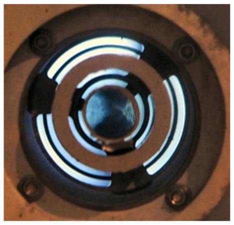
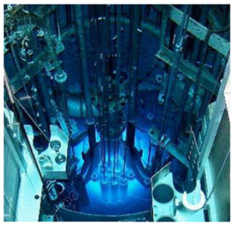
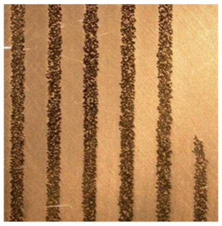
RERTR-2023

43rd International Meeting on Reduced Enrichment for Research and Test Reactors



November 5-8, 2023
Denver, Colorado, USA

www.rertr.anl.gov

Sunday, November 5

Registration: 4:30 – 6:00 PM, Embassy Suites by Hilton Denver Downtown Convention Center, Crystal Foyer, Third Floor

Welcome Reception: 6:00 – 8:00 PM, Crystal Ballroom

#	Session Title	Time	Paper Title	Presenter	Organization/ Country

Monday, November 6

Meeting Room: Crestone Ballroom, Third Floor

#	Session Title	Time	Paper Title	Presenter	Organization/ Country
	Welcome to Denver and RERTR-2023 International Meeting: Research Reactors & HALEU for a Brighter Future Chaired by Chris Landers (NNSA) and John Stevens (ANL)	9:00 am	Opening Remarks	Jeff Chamberlin	NNSA/US
		9:15 am	BR2 LEU Conversion Status: Initial Post-Irradiation Neutronics Analysis of COBRA-LEU-LTAs in BR2 Operational Cycles	Silva Kalcheva	SCK/Belgium
		9:35 am	Capabilities of the Converted KUCA LEU Cores for HALEU Criticality Studies	Yoshiyuki Takahashi	KUCA/Japan
		9:55 am	The New Multi-Purpose Research Reactor in Nigeria (MPRR): Alignment with the IAEA Milestones Approach and Integration of Proliferation Resistance Measures	Sunday Jonah	CERT/Nigeria
		10:15 am	Kindai University Reactor Role in Assuring the Future Nuclear Workforce	Genichiro Wakabayashi	Kindai/Japan
			10:35 AM Refreshment Break		
1	**HEU Minimization Programs** Chaired by Allison Johnston (NNSA)	11:00 am	The Conversion of FRM II to LEU – Status and Path Forward	Bruno Baumeister	TUM/ Germany
		11:20 am	Development of High-Density LEU Targets Based on Atomized Powder	Kinam Kim	KAERI/Korea
		11:40 am	Silo Storage for IVG.1M Irradiated Highly Enriched Uranium	Kris Gaines	ORNL (Remove)/US
		Noon	Fireside Chat with Chris Landers and Guests	Chris Landers et al	
			12:30 PM Lunch Break		
2	**Proliferation Resistance Now and for the Future** Chaired by Alex Meehan (NNSA)	2:00 pm	IAEA Support for Research Reactor Fuel Cycle Programmes	John Dewes	IAEA
		2:20 pm	INVAP Progress and Plans for the Next Generation of Research Reactors	Matias Marquez	INVAP/ Argentina
		2:40 pm	Flux Traps in Research Reactors: Implications for Neutron Flux and Conversion to Low-Enrichment Uranium Fuel	Hadi Abuzlf	Ben-Gurion Unive/Israel
		3:00 pm	TRIGA Reactor Digital Control and Monitoring for Operation and Operator Training	Alex Nellis	Penn State/USA
			3:05 PM Refreshment Break		
3	**Fuel Qualification and Irradiation Campaigns** Chaired by Abdellatif Yacout (ANL)	3:30 pm	An Update on the High Density Silicide Fuel Post Irradiation Examinations	Ann Leenaers	SCK/Belgium
		3:50 pm	Update on Recent Results from the Scanning Electron Microscopy Characterization of Fuel Plates Irradiated in EMPIrE	Dennis Keiser	INL/USA
		4:10 pm	FUTURE-MONO-1 Irradiation Test Status Update	Christian Schwarz	TUM/ Germany
		4:30 pm	Thermophysical Property Characterization of Irradiated U-10wt.%Mo Mini-plates from the MP-1 Experiment	Tsvetoslav Pavlov	INL/USA
		4:50 pm	Conversion of NNC reactors: Features of the Energy Start-up of the IVG.1M Reactor	Irina Prozorova	NNC/ Kazakhstan
			5:10 PM Adjourn		

Tuesday, November 7

Meeting Room: Crestone Ballroom, Third Floor

4	**Fuel Fabrication Updates** Chaired by Christian Schwarz (TUM)	8:30 am	From Whiteboards to Excel to SQL: Managing DRIP	Camilla McKinnon	BYU/USA
		8:35 am	New Manufacturing Process Line at Framatome CERCA's Facility for TRIGA Fuel Production	Jerome Allenou	CERCA/ France
		8:55 am	A Critical Review on the History of Fabricating Monolithic U-Mo Fuel Plates	Jason Schulthess	INL/USA
		9:15 am	Status of U-10Mo Fuel Fabrication at BWX Technologies	Robert Johnson	BWXT/USA
		9:35 am	USHPRR Fuel Fabrication Pillar: A Modeling Framework to Inform the Fabrication of U-10Mo Monolithic and U3Si2 Fuel	Ayoub Soulami	PNNL/USA
			9:55 AM Refreshment Break		
5	**Applications/Operations** Chaired by Chris Heysel (McMaster)	10:30 am	Installation of and Operational Experiences with the Penn State Breazeale Reactor's New Digital Control System	Ethan Kunz	Penn State/USA
		10:35 am	Special Hardware for Irradiation Testing of the Design Demonstration Elements	Greg Housley	INL/USA
		10:55 am	Capabilities of Advanced Irradiations and Fuel Testing at MARIA Reactor	Gaweł Madejowski	NCNR/Poland
		11:15 am	Twenty-Nine Years of Operating a Research Reactor: Evaluation of Ghana's MNSR	Henry Odoi	GAEC/Ghana
		11:35 am	Opportunities and New Challenges – FRM II Sailing in Troubled Waters	Axel Pichlmaier	TUM/ Germany
		11:55 am	Development of a Sandwich Formula Uncertainty Quantification Widget for MCNP	Caroline Sears	MIT/USA
			Noon Lunch Break		
6	**U.S. High Performance Reactor Conversions** Chaired by Dhongik Yoon (ANL)	1:30 pm	Massachusetts Institute of Technology Reactor (MITR) LEU Conversion Update	Lin-wen Hu	MIT/USA
		1:50 pm	University of Missouri Research Reactor (MURR) LEU Conversion Update	Maria Pinilla	MURR/USA
		2:10 pm	NBSR Update and Progress in Analytical Studies to Support the NBSR Conversion	Lap Cheng	BNL/USA
		2:30 pm	ATR Updates for Using Low Enriched Fuel	Collin Clark	INL/USA
		2:50 pm	High Flux Isotope Reactor Low-Enriched Uranium Conversion Activities – 2023 Status Update	Carol Sizemore	ORNL/USA
			3:10 PM — Refreshment Break		
7	**International Reactor Conversion Progress and Partnerships** Chaired by Caleb Braun (MURR)	3:30 pm	Validation and Comparison of 2D and 3D RANS Models for Flow through Curved Channels in MTR-type Nuclear Fuel Assemblies	Evan Bures	Texas A&M/ USA
		3:35 pm	Modeling Updates for System Transient Simulations of RHF Research Reactor Using RELAP5	Yeongshin Jeong	ANL/USA
		3:55 pm	Development of a Composite Involute Fuel Plate Model for Fluid-Structure Interaction Analyses	Marta Sitek	ANL/USA
		4:15 pm	Present Status and Future Plans of the Low-Enrichment Conversion Project at UTR-KINKI	Masato Tabuchi	Kindai/Japan
		4:35 pm	Conversion of NNC Reactors: Conversion of IGR Reactor	Ruslan Irkimbekov	NNC/ Kazakhstan
			4:55 PM - Break between Oral Sessions and Poster Session		

6:00 PM Poster Session and Reception Crystal Ballroom, Third Floor					
8	**Poster Session and Reception** Chaired by Caryn Warsaw (ANL)	6:00 – 7:30 pm	Building a Robust Fuel Assembly Measurement System	Daniel Fuentes Rodriguez	PNNL/USA
			TRIGA Reactor Digital Control and Monitoring for Operation and Operator Training	Alex Nellis	Penn State/USA
			Data Interpretation and Machine Learning for U-10Mo Non-Destructive Examination	Camilla McKinnon	PNNL/USA
			Development of a Sandwich Formula Uncertainty Quantification Widget for MCNP	Caroline Sears	MIT/USA
			From Whiteboards to Excel to SQL: Managing DRIP	Camilla McKinnon	PNNL/USA
			Installation of and Operational Experiences with the Penn State Breazeale Reactor's New Digital Control System	Ethan Kunz	Penn State/USA
			Validation and Comparison of 2D and 3D RANS Models for Flow through Curved Channels in MTR-type Nuclear Fuel Assemblies	Evan Bures	Texas A&M/USA
			Tritium Yield Measurements for Lithium Containing Salts with Nuclear Track Method	Andrew Maier	Ohio State/USA
			Innovative Surface Coating Technology for U-Mo Nuclear Fuel Performance under Neutron Irradiation	Dongjun Park	KAERI/Korea
			Flow Behavior in Involute-plate Research Reactor: RANS, LES and DNS Simulations	Jeremy Licht	ANL/USA
			Mechanical Properties of Irradiated U-10wt.%Mo Alloy Degraded by Porosity Development	Jeffrey Giglio	INL/USA
			Characterization of Zirconium Diffusion-Barrier Interlayer in Irradiated MP 1 Fuel Plates	Jan-Fong Jue	INL/USA
			MP-2 Design, Irradiation Status, and Expected Sample Availability	Margaret Marshall	INL/USA
			Porosity Observations with FUTURE-HFIR Silicide Plates	Theron Marshall	INL/USA
			COMSOL Results for the Proposed LEU Silicide Fuel Designs under 95 MW Nominal Conditions for HFIR Conversion	Prashant Jain	ORNL/USA
			Verification and Validation of RELAP5/MOD3.3 to Support the Research and Test Reactors Program	Dhongik Yoon	ANL/USA
			Modelling NBSR Water Gap Flow Distributions for DDE and LEU Element	Mauricio Tano	INL/USA
			CARTHAGE Flow Test Facility and DDE Modeling	Theron Marshall	INL/USA
			Flow Test of USHPRR LEU Fuel Elements for Hydraulic Performance Evaluation	Andrew Hebden	ANL/USA
			PARET/ANL 7.6 Modeling Approaches and Validations	Jeremy Licht	ANL/USA
			Safety Updates of NIRR-1 Oversight	Kayode Adedoyin	NNRA/Nigeria
			Economical Zirconium/Titanium Electroplating at Low-Throughput Commercial Scales	Vineet Joshi	PNNL/USA

Wednesday, November 8

Meeting Room: Crestone Ballroom, Third Floor

9	**Fuel Fabrication Challenges and Advances** Chaired by Bryon Curnutt (INL)	8:30 am	USHPRR Fuel Fabrication Pillar: Achievements and Advancements in Fabrication of U-10Mo Monolithic Fuel	Vineet Joshi	PNNL/USA
		8:50 am	Impact of the Fabrication Specification Tolerances of U-10Mo LEU Fuel on MITR and MURR	Valerio Mascolino	ANL/USA
		9:10 am	Uncertainty Quantification for the NBSR – Integrating Full-Core Sensitivity Coefficients and Nuclear Data Covariance Matrices	Lap Cheng	BNL/USA
9:30 AM Refreshment Break					
10	**Fuel Performance Analysis and Modeling** Chaired by Gawel Madejowski (NCNR)	9:50 am	Evaluation of Uranium Silicide (U3Si2) Fuel Using 80 MeV Xenon Ions at the ATLAS Material Irradiation Station	William Limestall	IIT/USA
		10:10 am	Validation of the Thermal Calculation in the DART Fuel Performance Code Through the Benchmark with the PLTEMP/ANL Code and the Comparison with Measured Oxide Layer Thickness Data	Bei Ye	ANL/USA
		10:30 am	Integrating a Mechanical Analysis Module into DART to Predict Local Swelling in Monolithic and Dispersion Fuel Plates	Yeon Soo Kim	ANL/USA
		10:50 am	Water Channel Thickness Estimation Through High Frequency Ultrasonic Device	Rhofrane Mrabti	Univ. of Montpellier/ France
		11:10 am	Irradiation Thermo-Mechanical Analysis of LEU Fuel for MURR Conversion	Firat Cetinbas / Dhongik Yoon	ANL/USA
11:30 AM Lunch Break					
11	**Design and Analysis Methods** Chaired by Dennis Vinson (SRNL)	1:00 pm	Preliminary Comparison of Depletion and Fuel Management Software MCODE and ADDER for the MIT Research Reactor	Maurane Garanzini	MIT/USA
		1:20 pm	Fluid Structure Interaction Modelling for MITR and NBSR DDEs	Mauricio Tano	INL/USA
		1:40 pm	Impacts of Irradiated MURR LEU Fuel Thermo-Mechanical Behavior on Thermal Hydraulics Safety Analysis	Dhongik Yoon	ANL/USA
		2:00 pm	System Code Modelling for NBSR DDE	Theron Marshall	INL/USA
2:30 PM Summary and Closure Jeff Chamberlin (DOE-NNSA) and John Stevens (ANL)					
3:00 PM Adjourn					

Welcome to Denver and RERTR-2023 International Meeting: Research Reactors & HALEU for a Brighter Future

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

BR2 LEU Conversion Status: Initial Post-Irradiation Neutronics Analysis of COBRA-LEU-LTAs in BR2 Operational Cycles

S. Kalcheva, J.M. Wight, G. Van den Branden, S. Van Dyck

SCK CEN, Boeretang 200, 2400 Mol – Belgium

This paper presents the recent key accomplishments of the BR2 LEU conversion project, named COBRA, which are the post-irradiation tests and neutronics analysis for confirming the irradiation conditions and neutronics parameters of three COBRA-LEU-LTAs. These LTAs were irradiated during several cycles in 2022 and 2023 and irradiations are still ongoing in current cycles. The purpose of the experiment is to confirm the fuel performance of high-density silicide LEU fuel plates in the BR2 fuel assembly geometry in support of fuel qualification for the LEU conversion of the BR2 reactor. Optimized core load configurations were designed and assembled for each cycle with the purpose to achieve target heat fluxes and burnups. Post irradiation neutronics calculations are performed by MCNP6.2 for several time steps during the cycle using a detailed MCNP geometry and burnup model. While the experiment is still ongoing, two of the LTAs have realized their end-of-life and the results presented demonstrate that they achieved their target conditions experiencing high peak heat fluxes (~ 480 W/cm-2) and very high average burnups (~ 60% U-235 burnup).

Capabilities of the Converted KUCA LEU Cores for HALEU Criticality Studies

Hironobu Unesaki, **Yoshiyuki Takahashi,** Tsuyoshi Misawa and Yasunori Kitamura

Institute for Integrated Radiation and Nuclear Science, Kyoto University, 1010, Asashiro-Nishi-2, Kumatori-cho, Sennan-gun, Osaka 590-0494 – Japan

Kyoto University Critical Assembly (KUCA) is a multi-core type critical assembly with two “Dry” cores using solid moderator materials and one “Wet” core using light water moderator. KUCA has been using 93% HEU since its startup in 1974; since 2018, the conversion project has been conducted. The LEU fuels to be used are U-Mo aluminum dispersion fuel for the Dry cores and U3Si2 aluminum dispersion fuel for the Wet core. After the conversion, KUCA would be the first critical reactor to be operated using solely U-Mo LEU fuel and also the only facility to be capable of performing systematic reactor physics experiments using 20% LEU. The anticipated extension of experimental performance after conversion in terms of neutron spectrum variety is expected to contribute as critical and subcritical experiment database for HALEU application and HPRR conversion. This paper describes the details of the KUCA conversion project and its expected contribution to nuclear science.

The New Multi-Purpose Research Reactor in Nigeria (MPRR): Alignment with the IAEA Milestones Approach and Integration of Proliferation Resistance Measures

Sunday A. Jonah

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

Stakeholders in the Nigeria Nuclear Sector are eagerly looking forward to the establishment of the proposed Multi-Purpose Research Reactor (MPRR) facility to complement the role currently being played by the Nigeria Research Reactor-1 (NIRR-1). MPRR will be the second Research Reactor in the country (NIRR-2) and is expected to be used for Radioisotope Production, Beam Physics Experiments

as well as in Education and Training towards the development of requisite Human Resources for the deployment of Nuclear Power plants in the country.

The MPRR Project from inception has been aligned with the IAEA Milestones Approach in order to derive the maximum benefits from the implementation through collaboration with International partners. In line with the IAEA milestones' approach, the Project is entering the Phase 2 stage, where the major activity is the bidding and selection of the reactor technology. Therefore, it is at this stage that the Proliferation Resistance (PR) measures can be incorporated into the reactor design and auxiliary equipment. Consequently, this paper discusses details of 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.

Kindai University Reactor Role in Assuring the Future Nuclear Workforce

Genichiro Wakabayashi

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

The Kindai University Reactor (UTR-KINKI) is a unique research reactor with a rated thermal power of 1 W, specially built for hands-on training in university nuclear engineering programs, and has been utilized in Japan's nuclear human resource development for more than sixty years since its first criticality in 1961. The reactor is currently the only place in Japan where students can practice using a real nuclear reactor, and is widely used not only for Kindai University students but by students from many other universities and technical colleges all over Japan including medical students. The reactor has also been utilized for outreach activities to secondary education and international workshops. As UTR-KINKI is the only reactor in Japan fueled with HEU, we plan to convert the reactor to LEU fuel, and the reactor will continue to play an irreplaceable role to assure the future nuclear workforce.

SESSION 1 HEU Minimization Programs

1.1 The Conversion of FRM II to LEU – Status and Path Forward

B. Baumeister, A. Pichlmaier

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

Technical University of Munich (TUM) is committed to convert its research neutron source Heinz Maier Leibnitz (FRM II) to a lower enriched fuel element. In spring 2023, the Bavarian and Federal German governments have ultimately decided to convert FRM II to a low-enriched uranium (LEU) fuel element using the most advanced fuel candidate, a monolithic U-10Mo alloy with a high uranium density of 15.5 gU/cm³. With this clear decision in hand, FRM II prepares detailed plans and starts to make arrangements to make this conversion a reality within the next decade. In this talk, we will present the current status and the path forward in the various areas of core analysis and optimization, fuel qualification, industrialization of the fabrication as well as the licensing process with the Bavarian and German authorities.

1.2 Development of High-Density LEU Targets Based on Atomized Powder

Kinam Kim, Taewon Cho, Sunghwan Kim, Dongjun Park and Yongjin Jeong

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

Tc-99m is the most widely used radiopharmaceutical isotope for medical diagnostic purposes. The key source for Tc-99m generation is Mo-99, produced from the nuclear fission of uranium in research reactors. Mo-99 producers have recently been attempting to replace conventional highly enriched uranium (HEU) targets with low enriched uranium (LEU) targets due to international non-proliferation policies. Consequently, there is a need to develop high-uranium-density targets using LEU to enhance the Mo-99 production efficiency of LEU targets. The Korea Atomic Energy Research Institute (KAERI) has successfully mass-produced U-Al powder for high-density targets through centrifugal atomization and fabricated high-density targets with uranium densities of 3.2 and 4.0 gU/cm^3. Atomization is a key technology for achieving high-uranium-density, as atomized powder can possess various U-Al compositions and a high uranium content. Four full-size target plates were irradiated in HANARO for one cycle (28 days). Post-irradiation examination (PIE) of the HANARO irradiation test was conducted to verify the safe in-pile performance of the high-density LEU dispersion target.

1.3 Silo Storage for IVG.1M Irradiated Highly Enriched Uranium

Kris Gaines

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

Ian Kapuza

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

Igor Bolshinsky, Stanley D. Moses

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

Erlan Batyrbekov, Viktor Baklanov, Vyacheslav Gnyrya, Alexandr Korovikov

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

The National Nuclear Center of the Republic of Kazakhstan's IVG.1M reactor was commissioned in 1975 and had a thermal neutron flux of 3.5×10^{14} $n/cm^{2}\cdot$ and peak power of 72 MW when operated. The original fuel assemblies contained 90 wt.% ^{235}U. The irradiated fuel assemblies were removed from the core in March 2022. In accordance with the U.S.-Russia government-to-government agreement, the irradiated HEU fuel from the IVG.1M reactor was supposed to be sent to Russia for processing and down-blending in 2023-2024. In February 2022, NNSA suspended all work for the IVG.1M fuel shipment due to Russia's expanded invasion of Ukraine. NNSA and NNC worked to evaluate different options for interim storage of the fuel until its final disposition could be determined.

Three dry storage technology options were considered based on their ability to meet all Kazakhstan regulatory and functional requirements within a reasonable cost and schedule: (1) vertical silo storage, (2) dual-use cask storage, and (3) a near-reactor storage well. A feasibility study was performed with each option evaluated by a committee of technical experts using criteria they had selected and prioritized. Vertical silo storage was determined to be the best option for the long-term temporary storage due to its enhanced security, lower cost, and ease of implementation and was presented to the Ministry of Energy of the Republic of Kazakhstan where it was approved in January 2023. NNSA is currently working with NNC to implement this storage option.

1.4 Fireside Chat with Chris Landers and Guests

SESSION 2 Proliferation Resistance Now and for the Future

2.1 IAEA Support for Research Reactor Fuel Cycle Programmes

John Dewes

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

2.2 INVAP Progress and Plans for the Next Generation of Research Reactors

M.G. Márquez

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

Throughout its 47 years of existence, INVAP has established itself as one of the main global players in the field of high performance LEU research reactors and associated facilities. INVAP's participation in different projects around the world and new opportunities, in addition to the constant assistance and aftersales services to existing research reactor fleet, defines the company as one of the cornerstones withstanding the efforts to reduce uranium enrichment and to increase the proliferation resistance.

This presentation describes the contribution of INVAP to enrichment reduction as a backbone of its business and summarizes the diverse experiences, challenges and milestones the company endeavored and presents the perspective for the future identifying the strategy for the ventures to come.

2.3 Flux Traps in Research Reactors: Implications for Neutron Flux and Conversion to Low-Enrichment Uranium Fuel

Hadi Abuzlf and Erez Gilad

The Unit of Nuclear Engineering, Ben-Gurion University of the Negev, Beer-Sheva 8410501 – Israel

Materials Testing Reactors are vital in diverse research and development applications. Flux traps, often designed using water gaps in the core, introduce safety concerns, generating highly localized power density increases that elevate fuel failure risks. This study develops a high-fidelity Monte Carlo model of an MTR core, including water gaps, fuel depletion, and fine-resolution spatial mesh, to explore local power peaking near flux traps. Simulation outcomes reveal highly localized thermal flux accumulation and increased power-peaking factors near the flux trap. A complementary experiment is designed and performed in the IRR-1 HEU MTR facility to measure the localized power peaking. The experimental measurements verify the simulation results. These findings hold critical implications for HEU and LEU-fueled MTR design and operation, warranting careful consideration of flux traps and advocating for refined reactor safety measures and fuel management strategies.

2.4 TRIGA Reactor Digital Control and Monitoring for Operation and Operator Training

J. A. Nellis

The Pennsylvania State University, 101 Breazeale Nuclear Reactor, University Park, PA 16802 – USA

At the Radiation Science and Engineering Center, recent completion and regulatory verification of a new digital control system and development of a high-fidelity console simulator, provide innovative avenues for safer and more efficient operation of the Mark III TRIGA reactor housed within the facility. The simulator, developed in Python, is a deployable application mimicking reactor operation yet requiring no reactor facilities or security controls. Utilizing intuitive computer interfacing and in-depth reactor behavior simulation, operators-in-training gain greater familiarity and skill in manipulating the reactor before their licensing by the Nuclear Regulatory Commission (NRC). For existing operators, the digital system enhances their ease of reactor control and, through the simulator, allows for effective practice, without requiring reactor usage and regulatory oversight. Moreover, this digital upgrade not only sets a precedent for digital control system integration in reactor design but also marks a significant departure from long-standing NRC policies of dependence on analog electronics' reliability.

SESSION 3 Fuel Qualification and Irradiation Campaigns

3.1 An Update on the High Density Silicide Fuel Post Irradiation Examinations

Ann Leenaers

SCK CEN, Boeretang 200, 2400 Mol – Belgium

3.2 Update on Recent Results from the Scanning Electron Microscopy Characterization of Fuel Plates Irradiated in EMPIrE

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

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

Bei Ye, Laura Jamison, and Gerard Hofman

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

Work has continued performing scanning electron microscopy (SEM) characterization of fuel plates irradiated in the European Mini-Plate Irradiation Experiment (EMPIrE). The characterization samples were transversely sliced from mini-plates (2.54 cm in width × 10.16 cm in length) that 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. The EMPIrE experiment tested the impact of different fuel fabrication variables on irradiation performance, which included: fuel powder heat treatment, coating methodology, coating thickness, fuel particle size distribution, Mo-alloying content, and fuel powder source. The SEM analysis was conducted in the context of improving understanding of the fuel plate swelling behavior, of determining the effectiveness of the ZrN coating in mitigating fuel/matrix interaction during irradiation, and of identifying the different features that comprise the irradiated fuel meat microstructure. The updated results of the SEM characterization are presented here.

3.3 FUTURE-MONO-1 Irradiation Test Status Update

C. Schwarz[1], B. Baumeister[1], T. Chemnitz[1], K. Buducan[1,2], B. Stepnik[2], J. Wight[3], S. Holmström[3]

[1]Forschungs-Neutronenquelle Heinz Maier-Leibnitz (FRM II), Technische Universität München Lichtenbergstr. 1, 85748 Garching b. München – Germany

[2]Framatome – Division CERCA, ZI les bérauds BP 1114, 26104 Romans-sur-Isere – France

[3]SCK CEN, Boeretang 200, 2400 Mol – Belgium

Mini-size fuel plates with monolithic U-10Mo and Zr coating applied by PVD were successfully irradiated and showed stable and predictable swelling behavior in the EMPIrE test. As a next step, the FUTURE-MONO-1 irradiation test is designed to demonstrate this fuel's performance in full-size geometry and thus qualify it generically for the conversion of European research reactors to low-enriched uranium fuel.

As part of this scale-up, Framatome-CERCA demonstrated their capabilities for U-Mo full-size foil fabrication by rolling and laser-cutting, while TUM's pilot PVD device is able to coat the full-size LEU foils produced this way with a Zr diffusion barrier. The subsequent C2TWP cladding process performed at CERCA provides the final fuel plates.

Six identical fuel plates will be fabricated this way, of which two get irradiated at the BR2 reactor to reach a burnup of 60-65% ^{235}U. After cooling, destructive and non-destructive examinations will be performed subsequently to investigate the behavior of the irradiated plates regarding local swelling, oxide formation as well as conditions of fuel meat, Zr coating, and AlFeNi cladding.

3.4 Thermophysical Property Characterization of Irradiated U-10wt.%Mo Mini-Plates from the MP-1 Experiment

Tsvetoslav R. Pavlov, Daniele Salavato, Ethan Hisle, Chuting Tan, Narayan Poudel, William A. Hanson, Jason L. Schulthess, Jeffrey J. Giglio, James I. Cole

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

Thermal diffusivity, specific heat and thermal conductivity of as fabricated and irradiated (with fission densities up to 6 $x10^{21}$ fissions cm^{-3}) U-10wt.%Mo fuel foils and miniplates were measured using the thermal conductivity microscope (TCM), differential scanning calorimeter (DSC) and laser flash analyzer (LFA) all of which are located in a shielded glove box – thermal property cell (TPC). An inverse method based on finite element analysis was developed to evaluate the thermal diffusivity of the 5-layered monolithic fuel mini-plates. The local thermal diffusivity and thermal conductivity measurements performed at room temperature do not show any spatial variation. Both the TCM and LFA measurements show thermal diffusivity and thermal conductivity to decrease with increasing fission density. Thermal conductivity is reduced at higher fission densities primarily due to the precipitation of thermally insulating fission gas bubbles.

3.5 Conversion of NNC Reactors: Features of the Energy Start-Up of the IVG.1M Reactor

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

Institute Atomic Energy, National Nuclear Center of the Republic of Kazakhstan, Beibit atom, 071110 – the Republic of Kazakhstan

NNC RK and ANL are jointly converting the IVG.1M reactor from highly enriched fuel to low enriched fuel.

The operability of the IVG.1M RR with the core equipped with the technological channels with LEU fuel with the parameters specified by the project has been confirmed.

On May 18, 2023, the act “On the results of the work of the commission of the RSE NNC RK to verify the readiness of the IVG.1M reactor for operation” was solemnly signed and the energy start-up of the reactor was carried out at a nominal power level of 10 MW, symbolizing the beginning of a new era for the IVG.1M reactor. After completion of the next stage – the energy start-up of the reactor and the experimental determination of all the necessary characteristics of the new core at power, the IVG.1M reactor was put into operation.

SESSION 4 Fuel Fabrication Updates

4.1 From Whiteboards to Excel to SQL: Managing DRIP

E.S. Sperry, **C.A. McKinnon**, D.A. Fuentes Rodriguez, G.E. Lavender, A. Bernat

Pacific Northwest National Laboratory, 902 Battelle Boulevard, Richland, WA 99352 – USA

In today’s data-centric landscape, effective data management and sharing is crucial. Data management involves organized collection, storage, and visualization for accessibility and precision, while efficient sharing ensures controlled dissemination. At the onset of the United States High Performance Research Reactor (USHPRRP) Project, rapid data influx into the Fuel Fabrication pillar came without any predetermined management strategy. The “data rich information poor (DRIP)” challenge was evident, as the data was severely fragmented, multimodal, and key relevant data was difficult to access.

Combining data cleaning, databases, and visualization efforts have significantly enhanced organization. This centralized system, backed by advanced SQL integration and a customized Tableau dashboard, combines measurements from all parts of the fabrication processes. With over 120,000 data links, the fundamental organization of information is represented through parent-child relationships. These links allow the connection of many data points in order to solidify the framework for heightened information sharing and analysis.

4.2 New Manufacturing Process Line at Framatome CERCA's Facility for TRIGA Fuel Production

J. Allenou, C. Blay, C. Chazalet-Montard, D. Geslin

Framatome CERCATM Division, 1 place Jean Millier, Tour Areva, 92400 Courbevoie – France

T. Veca

General Atomics, 3550 General Atomics Drive, San Diego, California, 92121 – USA

D. Morrell

Idaho National Laboratory, 2525 Fremont Avenue, Idaho Falls, ID 83402 – USA

TRIGA is a nuclear reactor type designed by General Atomics and used for research and training activities by scientific institutions and universities. The TRIGA reactor uses low-enriched, long-life Uranium and Zirconium Hydride alloy (UZrHx where x corresponds to the H/Zr ratio). The large prompt negative temperature coefficient of reactivity characteristics of the UZrHx fuel results in a safe design.

In 1995, Framatome CERCA was selected as preferred partner for General Atomics to manufacture TRIGA fuel. TRIGA International was created resulting of the joint venture between Framatome CERCA and General Atomics.

After a decade without any production, Framatome CERCA has restarted the production of TRIGA fuel element at the manufacturing facility located at Romans-sur-Isère in France. A full dedicated production line with new equipment, funded by DOE and TRIGA International, has been successfully installed and qualified. More than 30 standard fuel elements have been produced and delivered to the customer at the end of September 2023.

This paper will present the new TRIGA manufacturing process line including the results from the first production delivery of standard TRIGA fuel elements.

4.3 A Critical Review on the History of Fabricating Monolithic U-Mo Fuel Plates

J.L. Schulthess, W.J. Williams, G.A. Moore, J.F. Jue, R.B. Nielson

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

Since 2004, there has been extensive effort towards the development of a uranium molybdenum monolithic fuel system to convert high performance research and test reactors. The RERTR-6 experiment was the first to attempt a monolithic fuel instead of a dispersed fuel form. The fabrication methods evolved over time and provided the basis for current fabrication methods. The various steps in the process inevitably tailor the fuel alloy microstructure which is known to influence irradiation behavior. This document aims to present and discuss the fabrication evolution that transpired through a recounting of historical data from the various fabrication campaigns. By overlaying this data with basic science studies on the U-Mo that explored transformation kinetics, it is possible to estimate a measure of the impact heat treatments have on the final as-fabricated microstructure.

4.4 Status of U-10Mo Fuel Fabrication at BWX Technologies

R.E. Johnson, R.M. Mayfield, and G.M.P. Argon,

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

BWX Technologies, Nuclear Operations Group, Inc., in Lynchburg, Virginia, is currently supporting the DOE NNSA Materials Management and Minimization (M3) Conversion program, which aims to reduce or eliminate the use of high enriched uranium (HEU) dispersion fuels in high-powered research reactors in the United States by replacement with low enriched (LEU) alloy monolithic fuels. This presentation discusses the progress and optimizations that have been achieved in the fabrication of the monolithic U-10Mo fuel system. Results and observations from the foil, plate, and assembly fabrication processes will be presented. The improvements achieved via recent process modeling, optimization studies, and characterization will also be presented, as well as plans for future fabrication efforts.

4.5 USHPRR Fuel Fabrication Pillar: A Modeling Framework to Inform the Fabrication of U-10Mo Monolithic and U_3Si_2 Fuel

Ayoub Soulami, Vineet V. Joshi, Zach Huber, Michael Catalan, Mark Rossiter, Curt Lavender

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

Low-enriched uranium alloyed with 10 wt.% molybdenum (U-10Mo) has been identified as a promising alternative to highly enriched uranium as fuel for the United States high-performance research reactors. U_3Si_2 dispersion fuels to be used in the High Flux Isotope Reactor (HFIR) have complex fuel geometries and use a discrete burnable absorber layer creating additional challenges in Manufacturing. Fabricating these fuel plates consists of multiple complex thermomechanical processes, which lead to changes in the microstructure and can lead to defects. 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 we will discuss how series of models have been informing the fabrication process. The models have directly impacted the fabrication at BWXT and have been used to optimize the rolling schedule, reduce time and cost of fabrication, eliminate fabrication defects, and meeting the specifications.

SESSION 5 Applications/Operations

5.1 Installation of and Operational Experiences with the Penn State Breazeale Reactor's New Digital Control System

Ethan Kunz, Adams Tong, Jeffrey Geuther, Daniel Beck, Sean Herrmann, Gokhan Corak, and Kenan Unlu

The Pennsylvania State University, 101 Breazeale Nuclear Reactor, University Park, PA 16802 – USA

The Penn State Breazeale Reactor (PSBR) at the Radiation Science and Engineering Center (RSEC) installed a new digital control system in 2021. Previously, the system was a digital PROTROL system installed in 1991; this system was beginning to show obsolescence in its software, and preventive and corrective maintenance required access to parts no longer in manufacture. The control system and safety systems have digital replacements, but the new digital safety system (TRICONEX fault-tolerant processing system), set to replace the analog safety system currently in place, has not been implemented. The new control system was designed using industry standard Foxboro software and a redundant set of field control processors linked to dual workstations for operations and data analysis. The software consists of control blocks mimicking the logic of the legacy digital system. These blocks accept inputs from reactor instrumentation through the analog safety system, calculate parameters necessary for control of the reactor, and output these to safety and control logic blocks, as well as to the display workstations. The new design of the control system was made to be as similar as possible to its predecessor for two reasons: ease of implementation on the regulatory side and familiarity for long-standing operations staff. This approach allowed the facility to make the change to the new digital control system without a license amendment with 50.59 review. The digital safety system upgrade will require a license amendment and will be implemented in the near future.

The new digital control design has come with unique benefits and challenges. For instance, the engineering team has used the flexibility of the new system to install safety features such as interlocks not specifically required by technical specifications, as well as less critical applications such as operator aids. The Foxboro software has had issues with its internal clock, causing the data collection system to inaccurately calculate integrals such as thermal energy produced (MW-hr). This problem arose due to inaccuracies when summing very small numbers as required by the integration of power over time; the result was a non-uniform error of up to 20% in either direction. The error was caught by senior operators during a critical timed experiment and led to re-evaluation of the reactor's MW-hrs since the installation of the upgrade. The system has also experienced numerous "watchdog" scrams due to rod motor controller malfunctions tied to the design of the field bus modules used to send inputs and outputs to and from the rod drive motors. The engineering team is still in the process of troubleshooting this ongoing issue, but the suspected cause is signal paralysis caused by an overwhelming amount of traffic between the field bus modules and the motor controllers. Experiments using different signal packet forms will be conducted on identical devices in the RSEC electronics testing lab.

5.2 Special Hardware for Irradiation Testing of the Design Demonstration Elements

G.K. Housley, A. L. Crawford, B. Rossaert, T.D. Marshall, H.T. Hartman, I.Y. Glagolenko, T.D. Howell

Idaho National Laboratory, 2525 Fremont Avenue, Idaho Falls, ID 83402 – USA

Design Demonstration Elements (DDEs) are being designed for irradiation testing prior to conversion of High-Performance Research Reactors (HPRRs). Preceded by qualification testing of small mini plates (MPs) and large full-size plates (FSPs), the DDE irradiations will verify performance of the representative fuel elements. During irradiation, analytically predicted performance of the fuel will be verified using Ultrasonic (UT) tools capable of detecting fuel swelling, potential debonding, and deformations in certain fuel plates. Flow testing hardware will support testing of the non-fueled (dummy) DDEs in prototypic flow conditions prior to irradiation. The DDEs will be irradiated in selected test reactors using special irradiation vehicles. These vehicles will control the flow through the DDEs and help to replicate, as much as possible, the prototypic power density distribution across the DDE fuel plates when irradiated in an approved test reactor. Some vehicles are instrumented for on-line monitoring of thermal-hydraulic conditions during irradiation.

5.3 Capabilities of Advanced Irradiations and Fuel Testing at MARIA Reactor

G. Madejowski

Nuclear Energy Division, National Centre for Nuclear Research, Soltana 7, 05-400 Otwock, – Poland

MARIA is the high-flux material testing research reactor located in Poland. Its design derived from a multi-loop reactor, allowing for setting multiple testing loops in the core. Using beryllium as a moderator allowed for high fuel pitch, providing a large capacity for target materials and irradiation rig. The modular construction of the core allows for adjusting the neutron spectrum for target materials, whereas open pool structure eases measurements, heating the targets etc. MARIA's primary cooling system consist of the array of channels containing fuel elements. Each element has an individual cooling loop, which allow for the measurement of the discharge, water temperature rise as well as for sampling the water and directing it from each loop to the Fuel Leakage Detection System. It detects fission products in the water by detecting delayed neutrons. The system has proved to be effective in incident detection and now it is undergoing a comprehensive modernization. Aforementioned capabilities along with proved technology to measure neutron flux density distribution inside the fuel element makes MARIA reactor suitable for fuel testing. This study summarizes irradiation capabilities and fuel testing prospects in MARIA reactor.

5.4 Twenty-Nine Years of Operating a Research Reactor: Evaluation of Ghana's MNSR

H.C. Odoi, K. Gyamfi, W. Osei-Mensah and P. Dordoh-Gasu

National Nuclear Research Institute, Ghana Atomic Energy Commission, Atomic Road, Kwabenya Accra – Ghana

The Ghana Research Reactor-1 (GHARR-1), which is one of the Chinese Miniature Neutron Source Reactors (MNSRs), has been in operation since it was installed in December 1994. The GHARR-1 initially used a 90.2 % Highly Enriched Uranium till the fuel was converted to a 13.0 % Low Enriched Uranium in 2017. The reactor has been used for a number of programmes including Research and Development, Training of Students from the various tertiary institution in Ghana, and more importantly development of Human Resource for Ghana's Nuclear Power Programme. The GHARR-1 and its associated facilities (the International MNSR Training Centre in particular) is being utilized as a training facility for the removal of HEU fuel for MNSR Countries with support from the United States Department of Energy (US DoE). The paper evaluates the utilization of GHARR-1 in the aforementioned programs and the likes since the reactor was first commissioned.

5.5 Opportunities and New Challenges – FRM II Sailing in Troubled Waters

A. Pichlmaier, F. Jeschke, F. Schätzlein

Technical University of Munich – FRM II, Lichtenbergstr. 1, 85748 Garching – Germany

The Technical University of Munich (TUM) operates the FRM II as a world class multi-purpose research reactor. While formally the German nuclear phase-out only affects commercial production of electricity, the remaining German research reactors are also significantly affected: essential industry reduces or even ceases their nuclear activities while from the side of the federal regulatory body and its TSO the focus shifts more and more towards research reactors that find it harder and harder to meet all the requirements that get ever closer to those originally meant to be applied to nuclear power plants. From the operators' point of view, this jeopardizes the future of the German research reactor fleet. At the same time, there is no questioning of the science case and the benefits for society that arise from operation of a research reactor and the state-and university backing is still strong. Our presentation will give an analysis of the situation as well as a personal biased outlook based on the experience gained at FRM II.

5.6 Development of a Sandwich Formula Uncertainty Quantification Widget for MCNP

C.J. Sears[1], A. Cuadra[2], M. Kinsinger[3]

[1]Massachusetts Institute of Technology Nuclear Reactor Laboratory, 138 Albany Street, Cambridge, MA 02139 – USA
[2]Brookhaven National Laboratory, P.O Box 5000 Upton, NY 11973 – USA
[3]Smith College, 10 Elm Street, Northampton, MA 01063 – USA

The sandwich formula code package is a UQ tool that pairs with MCNP. The sandwich formula states that the uncertainty on keff can be found by multiplying nuclear data covariance matrices with k-eigenvalue sensitivity vectors. The code package uses NJOY2016 to produce covariance matrices from ENDF/B-VIII.0 and ENDF/B-VII.1 evaluations. It also includes a Python-based widget that processes NJOY and MCNP outputs to supply data to the Python sandwich formula implementation. This tool was benchmarked against TSUNAMI and found to be in good agreement. For an NBSR LEU fresh-fuel, single element model test case, the widget estimated a total uncertainty of 1.11% where keff=1.25818. This code is intended to illuminate how nuclear data covariances propagate to uncertainty on keff, and to quantify the extent to which individual isotopes bias MCNP calculations. This method can be applied to the LEU conversion study of other High-Performance Research Reactors.

SESSION 6 U.S. High Performance Reactor Conversions

6.1 Massachusetts Institute of Technology Reactor (MITR) LEU Conversion Update

L. Hu and M. Garanzini

Massachusetts Institute of Technology Nuclear Reactor Laboratory, 138 Albany Street, Cambridge, MA 02139 – USA

K. Anderson, V. Mascolino, S. Yang, C. Bojanowski, G. Wang, A. Hebden, W. Mohamed, D. Yoon, D. Jaluvka, E. Wilson

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

The Massachusetts Institute of Technology Reactor (MITR) is a 6 MW research reactor currently operating with highly enriched uranium (HEU) finned plate-type fuel. MITR is planning to undergo conversion to a low-enriched uranium (LEU) high-density uranium-10 wt.% molybdenum (U-10Mo) unfinned plate-type fuel. Conversion activities have been ongoing, and the current LEU fuel element design and safety analyses are technically mature. The MITR LEU core is planned to operate at 7 MW and have higher primary and secondary coolant flow rates. Over the past year, facility upgrades have been started to prepare for conversion, including upgrading the primary and secondary coolant systems and fabricating new fuel storage racks. Concurrently, a variety of engineering activities have been conducted by MIT and Argonne National Laboratory staff in support of conversion. These include: improving the LEU fuel specifications by analyzing the impact of fabrication tolerances on reactor performance and safety; completing a preliminary design for flow testing of the LEU fuel element; evaluating the use of a reactor simulator digital twin for operator training; assessing the feasibility of adopting a new fuel management software at MIT; and assisting in the design and planning of a full element irradiation test, among other activities.

6.2 University of Missouri Research Reactor (MURR) LEU Conversion Update

M. Pinilla, R. Astrino, C. Braun, K. Kutikkad

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

C. Bojanowski, F. Cetinbas, V. Mascolino, G. Wang, D. Yoon, J. Stillman, D. Jaluvka, A. Hebden, W. Mohamed, E. Wilson

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

The University of Missouri Research Reactor (MURR) is a 10 MW research reactor currently operating with highly enriched uranium (HEU) curved plate-type fuel. MURR is planning to undergo conversion to a low-enriched uranium (LEU) high-density uranium-10 wt.% molybdenum (U-10Mo) curved plate-type fuel. The MURR LEU core will be uprated to a power of 12 MW that meets all safety requirements and ensures an equivalent experimental performance as the current HEU core. Conversion activities have been ongoing, and the current LEU fuel element has completed preliminary design and preliminary safety analysis report documentation. A variety of engineering activities that have been conducted by MURR and Argonne National Laboratory staff in support of conversion are reported. These include, among others, analyzing the impact of the LEU fuel specification fabrication tolerances on reactor performance and safety; LEU fuel element preliminary flow test design; plate- and element-level thermo-mechanical (TM) and fluid-structural interaction (FSI) analyses of the LEU fuel element; and evaluation of the impact of the TM and FSI modeling on the thermal hydraulic (TH) safety margins.

6.3 NBSR Update and Progress in Analytical Studies to Support the NBSR Conversion

L.-Y. Cheng, A. Varuttamaseni, C. Lu, C. Sears, A. Cuadra, P. Kohut

Brookhaven National Laboratory, P.O Box 5000 Upton, NY 11973 – USA

A. Weiss, D. Sahin, T. Newton

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

Preparations are well underway at the NBSR to install a liquid deuterium cold source to mitigate the effects of thermal neutron flux loss due to conversion of the NBSR to LEU fuel. In addition, a pre-conceptual design study is in preparation for a reactor using U-Mo LEU fuel to replace the NBSR.

Two separate studies were conducted for the conversion of the NBSR focusing on the impact of fuel specification tolerances on thermal limits, and the Keff. A related study evaluated the uncertainty in the K_{eff} as a result of nuclear data uncertainty. Ten test matrix branch cases were considered to analyze the impact of the fabrication uncertainties (fuel plate dimension and isotopic composition) on the Keff. The impact on the two thermal limits, the critical heat flux ratio and the onset of flow instability ratio, were evaluated statistically by randomly combining variations in channel dimension and the fuel loading.

6.4 ATR Updates for Using Low Enriched Fuel

E.C. Woolstenhulme and **C.M. Clark**

Idaho National Laboratory, 1955 Fremont Avenue, Idaho Falls, ID 83402 – USA

The Advanced Test Reactor at Idaho National Laboratory has made significant progress towards inserting lead test assemblies of low enriched uranium fuel. The fuel test campaigns have been redeveloped to be more practical and efficient. This will reduce the burden on fuel fabrication efforts. The safety bases for regulatory approval of lead test assembly insertion are in a mature state. Specifically, the ATR and ATRC safety analysis report addenda should be submitted for regulatory review soon. Flux trap modifications are also being evaluated for low enriched fuel. The purpose of flux trap modifications is to ensure that the irradiation conditions needed by users can be attained.

6.5 High Flux Isotope Reactor Low-Enriched Uranium Conversion Activities – 2023 Status Update

C.W. Sizemore, Z.A. Bacon, J.W. Bae, K. Borowiec, K.M. Burg1, D. Chandler2, V. D. Fudurich, D. Hartanto, C.J. Hurt, P.K. Jain, W.C. Lowe, E. Popov, T.B. Smith

Oak Ridge National Laboratory, 1 Bethel Valley Road, Oak Ridge, TN 37830 – 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, materials irradiation, and neutron activation analysis. 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) designs. 2022-2023 progress includes developing Safety Design Strategy and performing preliminary transient analysis for the optimized High Density (5.3 gU/cc) and Low Density (4.8 gU/cc) uranium silicide fuel designs. Reactor physics progress includes developing a validation roadmap, performing sensitivity analysis and uncertainty studies, evaluating fuel inhomogeneity impacts, and executing experiment performance comparisons for materials research and key isotope production, including Cf-252. Thermal-hydraulic progress includes direct numerical simulation (DNS) of a HFIR involute channel and multi-physics modeling to benchmark Cheverton-Kelley and Gambill-Bundy experiments.

SESSION 7 International Reactor Conversion Progress and Partnerships

7.1 Validation and Comparison of 2D and 3D RANS Models for Flow through Curved Channels in MTR-type Nuclear Fuel Assemblies

E. Bures[1,2,] A.G. Weiss[2], M.L. Kimber[1]

[1]Texas A&M University, 429 Spence Street, College Station, TX 77843 – USA
[2]NIST Center for Neutron Research, 100 Bureau Drive, Gaithersburg, MD 20899 – USA

This work assesses the feasibility of using 2D RANS models to ascertain velocity profiles within the curved MTR-type fuel plate channels. The analysis results will also be used to support the pre-conceptual design activities of the National Institute of Standards and Technology (NIST) replacement reactor, namely the NIST Neutron Source (NNS). For the assessments, simulations were conducted using OpenFOAM v2206 to compare 2D and 3D RANS models of fuel channels. A Grid Convergence Index was computed for both models, revealing an average numerical error of less than 4% for each of the models. The SIMPLE algorithm was utilized, employing the Spalart-Almaras, k - ϵ Realizable, and k - ϵ standard models. Consistent convergence was observed across all models without discernible outliers. Centerline velocities in the 3D model were approximately 5% higher than measurements from an experimental mock-up of the MTR-type fuel in literature. The 2D models consistently overestimated the centerline velocity within 10 % to 11%. The 3D model aligned closer with the near-wall data from the experiment, whereas the 2D model showed notable deviations. The k - ϵ Realizable model matched best with experimental data, especially near the walls. It was found that 2D models can serve as a viable option for estimating centerline velocity in the MTR-type fuel plate channels via an adjustment factor of about 10.5 %, which is generic to any MTR-type channel.

7.2 Modeling Updates for System Transient Simulations of RHF Research Reactor Using RELAP5

Y. Jeong[1], A. Bergeron[1], F. Thomas[2], Y. Calzavara[2], J. R. Licht[1]

[1]Argonne National Laboratory, 9700 S. Cass Avenue, Lemont, IL 60439 – USA
[2]Institut Laue-Langevin, 71 Av. Des Martyrs, Grenoble, 38042 – France

Argonne National Laboratory (ANL) collaborates with the Institut Laue-Langevin (ILL) to support conversion of the Réacteur à Haut Flux (RHF) reactor in Grenoble, France from Highly Enriched Uranium (HEU) to Low-Enriched Uranium (LEU) fuel. The RELAP5/Mod 3.3 code is being used by ANL to perform RHF reactor transient analyses. The RELAP5 model, first built to perform a verification of the RHF system transients previously simulated with the CATHARE code, is being improved to better simulate the key RHF reactor transient scenarios. The major updates include, for example, the replacement of a pressurization circuit model with full piping and a pressurization pump model with recalibration of the steady state conditions. The presentation will describe the current RELAP5 model of RHF, including the recent modeling improvements, along with plans for transient analyses that will be performed by ANL, both for HEU and LEU designs, as an independent verification to analyses being performed with the CATHARE code.

7.3 Development of a Composite Involute Fuel Plate Model for Fluid-Structure Interaction Analyses

M. Sitek, A. Bergeron, J. Licht

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

Conversion from HEU to LEU may result in changes in the fuel plate design. So far, computational fluid-structure interaction and thermal expansion sensitivity analyses were performed on a single involute plate model. In this phase of the study, the model has been extended by (1) explicit representation of

the fuel plate components i.e., core and cladding, (2) implementation of temperature-dependent material properties of the fluid and solids, (3) addition of uniform and nonuniform distributed heat source in the fuel, and (4) variation of the constraints between fixed to deformable side-plates. The paper will cover the key findings of the structural analysis of the improved involute plate model with the goal of presenting a more comprehensive overview of the LEU plate response to anticipated loadings. This study helps verifying the soundness of the involute LEU plate design.

7.4 Present Status and Future Plans of the Low-Enrichment Conversion Project at UTR-KINKI

M. Tabuchi, G. Wakabayashi, N. Sugiura, H. Shiga, S. Hohara, and H. Yamanishi

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

C. H. Pyeon

Institute for Integrated Radiation and Nuclear Science, Kyoto University, 2-1010, Asashiro-nishi, Kumatori-cho, Sennan-gun, 590-0494 Osaka – Japan

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

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

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 trainings 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. From this background, we have initiated technical studies to realize a rapid conversion to LEU. At present, we have established a calculational model that reproduces the geometry of the core based on the results of previous inspections and correctly calculates measurements made on the HEU cores. For the next step, the details of the LEU fuel design will be completed and confirmed with calculational results for the LEU fuel plate design loaded into the core.

7.5 Conversion of NNC Reactors: Conversion of IGR Reactor

Gnyrya V.S., **Irkimbekov R.A**, Surayev A.S., Prozorova I.V

Institute Atomic Energy, National Nuclear Center of the Republic of Kazakhstan, Beibit atom, 071110 – The Republic of Kazakhstan

In 2023, the National Nuclear Center of the Republic of Kazakhstan successfully completed the conversion of the IVG.1M reactor. Many organizational and engineering issues related to the reactor conversion have been solved, and extensive experience in implementing various activities was accumulated.

Currently, NNC is at the stage of a renewed feasibility study of the IGR reactor conversion. The first attempt to justify the reactor conversion was deemed impossible when the LEU uranium-graphite sample fuel produced was not mechanically strong enough.

In recent years, computational and experimental studies of the IGR reactor have been conducted. Reactor performance parameters were measured in modes that approached the threshold operating parameters of the IGR reactor. The conditions for the possibility of reactor conversion were thoroughly investigated. The main parameters now include the parameters of the neutron field in the empty central experimental channel (CEC) uncompensated channel and the parameters for testing assemblies containing nuclear fuel in the CEC. Enhanced modeling of the complex changes in the irradiation parameters of various experimental devices during heating of the reactor core have been developed and applied. The current foci of work are on alternative uranium-graphite fuel supply and optimization of the core for the LEU fuel being pursued.

SESSION 8 Poster Session

8.1 Building a Robust Fuel Assembly Measurement System

E.S. Sperry, C.J. Lansing, **D.A. Fuentes Rodriguez**

Pacific Northwest National Laboratory, 902 Battelle Boulevard, Richland, WA 99352 – USA

The dimension properties of fuel foils, plates, and elements are vital to the continued conversion of reactors to low-enriched uranium (LEU). Precise measurements must be obtained to ensure that fuel elements meet specifications. Currently, fuel assemblies are analyzed with Quality Assured systems. While accurate, these systems don't provide data in a timely fashion.

In this project, we test the use of thin film capacitive sensors, 3D measurement tools, lasers, and blue-structured light systems in the measurement of components. This technology should not only run faster, but also provide highly detailed data. By introducing these testing methods, we aim to identify potential errors during production and remedy them before wasting essential resources. These standalone systems should also provide in-process measurements to the Reactor Conversion pillar to refine models based on actual fabrication tolerances.

8.2 TRIGA Reactor Digital Control and Monitoring for Operation and Operator Training

J. A. Nellis

The Pennsylvania State University, 101 Breazeale Nuclear Reactor, University Park, PA 16802 – USA

At the Radiation Science and Engineering Center, recent completion and regulatory verification of a new digital control system and development of a high-fidelity console simulator, provide innovative avenues for safer and more efficient operation of the Mark III TRIGA reactor housed within the facility. The simulator, developed in Python, is a deployable application mimicking reactor operation yet requiring no reactor facilities or security controls. Utilizing intuitive computer interfacing and in-depth reactor behavior simulation, operators-in-training gain greater familiarity and skill in manipulating the reactor before their licensing by the Nuclear Regulatory Commission (NRC). For existing operators, the digital system enhances their ease of reactor control and, through the simulator, allows for effective practice, without requiring reactor usage and regulatory oversight. Moreover, this digital upgrade not only sets a precedent for digital control system integration in reactor design but also marks a significant departure from long-standing NRC policies of dependence on analog electronics' reliability.

8.3 Data Interpretation and Machine Learning for U-10Mo Non-Destructive Examination

C.A. McKinnon, E.S. Sperry, Jarrod Ver Steeg, Stacy Irwin

Pacific Northwest National Laboratory, 902 Battelle Boulevard, Richland, WA 99352 – USA

Low enriched uranium (LEU) alloyed with 10 wt.% molybdenum (U-10Mo) is a promising alternative fuel system for domestic high-performance research reactors that cannot use currently available LEU fuel systems. The monolithic U-10Mo fuel system comprises an aluminum-clad zirconium interlayer foil. Experimental fabrication involves thorough inspections of the cast ingots and fully processed fuel plates using ultrasonic (UT) and radiographic (RT) testing techniques.

Presently, 200+ plates have been visualized with UT and RT. The UT scan results are consolidated into an interactive viewer program and allows different inspections to occur within the same data set. RT images are analyzed for anomalies in fuel density and location. The interpretation of RT and UT data requires the expertise of highly trained inspectors. Future efforts to evaluate RT, UT and gamma scanning coupled with machine learning algorithms will improve inspectors' ability to interpret data about fuel zone location, homogeneity, and cladding condition.

8.4 Development of a Sandwich Formula Uncertainty Quantification Widget for MCNP

C.J. Sears[1], A. Cuadra[2], M. Kinsinger[3]

[1]Massachusetts Institute of Technology Nuclear Reactor Laboratory, 138 Albany Street, Cambridge, MA 02139 – USA
[2]Brookhaven National Laboratory, P.O Box 5000 Upton, NY 11973 – USA
[3]Smith College, 10 Elm Street, Northampton, MA 01063 – USA

The sandwich formula code package is a UQ tool that pairs with MCNP. The sandwich formula states that the uncertainty on k_{eff} can be found by multiplying nuclear data covariance matrices with k-eigenvalue sensitivity vectors. The code package uses NJOY2016 to produce covariance matrices from ENDF/B-VIII.0 and ENDF/B-VII.1 evaluations. It also includes a Python-based widget that processes NJOY and MCNP outputs to supply data to the Python sandwich formula implementation. This tool was benchmarked against TSUNAMI and found to be in good agreement. For an NBSR LEU fresh-fuel, single element model test case, the widget estimated a total uncertainty of 1.11% where k_{eff}=1.25818. This code is intended to illuminate how nuclear data covariances propagate to uncertainty on k_{eff}, and to quantify the extent to which individual isotopes bias MCNP calculations. This method can be applied to the LEU conversion study of other High-Performance Research Reactors.

8.5 From Whiteboards to Excel to SQL: Managing DRIP

E.S. Sperry, **C.A. McKinnon**, D.A. Fuentes Rodriguez, G.E. Lavender, A. Bernat

Pacific Northwest National Laboratory, 902 Battelle Boulevard, Richland, WA 99352 – USA

In today's data-centric landscape, effective data management and sharing is crucial. Data management involves organized collection, storage, and visualization for accessibility and precision, while efficient sharing ensures controlled dissemination. At the onset of the United States High Performance Research Reactor (USHPRRP) Project, rapid data influx into the Fuel Fabrication pillar came without any predetermined management strategy. The "data rich information poor (DRIP)" challenge was evident, as the data was severely fragmented, multimodal, and key relevant data was difficult to access.

Combining data cleaning, databases, and visualization efforts have significantly enhanced organization. This centralized system, backed by advanced SQL integration and a customized Tableau dashboard, combines measurements from all parts of the fabrication processes. With over 120,000 data links, the fundamental organization of information is represented through parent-child relationships. These links allow the connection of many data points in order to solidify the framework for heightened information sharing and analysis.

8.6 Installation of and Operational Experiences with the Penn State Breazeale Reactor's New Digital Control System

Ethan Kunz, Adams Tong, Jeffrey Geuther, Daniel Beck, Sean Herrmann, Gokhan Corak, and Kenan Unlu

The Pennsylvania State University, 101 Breazeale Reactor, University Park, PA 16802 – USA

The Penn State Breazeale Reactor (PSBR) at the Radiation Science and Engineering Center (RSEC) installed a new digital control system in 2021. Previously, the system was a digital PROTROL system installed in 1991; this system was beginning to show obsolescence in its software, and preventive and corrective maintenance required access to parts no longer in manufacture. The control system and safety systems have digital replacements, but the new digital safety system (TRICONEX fault-tolerant processing system), set to replace the analog safety system currently in place, has not been implemented. The new control system was designed using industry standard Foxboro software and a redundant set of field control processors linked to dual workstations for operations and data analysis. The software consists of control blocks mimicking the logic of the legacy digital system. These blocks accept inputs from reactor instrumentation through the analog safety system, calculate parameters necessary for control of the reactor, and output these to safety and control logic blocks, as well as to the display workstations. The new design of the control system was made to be as similar as possible to its predecessor for two reasons: ease of implementation on the regulatory side and familiarity for long-standing operations staff. This approach allowed the facility to make the change to the new digital control system without a license amendment with 50.59 review. The digital safety system upgrade will require a license amendment and will be implemented in the near future.

The new digital control design has come with unique benefits and challenges. For instance, the engineering team has used the flexibility of the new system to install safety features such as interlocks not specifically required by technical specifications, as well as less critical applications such as operator aids. The Foxboro software has had issues with its internal clock, causing the data collection system to inaccurately calculate integrals such as thermal energy produced (MW-hr). This problem arose due to inaccuracies when summing very small numbers as required by the integration of power over time; the result was a non-uniform error of up to 20% in either direction. The error was caught by senior operators during a critical timed experiment and led to re-evaluation of the reactor's MW-hrs since the installation of the upgrade. The system has also experienced numerous "watchdog" scrams due to rod motor controller malfunctions tied to the design of the field bus modules used to send inputs and outputs to and from the rod drive motors. The engineering team is still in the process of troubleshooting this ongoing issue, but the suspected cause is signal paralysis caused by an overwhelming amount of traffic between the field bus modules and the motor controllers. Experiments using different signal packet forms will be conducted on identical devices in the RSEC electronics testing lab.

8.7 Validation and Comparison of 2D and 3D RANS Models for Flow through Curved Channels in MTR-type Nuclear Fuel Assemblies

E. Bures[1,2,] A.G. Weiss[2], M.L. Kimber[1]

[1]Texas A&M University, 429 Spence Street, College Station, TX 77843 – USA
[2]NIST Center for Neutron Research, 100 Bureau Drive, Gaithersburg, MD 20899 – USA

This work assesses the feasibility of using 2D RANS models to ascertain velocity profiles within the curved MTR-type fuel plate channels. The analysis results will also be used to support the pre-conceptual design activities of the National Institute of Standards and Technology (NIST) replacement reactor, namely the NIST Neutron Source (NNS). For the assessments, simulations were conducted using OpenFOAM v2206 to compare 2D and 3D RANS models of fuel channels. A Grid Convergence Index was computed for both models, revealing an average numerical error of less than 4% for each of the models. The SIMPLE algorithm was utilized, employing the Spalart-Almaras, k - ϵ Realizable, and k - ϵ standard models. Consistent convergence was observed across all models without discernible outliers. Centerline velocities in the 3D model were approximately 5% higher than measurements from an experimental mock-up of the MTR-type fuel in literature. The 2D models consistently overestimated the centerline velocity within 10 % to 11%. The 3D model aligned closer with the near-wall data from the experiment, whereas the 2D model showed notable deviations. The k - ϵ Realizable model matched best with experimental data, especially near the walls. It was found that 2D models can serve as a viable option for estimating centerline velocity in the MTR-type fuel plate channels via an adjustment factor of about 10.5 %, which is generic to any MTR-type channel.

8.8 Tritium Yield Measurements for Lithium Containing Salts with Nuclear Track Method

Andrew Maier, Praneeth Kandlakunta, Lei R. Cao

The Ohio State University, 281 W. Lane Avenue, Columbus, OH 43210 – USA

Methods of triton detection from lithium-6 enriched salts utilizing CR-39 solid state nuclear track detectors are being investigated with neutrons as the interrogation particles. The unwanted tritium production from the trace lithium-6 present in molten salt fuels requires processing to minimize the lithium-6 isotopic fraction. On the other hand, high lithium-6 isotope fraction salts may be used as a mean for high purity tritium gas production. A lithium compound enriched in lithium-6 can be irradiated in-core or ex-core at a research reactor to harvest the triton particles, leading to a pure tritium gas accumulation, that is sufficient to be used to fuel a portable DT neuron generator. One critical aspect of the method is to determine the lithium-6 neutron capture rate, i.e., the tritium gas production rate, using the nuclear track method. In this study, CR-39 is utilized to capture triton particles, which can then be read out by an optical microscope and evaluated either directly by a human reader or by a deep-learning image processing model. This measurement was conducted using an external neutron beam, which provides a high detection efficiency if tritons are emitted into CR-39 which is then properly etched. Furthermore, the solid angle of detection can be nearly 2-π if one CR-39 is applied, or 4-π if two CR-39 are applied. Finally, this method provides a basis for passive neutron imaging with lithium compound (Lif:ZnS) and CR-39 films.

8.9 Innovative Surface Coating Technology for U-Mo Nuclear Fuel Performance Under Neutron Irradiation

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

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

To enhanced U-Mo nuclear fuel performance, we employ a novel surface coating technique to counteract adverse reactions with the Al matrix. Previous attempts to enhance fuel by adding Si into the Al matrix showed limited success, resulting in rapid reaction layer formation and potential fuel plate integrity issues. Microstructural analysis of as-fabricated U-Mo fuel revealed incomplete Si diffusion into U-Mo particles. To address these issues, we developed a coating process using Physical Vapor Deposition, applying a Mo layer to U-Mo powder. Plate-type nuclear fuel for research reactor was fabricated by dispersing the coated U-Mo powder in an aluminum matrix. We assessed the performance through irradiation tests, achieving up to 66.2% U-235 burnup. Post-irradiation analysis covered microstructure in the fuel meat region and the U-Mo powder-matrix interfacial region, utilizing optical microscopy, scanning electron microscopy, and electron probe micro-analysis.

8.10 Flow Behavior in Involute-Plate Research Reactor: RANS, LES and DNS Simulations

Y.Q. Yu, A. Bergeron, **J. R. Licht**

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

By modeling the turbulence with different numerical approaches, i.e., Reynolds-Averaged Navier Stokes simulation (RANS), Large Eddy Simulation (LES) and Direct Numerical Simulation (DNS), this study simulates the flow behavior in involute-plate research reactors with three levels of fidelity. For RANS simulations, three widely used turbulence models are applied by using the commercial Computational Fluid Dynamic (CFD) code STAR-CCM+. As for LES and DNS, the open-source CFD code, Nek5000, is applied given its outstanding scalability on High Performance Computer (HPC) and high-order technique. A single coolant channel is chosen as the computational domain. The velocity magnitude, lateral velocity and turbulence kinetic energy from different models are compared and analyzed. The results from RANS are found to be in good agreement with LES and DNS data. Although some discrepancies are found between LES and DNS results in some areas of the domain, the deviations are found to be small. Given that the computational cost of DNS calculation is an order of magnitude higher than LES, using LES for benchmarking RANS model appears to be a cost-effective approach. The study supports the validation of modern methods for the analyses of involute-plate reactors, which may help simplify designs and therefore reduce the cost and time required to achieve conversion.

8.11 Mechanical Properties of Irradiated U-10wt.%Mo Alloy Degraded by Porosity Development

Jason L. Schulthess, Katelyn Baird, Philip Petersen, Daniele Salvato, Hakan Ozaltun, William A. Hanson, **Jeffrey Giglio**, James I. Cole

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

A nuclear fuel consisting of a monolithic plate with a solid foil of U-10wt.%Mo is under development for use in the United States' high performance research reactors. In support of developing this fuel, for the first time the fuel has been fabricated by a commercial fuel fabrication vendor and subsequently irradiated in a test reactor. This provides an opportunity to evaluate post-irradiation mechanical properties of commercially fabricated fuel. Four-point bend testing was conducted on the irradiated U-Mo fuel and the data produced includes bending strength and Young's modulus. Although the material behaves in a brittle manner due to the developed porosity general trends of reducing strength and modulus are found as fission density increases. The data produced are evaluated using both Weibull statistics and a modulus degradation model with recommendations provided.

8.12 Characterization of Zirconium Diffusion-Barrier Interlayer in Irradiated MP 1 Fuel Plates

Jan-Fong Jue, Jatu Burns, Tammy Trowbridge, Daniel Murray, Scott Anderson, James Cole, and Jeffrey Giglio

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

Zirconium and its alloys have been used for nuclear applications for decades and have been studied extensively. Fuel plates with a zirconium diffusion barrier have been fabricated and irradiated in the Advanced Test Reactor (ATR) at Idaho National Laboratory (INL). Results from preliminary post-irradiation examinations have shown that monolithic U-10Mo fuel plates, including a Zr diffusion barrier, remain intact after irradiation up to very high burnups (i.e., $>8 \times 10^{21}$ fission/cm^3). For low-burnup irradiations, in general, irradiation to a burnup up to 2.3×10^{21} fission/cm^3, the irradiation does not seem to significantly impact the zirconium diffusion-barrier interlayer or the interaction interlayer between Zr and U-10Mo. Porosity or bubble-like features start to appear in the low-Mo interlayer and some of the U-10Mo grain boundaries at a burnup of less than 0.85×10^{21} fission/cm^3. The diffusion barrier is behaving appropriately in all low-burnup fuel plates characterized in this study. For medium-burnup irradiations, up to a burnup of 5.4×10^{21} fission/cm^3, the irradiation behavior of Zr diffusion-barrier interlayer in the monolithic fuel plates is stable and predictable.

8.13 MP-2 Design, Irradiation Status, and Expected Sample Availability

M.A. Marshall, D.O. Choe, G.L. Hawkes, S.J. Byington, P.E. Gilbreath, J.W. Barney, E.S. Rosvall

Idaho National Laboratory, 2525 N. Fremont Avenue, Idaho Falls, ID 83402 – USA

The Mini-plate-2 (MP-2) experiment started irradiation at Idaho National Laboratory's Advanced Test Reactor in April of 2023. It is designed to cover the operating envelop of U-10Mo monolithic in four U.S. high performance research reactors. The purpose of MP-2 is to gather a statistically significant amount of irradiated fuel data to demonstrate fuel performance as required for fuel qualification. Irradiation will continue through 2026 but samples will be shipped periodically starting in 2025 to enable post irradiation examinations (PIE) to occur concurrently with the irradiation campaign. The design of the MP-2 experiment will be summarized including a comparison with irradiation conditions

of MP-1, a mini-plate irradiation concluded in 2019. Status of irradiation will be shared. And planned sample availability for PIE will be discussed.

8.14 Porosity Observations with FUTURE-HFIR Silicide Plates

J. F. Jue, **T.D. Marshall**, F. G. Di Lemma, J. Burns, J. J. Giglio, H.T. Hartman, and G.K. Housley

Idaho National Laboratory, 2525 N. Fremont Avenue, Idaho Falls, ID 83402 – USA

Characterization of as-fabricated U_3Si_2 fuel plates is used to establish a baseline for comparison of non-irradiated and irradiated fuel observations by microscopic methods. Two types of U_3Si_2 fuel plates were sectioned, polished, and evaluated by scanning electron microscopy (SEM) for microstructural characteristics. An additional analysis for porosity was performed using different methodologies, in order compare results and minimize the source of errors. New technical methodologies were employed in this work to facilitate the samples microscopic characterization. The focused ion beam (FIB) based specimen preparation method has been used to minimize the potential effects of matrix smearing and "pull-out" of fuel particles from the sample matrix. The advantages and challenges for different methodologies for the measurement of porosity in the fuel zone of the FUTURE-HFIR silicide archive fuel plates will also be presented.

8.15 COMSOL Results for the Proposed LEU Silicide Fuel Designs Under 95 MW Nominal Conditions for HFIR Conversion

Prashant K. Jain

Oak Ridge National Laboratory, 1 Bethel Valley Road, Oak Ridge, TN 37830 – USA

Ongoing engineering design studies at Oak Ridge National Laboratory are exploring the feasibility of converting the High Flux Isotope Reactor (HFIR) from highly enriched uranium (HEU) to low-enriched uranium (LEU) fuel. HFIR is a pressurized, light water-cooled and -moderated research reactor with a core composed of involute-shaped, HEU containing fuel plates and coolant channels. Advanced multiphysics computational fluid dynamics models have been developed in COMSOL to simulate the steady-state operating conditions for the proposed LEU U_3Si_2-Al (uranium silicide dispersion) fuel designs. The HFIR inner and outer fuel element models incorporate various essential physics, such as spatially dependent nuclear heat deposition, multilayer heat conduction, conjugate heat transfer, turbulent flows (using Reynolds Averaged Navier Stokes turbulence models), structural mechanics (thermal-structural interactions and fuel swelling), and oxide layer build-up. This poster presents the best-estimate nominal thermal hydraulics results for both the low and high-density silicide LEU core designs under 95 MW nominal operating conditions.

8.16 Verification and Validation of RELAP5/MOD3.3 to Support the Research and Test Reactors Program

D. S. Yoon, E. E. Feldman, J. W. Thomas, and E. H. Wilson

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

Verification and validation (V&V) of RELAP5/MOD3.3 patch 5 has been performed for the capabilities selected based on the intended use in Research and Test Reactors Programs at Argonne National Laboratory. Five categories of capabilities were identified: 1) steady-state single-phase liquid thermal hydraulics capabilities, 2) transient single-phase liquid thermal hydraulics capabilities, 3) mechanical component capabilities, 4) reactor kinetics capabilities, and 5) two-phase thermal hydraulics capabilities. A suite of RELAP5 test cases has been assembled to test the identified capabilities. For the first four categories, acceptable values for key variables in all of the test cases were obtained by hand calculations or code-to-code comparisons. For the verification of the two-phase thermal hydraulics capabilities, the RELAP5 developmental assessment problems were leveraged, which compare code predictions to experimental results. Overall, good comparison results were obtained and the identified capabilities of RELAP5 are satisfactorily verified. The poster summarizes the verification methodology and provides examples of these comparisons. By openly sharing our approach, the research reactor community may develop best practices for software quality assurance.

8.17 Modelling NBSR Water Gap Flow Distributions for DDE and LEU Element

M.E. Tano, T.D. Marshall, G.K. Housley, H.T. Hartman

Idaho National Laboratory, 2525 N. Fremont Avenue, Idaho Falls, ID 83402 – USA

The Low Enriched Uranium (LEU) Design Demonstration Element (DDE) for the National Bureau of Standards Reactor (NBSR) features a two-inch water gap between its upper and lower fuel plates, whereas the NBSR fuel utilizes a six-inch water gap. Through isothermal Monte Carlo neutron analyses, the shorter DDE gap was intentionally designed to maintain the neutron flux peaking factor between the NBSR fuel and DDE irradiation experiment. However, this shorter gap diminishes flow mixing, potentially leading to a coupling of vortex detachment from the upper set of plates with flow development at the lower plates. We conducted Computational Fluid Dynamics (CFD) modeling to compare flow distributions between the two- and six-inch water gaps. The CFD analysis assessed flow streamlines, gap vorticity, and channel velocity distribution for both configurations, with results demonstrating essential equivalence in flow patterns between the two areas. A separate neutronics analysis aimed at investigating moderation effects is planned.

8.18 CARTHAGE Flow Test Facility and DDE Modeling

B. Rossaert[1], **T. Marshall**[2], G. Housley[2], H. Hartman[2], and J. Wight[1]

[1] SCK CEN, Boeretang 200, 2400 Mol – Belgium
[2] Idaho National Laboratory, 2525 N. Fremont Avenue, Idaho Falls, ID 83402 – USA

The Design Demonstration Element (DDE) has the goal of verifying fuel performance modeling for low enriched uranium conversion. Two DDE irradiation experiments will occur in the BR2 reactor of SCK CEN. Flow velocities and pressure losses are critical measurements from the experiment. The CARTHAGE flow test facility at SCK CEN is a full-scale mockup of the coolant flow path within channels of the BR2 core. Flow through a DDE irradiation vehicle for BR2 will be thoroughly characterized using CARTHAGE. This characterization includes 3-sigma statistical confidence values for the measurements of coolant bulk temperature and pressure, inlet plenum flow rates, pressure loss across the DDE, and leakage flow rates. Thermal hydraulic modeling of the DDE was performed with both the RELAP5-3D system code and STAR-CCM computational fluid dynamics code. Structural modeling was performed with the ABAQUS mechanical code. Modeling predictions are for the expected data ranges and will be verified with measured data from CARTHAGE.

8.19 Flow Test of USHPRR LEU Fuel Elements for Hydraulic Performance Evaluation

Guanyi Wang, Cezary Bojanowski, **Andrew Hebden**, Walid Mohamed, Erik Wilson

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

Aaron Weiss, Michael Legatt, Wade Marcum

Oregon State University, 3451 SW Jefferson Way, Corvallis, OR 97333 – USA

To support the conversion of U.S. High Performance Research Reactors (USHPRR), specifically MITR, MURR, and NBSR, from highly enriched uranium (HEU) to low-enriched uranium (LEU) fuel, a flow test campaign as an essential part of the hydraulic performance evaluation will be conducted. The purpose of the hydraulic performance evaluation is to test a prototypic commercially-fabricated LEU fuel element to determine whether any flow-induced failure modes are observed or predicted in the fuel element, including significant deformations such as plate bending, twisting, or plate detachment from the side plates under reactor flow conditions accounting for several relevant safety basis parameters. This evaluation will be performed by means of an out-of-pile flow test at the Hydro-Mechanical Fuel Test Facility (HMFTF) of Oregon State University (OSU) and computational analysis. This work summarizes the ongoing design efforts of the flow test vehicle and instrumentation approach.

8.20 PARET/ANL 7.6 Modeling Approaches and Validations

M. Sharabi, A. P. Olson, S. R. Yang, M. Kalimullah, and **J. R. Licht**

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

PARET/ANL 7.6 software is developed at Argonne National Laboratory (ANL) under the sponsorship of the U.S. Department of Energy/National Nuclear Security Administration (NNSA). The code is used in the safety analyses of research and test reactors to analyze severe accidents caused by reactivity insertion events in reactor cores cooled by light and heavy water, with fuel composed of either plates or pins. The code solves transient coupled neutronics, heat transfer and hydrodynamics equations with reactivity feedback including fuel expansion, coolant density and temperature, and Doppler effect. Recently, PARET/ANL 7.6 has been subjected to a software dedication procedure in accordance with the NQA-1 standard to assure that it can successfully perform its intended safety function. The current work shows the general modeling approaches adopted in the code development, and in addition, describes details of code Verification and Validation (V&V) needed to demonstrate code capabilities.

8.21 Safety Updates of NIRR-1 Oversight

Kayode James Adedoyin

Nigerian Nuclear Regulatory Authority, Plot 564\565, Airport Road, Central Business District Abuja – Nigeria

Objective

• To bring out the Safety Requirements and Safety Assessment of Licensing Research Reactors (NIRR-1).

Introduction

• This paper contains details of Safety Analysis performed for the NIRR-1 Core Conversion from Highly Enriched Uranium (HEU) fuel to Low Enriched Uranium (LEU) fuel and the Lessons Learnt from the various licensing stages. It also highlights the various Safety Assessment and Inspection findings of the Facility Operations. This includes Radiation Monitoring, Physical Security, Quality Assurance and Maintenance. This is needed to ensure safety is given adequate priority.

• The Nuclear Safety and Radiation Protection Act 19 of 1995 established the Nigerian Nuclear Regulatory Authority (NNRA). The NNRA has responsibility for Nuclear Safety and Radiological Protection Regulation in Nigeria. Among the activities regulated by the NNRA is the Miniature Neutron Source Reactor (MNSR) at CERT.

Conclusion:

This paper highlights the Safe Operations of NIRR-1 (post fuel conversion), as regards its safety operation.

8.22 Economical Zirconium/Titanium Electroplating at Low-Throughput Commercial Scales

C. Arendt and L. Hubbard

Pacific Northwest National Laboratory, 902 Battelle Boulevard, Richland, WA 99352 – USA

Refractory metal electroplating (zirconium, titanium, etc.) is commercially needed for nuclear fuels, high-vacuum equipment, and electronic technology. Currently, commercial refractory metal electroplating is limited to very high-temperature baths, restricting its broad applicability. We have demonstrated the process at low-throughput commercial scales for the electroplating of zirconium and titanium from near room-temperature ionic mixtures. Our system is significantly less energy intensive, more economical, and produces less waste than current alternatives. Initial testing vs. vapor deposited metal layers showed similar performance in terms of bond strength and reliability with the added capability of plating 3D workpieces. The demonstration of an economical plating method for refractory metals enables a plethora of game-changing nuclear, medical, aerospace, electronic, and thermal systems that would rely on a tough metal coating to protect delicate internal components.

SESSION 9 Fuel Fabrication Challenges and Advances

9.1 USHPRR Fuel Fabrication Pillar: Achievements and Advancements in Fabrication of U-10Mo Monolithic Fuel

Vineet V. Joshi, Ayoub Soulami, Michael Catalan, Mark Rossiter, Curt Lavender

Pacific Northwest National Laboratory, 908 Battelle Boulevard, Richland, WA 99354, USA.

The National Nuclear Security Administration's Material Management and Minimization Office is continuing to demonstrate the fabrication capability of an acceptable low-enriched uranium nuclear fuel for the conversion of U.S. research reactors from highly-enriched uranium to low-enriched uranium. Through extensive efforts at the Y-12 National Security Complex and BWXT, the Fuel Fabrication Pillar has made substantial strides in demonstrating and refining the baseline U-10Mo fabrication process. The manufacturing of the U-10Mo alloy involves a series of thermomechanical processing steps and achieving a consistent product that meets the specifications with high yield is of utmost importance. This paper elucidates the notable progress achieved in comprehending the U-10Mo fuel characteristics. These advancements have helped with improving the process yields while concurrently minimizing the waste. Furthermore, the paper discusses how these findings can be applied to future fabrication campaigns and irradiation experiments, enhancing our ability to consistently produce U-10Mo alloy that meets the specifications while potentially optimizing the manufacturing process for greater efficiency and effectiveness.

9.2 Impact of the Fabrication Specification Tolerances of U-10Mo LEU Fuel on MITR and MURR

V. Mascolino, S. Yang, D. Yoon, K. Anderson, J. Stillman, D. Jaluvka, W. Mohamed, E. Wilson

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

M. Pinilla

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

L. Hu

Massachusetts Institute of Technology Nuclear Reactor Laboratory, 138 Albany St, Cambridge, MA 02139 – USA

MITR and MURR are two of the U.S. High Performance Research Reactors (USHPRR) that will undergo conversion from highly enriched uranium (HEU; ≥20 wt.% U-235) to low-enriched uranium (LEU; <20 wt.% U-235). For both reactors, the proposed LEU fuel system is a high-density monolithic alloy of uranium-10 wt.% molybdenum (U-10Mo). The preliminary LEU fuel element designs for both MITR and MURR have been confirmed to meet key safety and operational requirements during normal operation and transient conditions. The impact of the fabrication specification tolerances of the LEU fuel element design, including the prototypic U-10Mo fuel core and AA6061 cladding and side plates material compositions, was evaluated for both the USHPRR to assess the sensitivity of neutronics and thermal hydraulics safety and operational characteristics of the designs. Recommendations were provided for maintaining or adjusting tolerances for key specification parameters based on the results of the analysis. These recommendations are expected to be implemented in the next planned revisions of the LEU fuel element specifications and drawings for these reactors.

9.3 Uncertainty Quantification for the NBSR – Integrating Full-Core Sensitivity Coefficients and Nuclear Data Covariance Matrices

C. Lu, C. Sears, A. Cuadra, P. Kohut, and **L.-Y. Cheng**

Brookhaven National Laboratory, P.O Box 5000, Upton, NY 11973 – USA

This study quantified the impact of nuclear data uncertainties on the k_{eff} for the NBSR loaded with low-enriched uranium (LEU) fuel. The Uncertainty Quantification (UQ) analysis was performed by using the sandwich formula, integrating sensitivity coefficients with nuclear data covariance matrices. Sensitivity coefficients (k_{sen}) for some of the impactful isotopes were generated by using two alternate models, a full-featured MCNP model of the NBSR and a simplified full-core model. The execution of MCNP limited the two models to use different neutron histories. The covariance matrices were extracted from ENDF/B-VII.1 neutrons sub-library evaluation files. The matrix manipulations were automated by using an BNL-in-house Python-based sandwich formula widget. Results at both the Start-Up (SU) and the End-Of-Cycle (EOC) state points indicate that nuclear data uncertainties could lead to ~1700 pcm uncertainties in the predicted NBSR k_{eff}. The consequence of using two different models to calculate the sensitivity coefficients appears to be small.

SESSION 10 Fuel Performance Analysis and Modeling

10.1 Evaluation of Uranium Silicide (U_3Si_2) Fuel Using 80 MeV Xenon Ions at the ATLAS Material Irradiation Station

William Limestall, Peter Mouche, Gyuchul Park, Shipeng Shu, Sumit Bhattacharya, Bei Ye, and Abdellatif Yacout

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

Jeff Terry

Illinois Institute of Technology, 10 West 35th Street, Chicago, IL 60616 – USA

Mitigation of risks associated with the use of highly enriched uranium (HEU) in research and test reactors has directed focus towards alternative fuels. Uranium Silicide (U_3Si_2) emerges as a prominent candidate, primarily due to its exceptional uranium density and reliable irradiation performance. As a low-enriched uranium (LEU) fuel, U_3Si_2 not only offers non-proliferation advantages but also exhibits outstanding performance in neutron-rich settings. Additionally, its unique properties allow for the design of compact fuel elements compatible with existing reactor cores, negating the need for extensive modifications. Current studies are exploring the radiation endurance of ion beam irradiated fuels, using 80 MeV Xenon at the ATLAS Material Irradiation Station (AMIS). This methodology accelerates irradiation damage assessments in fuel candidates from potentially many months to years to merely days or hours.

10.2 Validation of the Thermal Calculation in the DART Fuel Performance Code Through the Benchmark with the PLTEMP/ANL Code and the Comparison with Measured Oxide Layer Thickness Data

Bei Ye, Hsun-Chia Lin, Shipeng Shu, Jeremy Licht, Patrick Garner

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

Ann Leenaers

SCK CEN, Boeretang 200, B-2400 Mol – Belgium

To support the safety analysis for BR2 conversion, the DART fuel performance code will be used to generate some required input data which cannot be measured during irradiation, i.e. coolant-side oxide layer thickness and fuel meat thermal conductivity as a function of fission density. To ensure the data quality, it is important to first validate the thermal calculation models in the DART code, as these parameters are temperature dependent. Due to the absence of measured temperatures across a fuel plate during irradiation, one feasible approach of checking the thermal calculation is to perform rigorous benchmark with other verified and validated computational codes, for example, the PLTEMP/ANL code, which is a computational code for thermal-hydraulic analysis of research and test reactors. Temperature distributions across a fuel plate at multiple interfaces along the thickness direction were calculated with the two codes independently and compared using a generic power history of a full-size fuel plate with a peak heat flux of 550 W/cm^2. The comparisons were made for the calculation results using either Colburn or the modified Dittus Boelter heat transfer coefficients between the coolant water and plate surfaces. In addition, the DART-calculated oxide layer

10.3 Integrating a Mechanical Analysis Module into DART to Predict Local Swelling in Monolithic and Dispersion Fuel Plates

Shipeng Shu, Bei Ye, **Yeon Soo Kim,** Aaron Oaks, Kun Mo, Yinbin Miao, Abdellatif M. Yacout

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

Ben Beeler

North Carolina State University, 3140 Burlington Engineering Labs, 2500 Stinson Drive, Raleigh, NC 27695 – USA

Adam Robinson

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

This study introduces the mechanical analysis (MA) module, integrated into the Dispersion Analysis Research Tool (DART) computational code. The resulting DART-MA code package enhances fuel performance simulations, specifically emphasizing local swelling variations in both monolithic and dispersion fuel plates. Accurate estimates of stresses within the plate are important, as they influence the behavior of fission gas bubbles, thus allowing for a more precise evaluation of factors such as the impact of coolant pressure. Using the finite element method and effective-medium approximation, the MA module underscores the effects of cladding constraints and irradiation creep at fuel meat edges, though the central mid-plane swelling, that is crucial for swelling correlations, remains mostly unaffected. Several parameters in the MA module were fitted to post-irradiation examination (PIE) data, and separate PIE data sets were applied to assess the predictive capabilities of DART-MA. Additionally, when coupled with the mechanistic fission gas behavior module in DART, DART-MA can be used to explore the influence of coolant pressure, a major operating parameter, on fuel plate swelling. The preliminary simulation results show that the potential influence of coolant pressure on mid-plate swelling is insignificant. The results from this study demonstrate DART-MA is a useful tool to understand the effects of local stress distribution. More experimental data is needed to further verify and refine the MA module.

10.4 Water Channel Thickness Estimation Through High Frequency Ultrasonic Device

R. Mrabti, G. Despaux and E. LE Clézio

University of Montpellier, IES, 860 Rue St Priest, 34095 Montpellier – France

Y. Calzavara

Laue Langevin Institute, 71 Rue des Martyrs, 38000 Grenoble – France

The PERSEUS project is a collaboration between the Laue Langevin Institute (ILL) and the Electronic and System Institute (IES) of Montpellier University. It aims at analyzing the inter-plate distance of a HPRR spent fuel element. This device being developed integrates two ultrasonic transducers resonating at frequencies around 100 MHz and is machined at the end of a 1 mm thick inox blade. The two transducers are coupled to an electronic system for the emission and acquisition of the ultrasonic signals after propagation in the water channel. In the present work, the ultrasonic transducer structure and wave propagation are simulated to analyze the echoes produced along the entire path. This paper will provide a comprehensive overview of the repeatability of laboratory measurements on 1.8 mm thick samples, and it will detail the results of the first effective in-situ measurement performed along a height of 50 cm of one of the ILL fuel element's water channels.

10.5 Irradiation Thermo-Mechanical Analysis of LEU Fuel for MURR Conversion

F. Cetinbas, W. Mohamed, **D. Yoon**, J. Stillman, V. Mascolino, and E. Wilson

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

M. Pinilla

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

The University of Missouri Research Reactor (MURR) is one of the U. S. High Performance Research Reactors collaborating with the National Nuclear Security Administration Material Management and Minimization Reactor Conversion Program to convert from the use of highly enriched uranium (HEU; ≥20 wt.% U-235) to low-enriched uranium (LEU; <20 wt.% U-235) fuel. Safety Analysis for the MURR conversion and licensing necessitates predicting and evaluating the irradiation thermo-mechanical behavior of MURR LEU fuel plates and elements. Three-dimensional finite element (FE) models are developed for MURR LEU plates with neutronics and thermal hydraulics data representing prototypic irradiation conditions for the proposed MURR LEU fuel core. Material behavioral models and properties from fuel qualification efforts are input into the FE models. FE analysis results for MURR LEU fuel plates 22 and 23 are used to calculate the thickness change in adjacent coolant channels to guide the thermal hydraulics safety analysis. Preliminary FE analysis results of the MURR LEU element-level modeling that account for other structure components and the effect of oxide layer growth are presented.

SESSION 11 Design and Analysis Methods

11.1 Preliminary Comparison of Depletion and Fuel Management Software MCODE and ADDER for the MIT Research Reactor

M. Garanzini and L. Hu

Massachusetts Institute of Technology Nuclear Reactor Laboratory, 138 Albany St, Cambridge, MA 02139 – USA

K. Anderson, V. Mascolino, E. Wilson

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

The Massachusetts Institute of Technology Reactor (MITR) is a 6 MW research reactor currently operating with a highly enriched uranium (HEU) plate-type fuel and planning to undergo conversion to a low-enriched uranium (LEU) fuel. Fuel management calculations and in-core experiment modeling for MITR are currently carried out using MCODE, which allows for the coupling of a neutronics code and a depletion code. The Advanced Dimensional Depletion for Engineering of Reactors (ADDER) software, developed and maintained by Argonne National Laboratory, is being evaluated to be used in place of MCODE as it provides a more flexible and performant approach to fuel management and depletion for both HEU and LEU fuel. A preliminary code-to-code comparison between MCODE and ADDER was carried out for simplified cases consisting of short operation cycles for a fresh HEU core with various types of in-core experiments. Analysis results showed good agreement between the two software. Future work includes 10-week fuel cycles and fresh LEU core studies, to better represent the use of ADDER as the MITR fuel management software after conversion.

11.2 Fluid Structure Interaction Modelling for MITR and NBSR DDEs

M.E. Tano, T.D. Marshall, C. Xing, C.J. Jesse, and A.M. Phillips

Idaho National Laboratory, 2525 Fremont Avenue, Idaho Falls, ID 83402 – USA

The USHPRR Fuel Qualification Pillar has developed a comprehensive thermomechanical Fluid-Structure Interaction (FSI) approach since 2015 to model Low Enriched Uranium (LEU) fuel plates and Design Demonstration Elements (DDEs). FSI analyses were conducted under prototypic conditions for irradiation experiments at MITR and NBSR. This approach involves tight coupling between high-fidelity wall-resolved Computation Fluid Dynamics (CFD) models for the flow field and a thermo-elasto-plastic model for the fuel plates, accounting for stress, temperature, and irradiation swelling and creep. The model enhances predictions of fuel and clad temperatures, mechanical stress, and elastic and plastic deformations during irradiation. Two key generalizable findings emerged: firstly, granular temperature data from CFD analyses significantly alter fuel performance predictions compared to system-code-based predictions, particularly at flow occlusion points. Secondly, the strong nonlinear coupling in FSI analyses emphasizes the importance of considering coupled fuel performance effects like swelling and creep, debunking the notion of neglecting deformation mechanisms a-priori.

11.3 Impacts of Irradiated MURR LEU Fuel Thermo-Mechanical Behavior on Thermal Hydraulics Safety Analysis

D. S. Yoon, F. Cetinbas, G. Wang, J. A. Stillman, W. Mohamed, and E. H. Wilson

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

M. Pinilla

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

The University of Missouri Research Reactor (MURR) is one of the U. S. High Performance Research Reactors collaborating with the National Nuclear Security Administration Material Management and Minimization Reactor Conversion Program to convert from the use of highly enriched uranium (HEU; ≥ 20 wt.% U-235) to low-enriched uranium (LEU; < 20 wt.% U-235) fuel. Recent analysis results for channel gap reductions due to the irradiation thermo-mechanical effects based on the three-dimensional finite element analysis of the MURR LEU fuel plates and element has enabled evaluation of impacts on MURR LEU thermal hydraulics safety margins. Specifically, a significantly larger channel gap reduction is predicted for the outermost channel gap at the channel centerline compared with modeling assumptions employed for preliminary safety analyses. This location was previously determined to be not limiting, and thus not evaluated for predicting the safety margins. This work presents a preliminary analysis for the evaluation of thermal hydraulics safety margins considering the recent analysis results from the thermo-mechanical modeling of MURR LEU fuel plates and element. Preliminary results show that the local safety margins in the outermost end channel at the centerline is decreased due to the larger gap reduction at the local level; therefore, it is recommended that this location is included in evaluating the safety margins. Sufficient margins are still maintained for the MURR LEU operation for all evaluated cases.

11.4 System Code Modelling for NBSR DDE

M.E. Tano-Retamales, **T.D. Marshall**, G.K. Housley, and H.T. Hartman

Idaho National Laboratory, 2525 Fremont Avenue, Idaho Falls, ID 83402 – US

Both commercial and research reactors have used established system codes for their safety analysis report because of the extensive experiment database from which these codes were developed. The Low Enriched Uranium (LEU) Design Demonstration Elements (DDEs) for the National Bureau of Standards Reactor (NBSR) was analyzed using the RELAP5-3D, version 4.5.2, system code for the prediction of channel flow rates, clad and fuel temperatures, pressure losses, and surface heat flux. The DDE results were compared with data from the NBSR Final Safety analysis report (FSAR) to demonstrate the model's accuracy. The NBSR DDE RELAP5-3D model was used to examine variations in the fuel plate power density and channel flow rate in order to understand potential variations in the irradiation experiment conditions. System code analysis is one of the critical components of thermal hydraulics modeling for the fuel qualification process.