoes through the overall path.

SESSION 10 Design and Analysis Methods

Session Chair: Diego Ferraro

10.1 Identification of Relevant Parameters for the Structural Analysis of an Involute LEU Fuel Plate

M. Sitek, C. Bojanowski, **A. Bergeron**, J. Licht
Argonne National Laboratory, 9700 S Cass Avenue, Lemont, IL 60439 – USA

The HFIR, RHF, and FRM II are in a unique class of research reactors that use involute-shaped fuel plates to form their cores. A conversion from HEU to LEU fuel plates will require a core redesign. The plate geometry and fuel thermal-physical properties are going to change and therefore, the structural response of the LEU plates from the pressure differential and temperature loads they will be subjected to may differ from the current design. Fluid structure interaction (FSI) and thermal analyses may be needed to verify that the LEU plates will maintain their structural integrity. To better understand how the design variables (e.g. plate and fuel dimensions, number of plates in the core, inlet plenum length, material properties and constraints) affect the structural response of the plates, multi-variate FSI and thermal expansion sensitivity analyses have been performed using a generic involute fuel plate. The paper will discuss the key results obtained and identify the parameters that have been found to be the most relevant for the structural response of an involute LEU fuel plate.

10.2 First Steps Towards the Development of a Tool for Sensitivity Analysis and Uncertainty Propagation Studies for Steady-State Thermal-Hydraulic Simulations of Research Reactors

R. Schönecker, W. Petry and C. Reiter
Forschungs-Neutronenquelle Heinz Maier-Leibnitz Technische Universität München, Lichtenbergstraße 1, 85748 Garching – Germany

F. Thomas and Y. Calzavara
Institut Laue-Langevin, 71 Avenue des Martyrs, 38000 Grenoble – France

P. Bianchini
Milano Multiphysics, c/o Polihub, Via Durando 39, 20156 Milano – Italy

The two high-performance research reactors FRM II and RHF are strongly working towards the conversion of their fuel elements to lower enriched uranium. To identify potential new fuel element designs, it is crucial to carry out steady-state thermal-hydraulic (SSTH) safety margin calculations, which take into account various sources of uncertainties influencing the reactor performance.

For this purpose, a python-tool for sensitivity analysis (SA) and uncertainty propagation (UP) is currently developed. With this tool, SA and UP studies based on statistical methods can be performed on models implemented in PLTEMP, a SSTH code from Argonne National Laboratory. However, to allow for a broader range of application, it is also possible to couple the tool to python models or to results from other codes. This paper provides an overview of the tool and presents first-order SA examples demonstrating different coupling schemes. Furthermore, steps towards the coupling to a commercial CFD software are presented. In the future, the capabilities will be extended to also enable the analysis of second-order effects.

10.3 Improvements to Thermal-Hydraulics Models and Methods for MTR-Type Reactors

B. Ozar, M. Kalimullah, E. Feldman, J. Stevens**,**
Argonne National Laboratory, 9700 South Cass Ave., Argonne, IL 60439 – USA

A. Doval, G. Natali L. Mataloni, E. Villarino, **M. Nasso**
INVAP S.E., Av Cmte. Piedrabuena 4950, R8404, Bariloche – Argentina

For research reactor analysis, largely 1D thermal-hydraulics models are commonly used because they are simple to apply and technically defendable. Improvements to methods and models can reduce modeling uncertainties and thereby allow greater reactor performance with no reduction in predicted safety margins. Using the OPAL reactor as a benchmark, ANL and INVAP collaborated to improve methods and models in two major areas: i) capturing the asymmetric cooling of the two end fuel plates of a fuel assembly by modeling the entire fuel assembly as a thermally coupled entity rather than as a series of individual independent symmetrically-cooled fuel plates and ii) using a separate 2D heat conduction model to inform the 1D model. This paper presents the results of these analyses. One of the major conclusions is that when the entire fuel assembly is modeled subdividing the fuel meat width into several parts, the calculated peak heat fluxes and temperatures of the 1D model envelope those in the 2D model without excessive conservatism and with much less modeling complexity and computational power. These improvements in the thermal-hydraulics models potentially expand the design space, making additional designs feasible, which, in turn, may achieve better fuel utilization and proliferation resistance in the designs.

10.4 Neutronic Simulation of Curved Fuel Plate with Flat Plate Geometry

C. Lu, A. Cuadra, P. Kohut and **L.-Y. Cheng**
Brookhaven National Laboratory, P.O. Box 5000 Upton, NY 11973-5000 – USA

This study examined the validity of using flat fuel plates to model an MTR-type curved plate fuel assembly in neutronic calculations. Neutronics analysis of the MTR-type fuel assembly often utilizes equivalent flat-fuel-plate models to simplify input preparation, such as employing repeated structures in the lattice geometry, and to accelerate the numerical computation. In this study, we demonstrated the validity of this approach for the NBSR low-enriched uranium (LEU) fuel design as a case study. By leveraging the stochastic neutronic tool, Serpent 2, we built a single assembly model based on the curved plate geometry and demonstrated that the k_{eff} of an NBSR curved-fuel-plate model agreed with that of an equivalent flat-fuel-plate model within 20 pcm. All the Serpent 2 calculations were verified by replicating the same models in MCNP6.2. We also showed that this equivalence becomes less valid, i.e., the deviation in k_{eff} increases, when the fuel plate curvature increases.

10.5 Process Modeling of U-10Mo and U_3Si_2 Using Integrated Computational Materials Engineering

Vineet V. Joshi, Ayoub Soulami, Zhijie Xu, Lei Li, Kyoo-Sil Choi, Yucheng Fu, William E. Frazier, Elise Conte, Taylor Mason, Zach Huber, **Curt Lavender**

Pacific Northwest National Laboratory, 908 Battelle Blvd, Richland, WA 99354 – USA

Low-enriched uranium alloyed with 10 wt% molybdenum (U-10Mo) and U_3Si_2 have been identified as a promising alternative to highly enriched uranium as fuel for the United States high-performance research reactors. Manufacturing these fuel plates consists of multiple complex thermomechanical processes, which is highly challenging for computational modeling. The integrated computational materials engineering (ICME) concept supports building a microstructure-based framework to investigate the effect of manufacturing processes on the material microstructure. In this presentation an ICME model that combines several steps such as homogenization, hot rolling, annealing, and cold rolling is presented for modeling the manufacturing processes of the U-10Mo and U_3Si_2 fuel. The models have directly impacted the fabrication at BWXT and the have been used to optimize the rolling schedule, reduce time and cost of fabrication, eliminate fabrication defects, purchase of new equipment, and meeting the specifications.

SESSION 11 Licensing and Conversion Reactor Experience

Session Chair: Dennis Vinson

11.1 Physical Start-up of IVG.1M Reactor with Low-Enriched Uranium Fuel

E.G. Batyrbekov, V.S. Gnyrya, R.A. Irkimbekov, **I.V. Prozorova**, A.S. Azimkhanov, A.V. Timonov, Yu.A. Popov, I.K. Derbyshev, A.V. Vdovin, A.P. Maslov

Institute of Atomic Energy branch of National Nuclear Center, 2B Beibyt Atom Street, Kurchatov, 071100 - Republic of Kazakhstan

N.A. Hanan, P.L. Garner, J.G. Stevens

Argonne National Laboratory, 9700 South Cass Avenue, Lemont, Ill 60439 - USA

The first stage of the physical start-up of the IVG.1M reactor with low enrichment uranium (LEU) fuel was carried out on May 5, 2022. A physical start-up program was previously developed, which determines tasks, scope, conditions, experiment procedure and related technological operations, as well as safety requirements

Preparations for the first stage of the physical start-up consisted of the following:

- loading of the IVG.1M reactor core with standard LEU-fuel channels;
- installation of an experimental channel in the center of the reactor;
- loading of the main and additional sources of neutrons;
- filling reactor vessel with water;
- reaching a critical state.

The IVG.1M reactor was brought to a critical state by turning the control drums (CD) system in steps of no more than 100 pitches (0.3 βeff) with the construction of countdown curves. In a critical state, the position of the CD, the readings of the impulse and current channels were recorded. All operations were carried out in accordance with the provisions of the requirements of regulatory documents, using the necessary means of radiation support under the control of the dosimetry service.

Many physics measurements and system checks of the IVG.1M reactor are being performed in preparation for ascent to power.

11.2 Six-Year Experience of the WWR-K Reactor Operation with LEU Fuel

A.A. Shaimerdenov, Sh.Kh. Gizatulin, D.S. Sairanbayev, A.L. Martyshov and S.K. Sakhiyev
The Institute of Nuclear Physics, 1 Ibragimov st., 050032 Almaty – Kazakhstan

This paper describes a six-year experience in operating a WWR-K reactor with LEU fuel. The equipment that was upgraded after the conversion of the reactor to LEU fuel is described. It is shown that the increase in the experimental characteristics of the reactor after conversion to LEU fuel had a positive effect on expanding the scope of the reactor in the field of peaceful use of atomic energy. The accumulation of poison nuclei in a beryllium reflector is estimated. Comparative data on the spatial-energy distribution of neutrons in the core of the WWR-K reactor with HEU and LEU fuels are presented.

11.3 Practical Application of LEU Fuel for NIRR-1 Safe Operation

Kayode James Adedoyin
Nigerian Nuclear Regulatory Authority, Plot 564/565, Airport Road. Central Business District. Abuja – Nigeria

This paper contains details of safety analysis performed during the NIRR-1 core conversion to Low Enriched Uranium (LEU) fuel including the vessel integrity inspection, Sub criticality measurement of NIRR-1, Operational procedures of HEU core discharge and LEU loading. NIRR-1 is a 31 kW miniature neutron source reactor (MNSR) located at the Centre for Energy Research and Training (CERT), ABU, Zaria. The reactor went critical on February 3, 2004. NIRR-1 high-enriched uranium (HEU) core was replaced with LEU fuel consisting of uranium dioxide (UO2) with nominal enrichment of 13%. NNRA is a Regulatory body and responsible for Nuclear Safety and Radiological Protection for peaceful uses of nuclear Energy. Our analysis revealed that NIRR-1 operating with LEU fuel has Increase ratio of neutron flux with better shut down margin, suitable fuel integrity and melting temperature. Reactivity coefficients meet required limits, improved safety margin and heat transfer with increased length of reactor operation.

11.4 Utilization and Operation of the Dalat Nuclear Research Reactor after Full Core Conversion

Kien-Cuong NGUYEN, Nhi-Dien NGUYEN, Ton-Nghiem HUYNH, Quang-Huy PHAM, Minh-Tuan NGUYEN and Quoc-Duong TRAN
Dalat Nuclear Research Institute, Vietnam Atomic Energy Institute (VINATOM) 01 Nguyen Tu Luc, 66000 Dalat – Vietnam

The Dalat Nuclear Research Reactor (DNRR) has been completely converted to using low enriched uranium (LEU) fuel since 2012. As the demand of using radioisotope I-131 in hospitals grew, the DNRR modified the reactor core by inserting two additional irradiation channels in cells 5-6 and 9-6 for the purpose of enhancing I-131 isotope production. With accumulation neutron irradiation of TeO_2 targets from the specimen racks to the neutron trap or two new irradiation channels and operating the DNRR within 85 to 100 hours continuously per week, the total activity archived can exceed 100 Ci in each month. In addition, the fuel loading patterns for the next operation cycle were established in order to increase the operation time by approximately 15,000 hours by adding six fresh LEU fuel assemblies (FAs) to replace six beryllium rods located around neutron trap in three steps with each step adding 2 fresh FAs. It shown that the DNRR has been operated using LEU fuel safely and effectively.

11.5 Five Years of Operating GHARR-1 on LEU Fuel: Successes and Challenges

H. C. Odoi, K. Gyamfi, E. Shitsi, E. K. Boafo, W. Osei-Mensah, H. K. Obeng, P. Dordoh-Gasu and I. K. Baidoo

Ghana Atomic Energy Commission, Atomic Road – Kwabenya, Accra – Ghana

The conversion of the Ghana Research Reactor-1 (GHARR-1) commenced with feasibility studies involving all MNRS operating countries. This was via a Coordinated Research Project (CRP) organized by the IAEA and supported by the US DoE, ANL. The Highly Enriched Uranium (HEU) fuel was removed in August 2016, kept on site for about a year and then repatriated to China, the country of origin. The Low Enriched Uranium (LEU) fuel was fabricated in China, flown (sent) to Ghana in June 2017 and inserted into the reactor on July 12, 2017. The GHARR-1 has been operating on the 13.0 % enriched LEU Fuel for five years since its commissioning in August 2017 with optimum flux. The achievements and challenges so far are presented and discussed in this paper.