ADVANCED FUELS CAMPAIGN 2023 Accomplishments





Nuclear Fuels Cycle & Supply Chain

Advanced Fuels Campaign 2023 Accomplishments

INL/RPT-23-75335

November 15, 2023

Compiled and edited by:

Phyllis King

Jeatter Anedema

Heather Medema

Approved by:

Daniel Wachs, FCT AFC National Technical Director

11-15-2023

11-15-2023

11-15-2023



TABLE OF CONTENTS

AFC MANAGEMENT AND INTEGRATION

1.1	Advanced Fuels Campaign Leadership Team	.10
1.2	From the Director	.12
1.3	Showcase Capabilities	.14
	THOR Advanced Fuel Test Device	14
	Massachusetts Institute of Technology Reactor (MITR) High-Temperature Water Loop Design	20
1.4	International Collaborations	.24
	AFC International Engagement	24



ADVANCED LWR FUELS / HIGH BURNUP

2.1	Fuel Fabrication and Properties	32
	Thermal Properties of ATF Claddings	32
	Summary of Laser Flash Analysis measurements on Cr-coated cladding	36
	Handbook on Accident Tolerant Fuel Doped UO ₂ Properties	40
2.2	LWR Core Materials	44
	Initial Deployment of MiniFuel in the Removable Beryllium Reflector	44
	FeCrAl Optimization and Corrosion Results	50
	Mechanical Properties of ATF Cladding	54
	Advanced Characterization of SiC Subjected to Transient Irradiation	58
	Thermal Conductivity Measurement of Fresh SiC CMC	62
	SiC/SiC Development Strategy and 5-Year Execution Plan	66
2.3	LWR Irradiation Testing	70
	ATF Rodlet Irradiation in ATR (ATF-2C)	70

2.4	LWR PIE	74
	Examinations of BWR Lead Test Rods	74
	Commercial Lead Test Rod Transportation and PIE	78
	PIE Technique Development for SiC Claddings	
	DIC Developments for RHT and Coating Behavior Assessment	
	Mechanical Testing – Methodology, Evaluation, Testing, and Analysis	
2.5	LWR Fuel Safety Testing	90
	SATS 2.0 Capability Demonstration	90
	tFGR Measurement and In-Cell Demonstration	94
	Performance of the HERA PreHydrided Experiments	98
	TWIST LOCA Device Assembly and Commissioning	102
	High Burnup Fuel Loss-of-Coolant Accident Experiment Design	108
	High Temperature Cladding Creep Performance of Cr-coated Cladding	112
	HBu UO2 Transmission Electron Microscopy	116
	Investigation of Degradation Mechanisms of Cr-coated Zirconium Alloy Cladding in Reactivity Initiated Accidents	120
2.6	Performance Assessment	124
	Analysis of Fuel Utilization for Microreactors	124
2.7	ATF Industry Advisory Committee	128
	Accident Tolerant Fuel Industry Advisory Committee 2023 Summary	128
2.8	ATF Industry Teams	130
	ATF Industry Teams Overview	130
	Accident Tolerant Fuel (ATF) Industry Teams – Westinghouse Electric Company FY23 Accomplishments	132
	Accident Tolerant Fuel (ATF) Industry Team - Framatome	136
	General Electric Progress in Developing Accident Tolerant Fuels	140
	ATF Industry Teams: General Atomics – Electromagnetic Systems (GA-EMS) Accomplishments	144



ADVANCED REACTOR FUELS

3.1	Fuel Fabrication and Properties	148
	Chemical Interaction and Compatibility of Uranium Nitride Fuels with Liquid Pb and Alumina-forming Austenitic Alloys	148
3.2	AR Core Materials	152
	Advanced Reactor Core Materials	152
	HT9 Modeling	
	Linear and Nonlinear Guided Ultrasonic Waves to Characterize Cladding of Accident Tolerant Fuel	156
3.3	AR Irradiation Testing and PIE Techniques	
	Examination of FAST-1 Accelerated Fission Rate Metallic Fuel Test	160
	Microstructure and Micromechanical Characterization of Cr Diffusion Barrier in ATR Irradiated U-10Zr Metallic Fuel	164
	Femtosecond Laser Ablation Machining & Examination - Center for Active Materials Processing (FLAME-CAMP)	170
3.4	AR Fuel Safety Testing	174
	Post-Transient Characterization of the Metal Fuel THOR-C-2 Experiment	174
	First Irradiated-Fuel Fast Reactor Experiments in TREAT: Assembly and Performance	176
	NEUP 20-19374: Maintaining and Building upon the Halden Legacy of In-situ Diagnostics	
3.5	AR Performance Assessment	
	Pre-Test Fuel Performance Evaluation of THOR-C Experiments	
	FAST to FBR-II Comparison and HT9 Applicability	190

CAPABILITY DEVELOPMENT

4.1	ATR Loop Installation	
	Expanding ATR Loop Testing Capability	194
4.2	TREAT LOCA Testing	
	TREAT LOCA Testing and Large Experiment Capability	
4.3	Refabrication and Instrumentation	
	Refabrication/Reinstrumentation Capability Development	
4.4	Fast Neutron Irradiation	
	Roadmaps and Collaborative Opportunities for Fast Neutron Irradiations	210



5 APPENDIX

5.1	Publications	
5.2	FY-23 Level 2 Milestones	
5.3	FY-23 Milestone Report List	
5.4	AFC Nuclear Energy University Projects (NEUP) Grants	
	Active Projects Awarded in 2018	226
	Active Projects Awarded in 2019	227
	Active Projects Awarded in 2020	228
	Active Projects Awarded in 2021	229
	Active Projects Awarded in 2022	230
	Active Projects Awarded in 2023	231
5.5	Acronyms	232
5.6	Divider Photo Captions	242

AFC MANAGEMENT AND INTEGRATION

- 1.1 Advanced Fuels Campaign Leadership Team
- 1.2 From the Director
- 1.3 Showcase Capabilities
- 1.4 International Collaborations

1.1 ADVANCED FUELS CAMPAIGN LEADERSHIP TEAM

Daniel Wachs National Technical Director (208) 526-6393 daniel.wachs@inl.gov





Edward Mai Deputy National Technical Director (208) 526-2141 edward.mai@inl.gov

Todd Pavey TREAT Project Management (208) 526-9023 todd.pavey@inl.gov

David Kamerman ATF Qualification Lead (208) 526-3128 david.kamerman@inl.gov



Takaaki Koyanagi SiC Development Qualification Lead (865)341-1927 koyanagit@ornl.gov

Kory Linton ORNL Project Management (865) 241-2767 lintonkd@ornl.gov



Nathan Capps High Burnup Oualification Lead (865) 341-0458 cappsna@ornl.gov

Douglas Porter Advanced Reactors Focus Lead (208) 533-7659 douglas.porter@inl.gov

Josh White Fabrication and Properties Technical Lead (505) 667-3879 jtwhite@lanl.gov



Nicolas Woolstenhulme

Irradiation Testing Technical Lead Advanced Fuel Design Strategic Lead (208)526-1412 nicolas.woolstenhulme@inl.gov

Pavel Medvedev Performance Assessment Technical Lead (208) 526-7299 pavel.medvedev@inl.gov



Tarik Saleh Core Materials Technical Lead (505) 665-1670 tsaleh@lanl.gov

Fabiola Cappia

Strategic Lead

(208) 533-7091

Nuclear Fuel Science

fabiola.cappia@inl.gov

Colby Jensen Safety Testing Technical Lead (208) 526-4294 colby.jensen@inl.gov



Luca Capriotti Post Irradiation Examination Technical Lead (208)533-7080 Luca.Capriotti@inl.gov

Stephen Novascone Fuel Modeling Strategic Lead (208)526-3729 stephen.novascone@inl.gov

Ed Beverly Project Controls (208)533-7709 edward.beverly@inl.gov





ReBekah Thompson Program Support Specialist (208)526-7956 rebekah.thompson@inl.gov



Arantxa Cuadra Technology Assessment Strategic Lead (631)344-2352 acuadra@bnl.gov



Phyllis King Systems Integration and Analysis Lead (208)526-4348 phyllis.king@inl.gov

2023 AFC ACCOMPLISHMENTS

1.2 FROM THE DIRECTOR



Daniel Wachs National Technical Director (208) 526-6393 daniel.wachs@inl.gov

orldwide urgency to develop and implement nuclear technology to deliver clean, reliable, and economic energy clearly sets the pace for research being conducted by DOE-NE and its industrial partners. It's essential to recognize that advanced fuel technology is at the center of that effort. Whether that stems from recent improvements in LWR fuel technology or emergence of 'new to industry' fuel technologies that are driving advanced reactors to the edge of deployment, the challenge we face as a research and development community is clear - the work we're doing now is the foundation for it all! Much of the current industrial opportunity is the logical end to a process that started 20 years ago when, recognizing the long term importance of nuclear energy to economic energy production, national security, and, ultimately, climate change, the office of Nuclear Energy moved out of the shadows and was once again elevated to the assistant secretary level within the DOE and federal investments in nuclear energy grew from ~\$100M in 2002 to nearly \$1.5B in 2023.

This revival has been partially fueled by the foundational accomplishments of DOE-NE's advanced fuels programs, most notably the accident tolerant fuels, metal fuels, and TRISO fuels programs that have delivered technology that underpins virtually every credible commercial reactor concept being considered by industry today. Without the development and qualification efforts performed by these programs (to date and those expected to complete in the next several years) none of these revolutionary reactor concepts would be viable for 'demonstration', let alone commercial deployment. This point clearly emphasizes the value and purpose of long-term R&D programs that operate with goals well beyond those that occupy the immediate attention of industry.

The AFC program holds a unique cross-cutting position in the fuel technology development ecosystem. Simultaneously investing in the discovery, development, and maintenance of technology being used now, in the near future, and in the long term. Balancing this portfolio is complex and challenging. At the current moment, our effort leans heavily into enabling expansion of nuclear technology use over the next decade by emphasizing qualification of the fuel systems needed to enhance the current fleet of LWRs and enable demonstration of advanced technologies (ranging from SMRs to LMRs and beyond).

After barely a decade of work, ATF technology is on the cusp of qualification and is entering the commercialization phase. Early phase ATF research by the labs and industry have enabled irradiation of lead test rods in commercial facilities that are now nearing full lifecycle (third cycle) irradiations. Examination of these rods in-canal and in DOE hot cell facilities at ORNL and INL are yielding significant results that will leverage further deployment and drive the simultaneous harvest of economic and safety benefits. Fuel vendors are building the infrastructure needed to deliver commercially relevant quantifies of fuel. Utilities are preparing to take advantage of the improved performance characteristics at their plants. The NRC is modernizing the regulatory framework to address modern methodologies and approaches needed to springboard the national energy portfolio into the future.

Throughout the process of reviving the national innovation engine for LWRs to drive ATF technology, the opportunity for self-reflection on existing technology has opened the door to enhancing the value of the current nuclear fuel technology as a stepping stone to the advanced designs. Opportunities to dramatically improve economic performance of the existing fleet have emerged. Primarily in the form of longer operating cycles at many plants (extending from 18 to 24 months) as well as realizing opportunities to potentially uprate powers. This requires developing the technical basis for increasing licensed burnup from ~62 GWD/MTU to as high as 75-85 GWD/MTU. This simultaneously requires extending enrichment beyond 5% to as high as 10%. Forging into these new service regimes requires extending both our scientific and engineering understanding of the stresses placed on fuel during service and how the fuel responds to them.

Facilitating this transformation has required that we simultane-

ously revitalize the US research infrastructure. Re-building the National LWR Testbed has been a high priority over the last decade and includes some very high water marks, including restart of the TREAT facility. However, it has also established more abstract strengths, like the re-nucleation and reconnection of scientific specialists that are capable of asking and answering the right questions. These talents have impact on resolution of challenges that are sprinkled throughout the nuclear energy ecosystem's body of stakeholders. While there's still work to do (looking at you ATR i-loops), the pivot toward harvesting the power of the National LWR Testbed is fast approaching.

But we're not satisfied at stopping there. Big things are on the horizon for nuclear energy. Interest and investment in the deployment of advanced reactor systems is at a level not seen in the United States since the 1960's. Deployment of demonstration facilities including both Natrium and X-energy are only possible following the decades of fuel development work performed on metal fuel by AFC and by our close sibling TRISO program. In fact, qualification of metallic fuel for broad industrial use is just about at our fingertips. Without the vision and effort of these foundational DOE programs, the emerging industrial deployment would simply not be possible. Its clear that these programs are the root of emerging Next Generation Fuels research and development program needed to

carry advanced reactor technology to the next level.

Again, over just the last 20 years, the DOE-NE budget has jumped by an order of magnitude as pressure to expand nuclear energy utilization began to grow. If we can capture the opportunity that faces us in the next several years, one can easily imagine a similar jump over the next 20 years as nuclear technology realizes its potential to satisfy energy needs of the nation and world. It's tremendously exciting to see that all of AFC is playing a growing role in the endeavor!

Dan Wachs

THOR Advanced Fuel Test Device

Principal Investigator: Colby Jensen

Team Members/Collaborators: Klint Anderson, Trevor Smuin, Jason Schulthess, Todd Birch, Ashley Lambson, Phil Peterson, Jordan Argyle, Clayton Turner, Sarah Khan, Randall Fielding, Devin Imholte, Scott Wilde, Austin Fleming, Nicolas Woolstenhulme, Andrew Chipman, Robert Armstrong, Matt Mihelish, Kelly Ellis, Sterling Morrill, Daniel Wachs and many more in the machine shops, HFEF, and TREAT

The complete deployment of the THOR test capsule system, from fresh to irradiated fuels, closes a significant gap for TREAT and for the sodium fast reactor fuels testbed to enable the next generation of nuclear reactors

he Transient Reactor Test Facility (TREAT) was restarted nearly six years ago. With it came the restoration of the most straightforward and sound approach of harnessing fission power to advance technology of the densest power source available to mankind. To this day, significant strides are being made to realize the full potential of TREAT through development of multiple in-reactor testbeds, which marry multiple unique nuclear fuel designs to their specific applications. Four years in the making, the sodium fast reactor fuel testbed has made a notable leap forward to tap TREAT's promise through the invention of the Temperature Heatsink Overpower Response (THOR) capsule, and fully establishing the framework and engineered systems that realize its usage as a sophisticated research and development (R&D) instrument. Significant investment of time, stress, intellect, consideration, ingenuity, and accountability in a high-performance team have accomplished this goal - demonstrated by the first global transient irradiation of a high burnup uranium-plutonium composition fuel, in more than two decades – as part of an international collaboration with the Japan Atomic Energy Agency (JAEA) in July 2023. The

THOR capsule is a major accomplishment in R&D capability to bridge towards safe deployment of advanced-nuclear-fueled reactors. The capsule and its infrastructure have paved the way for bigger goals in upcoming experiments testing high-burnup accident tolerant fuels and bringing the TREAT sodium loop online. Establishing the THOR testing system across Idaho National Laboratory (INL) laboratories and teams is a crucial step for execution in this new era of fuel safety testing.

Project Description:

The THOR capsule was born of a long-planned goal to restore transient testing capabilities to qualify safe, high-performing nuclear fuels for reactors that are among the most power dense and providing the most efficient path to closing the fuel cycle. The original concept was born in a proposal to support initiatives of TerraPower, but fully realized starting in 2019 when Japanese colleagues drafted a winning application to share U.S. R&D infrastructure, using capability in TREAT, not available in their own country or anywhere else in the world (Figure 1). The design of the capsule was created to provide a sodium reactor environment with excellent heat transfer, high temperature performance, sophisticated in-situ data



Figure 1. JAEA collaborator visits INL for final execution steps of a JAEA-INL joint experiment at TREAT

options, and robustness to test some of the most challenging environments on the planet. The central feature of the THOR capsule is a plain bar of metal, carefully selected to provide optimized heat transfer from a test specimen located in a centerline through-hole in the bar heatsink. The simplicity of the capsule was strategically selected to complement the more complex capability of the developing TREAT sodium loop. Figure 2 shows an overview of the capsule design highlighting critical components.



THOR Temperature Response





The THOR capsule utilizes specialized high temperature materials requiring unique fabrication approaches, mastered by the INL facilities over the past three years. Its high-temperature-strength alloy body are intertwined with cuttingedge instrumentation of optical fiber-based sensors, unique electromagnetic transducers uniquely produced in Norway, internal heating capability, and more than twenty thermocouple wire pairs to map thermal histories that tell the story of the fuel samples it holds. All these components must be fabricated and assembled carefully to ensure the performance of sensitive devices and signals, capitalizing on world-leading instrumentation laboratories and expertise in the Engineering Innovation Laboratory (EIL) at INL. Still, the biggest challenges of THOR lay ahead of the main capsule construction, in

ensuring logistical success to load sodium into the capsule and insert radioactive, high-burnup advanced nuclear fuel pins and segments. The delicate assemblies must be protected while processes and clever tooling devices were invented that ensure safe and successful capsule preparations. In all of this, complete engineering and detailed analysis records are produced under rigorous quality assurance standards to meet any potential customer requirements. Figure 3 highlights major experiment progressive steps.

Accomplishments:

A complex system such as the THOR device and its supporting infrastructure naturally come with surprises in the first-of-a-kind evolutions. A mountain of obstacles presents themselves even when led by the best irradiation testing and remote handling specialists in the world. For example, capsule leaks plagued Figure 3. Photos of an irradiated THOR experiment progression from fabrication and assembly through experiment irradiation (EIL High Temperature Test Laboratory -> Fuels and Applied Science Building -> HFEF -> TREAT->TREAT Control Room)



Figure 4. Select results of the THOR-C-2 experiment showing hodoscope detection of fuel failure and relocation compared to post-test x-radiography

Post-test X-radiograph

teams repeatedly due to strict leak rate requirements with many dozens of wires exiting the capsule and being handled across multiple nuclear facilities. The coordination and technical challenges of loading sodium in the only capable glovebox at INL, cask facility resources, navigating transportation of legacy radioactive special nuclear materials between multiple facilities were all overcome. The shocking upward ejection of an irradiated experimental breeder reactor (EBR)-II pin, out of the capsule during hot-cell insertion was fully resolved through creative thought and simple engineered solutions. A "Hail-Mary," ingenious solution to a faulty heater cable was developed, tested, and deployed in the TREAT core in a matter of days. A solution requiring invention of a novel repurposing of capsule thermocouples to energize the experiment without sacrificing any performance objectives and truly saving the experiment. A surprising discovery of a sodium

2023 AFC ACCOMPLISHMENTS

leak in the mixed oxide (MOX) experiment capsules, after prototype testing and four successful capsules, that finally resulted in rebuilding those experiments with some very simple design updates.

Overcoming each of these obstacles and more, were enormous successes along the way, which matured the TREAT teams to gain efficiency in working through high stakes challenges. Even more notable, the story of THOR's eventual success as a proven device is proven by highlights of: 1) achieving an engineered design that fully meets strict performance requirements developed jointly between JAEA and INL: 2) The successful irradiations of fresh U-10Zr fuel pins that provided power validation of the blind reactor physics calculations and validated the special thermal design of the system in 2021; 3) The following irradiation of a pressurized, fresh EBR-II driver fuel pin and a metallic fuel pin with the first infrared pyrometer inserted into a nuclear fuel in August 2022. These tests fully bore witness of the awesome TREAT hodoscope capability to measure fuel motion in-situ, giving truly unique insights into the true nature of even the worst conceivable reactor conditions (see Figure 4). The instrumentation showed its worth in precisely identifying the moment of fuel failure during the experiment while providing 3-d mapping of the driving thermal conditions of the

specimen. These tests have been going through advanced characterization described in another article in this report. Finally, this past summer the whole engineering package came together across the fabrication shops, instrumentation laboratory, the TREAT transportation cask, and three major nuclear facilities to assemble two THOR capsules with both high burnup MOX and metal fuel pins – also documented in more detail elsewhere in this report. One irradiated metallic fuel test was returned to the hot cell prematurely due to heater cable issues and a necessary push to complete the following MOX experiments prior to upgrading the TREAT core to support larger experiments. The following MOX experiment was irradiated in July 2022 to complete the THOR test platform and a level 2 milestone.

Looking ahead, the team is already implementing minor but impactful design improvements to ensure very high performance of the THOR capsule, to execute several more THOR experiments in 2024 in the larger test containment upgrade now deploying into TREAT. THOR's future entails a line of planned users, assuring its place in helping TREAT meet its national strategic mission of realizing advanced nuclear energy.

2023 AFC ACCOMPLISHMENTS

Massachusetts Institute of Technology Reactor (MITR) High-Temperature Water Loop Design

Principal Investigator: Kouroush Shirvan Team Members/Collaborators: Nicolas Woolstenhulme

> Technology Reactor (MITR) provides advanced capabilities for fuel cladding testing, particularly with its High Temperature Water Loop designed for boiling water reactor (BWR) conditions. This article delves into MITR's existing facilities and introduces the recent addition of a high-pressure water loop further enhancing its capacity for pressurized water reactor (PWR) environment testing.

Capability Description:

The MIT Nuclear Reactor Lab (NRL) offers a comprehensive set of irradiation experiment support, spanning vehicle design, fabrication, operation, to post-irradiation analysis and shipping. The 6 MW MITR can accommodate three primary in-core irradiation vehicles, supplemented by reflector and beamline irradiation positions. Additionally, pneumatic facilities cater to irradiations on smaller samples and conduct neutron activation analysis for safety reviews before in-core irradiation. The three main in-core positions at the MITR boast impressive neutron fluxes: 3.5x1013 thermal, 1.1x10¹⁴ E>0.1 MeV n/cm²-s, and around 1×10^9 R/hr gamma.

These slots utilize the core's full 18-inch height, plus some overhead, with a 2-inch diameter envelope inside each dummy fuel element for encapsulating irradiation vehicles. Presently, MITR's lab runs the High Temperature Water Loop (HTWL), an in-core pressurized water loop in one of the in-core experimental locations. It simulates prototypical BWR conditions, heating specimens up to 300°C, restricted mainly by the autoclave's in-vessel pressure rating (~10.3 MPa at present). All loop operations, from flow and heating to sampling, occur outside the primary reactor vessel. The in-vessel part positions samples in the core's peak neutron flux zone or gamma-only sections above the core. This loop is being extensively used for materials' corrosion, irradiation behavior, and real-time instrument testing. Typical samples in this setup include fuel cladding, corrosion tests, and isothermal specimens at light water reactor temperatures, generally up to 11 mm wide and 200 mm long. Various specimens and sensors can coexist, that provide temperature, neutron flux and electrochemical potential mapping. The capsule assembly is adaptable per cycle, ensuring alignment with project



goals and permitting sample swaps during outages.

NRL is integrating a novel water loop capability into the MITR, to be known as the Higher-Pressure Water Loop (HPWL). This HPWL can achieve elevated in-core temperatures, targeting peak PWR temperature conditions of approximately 360°C. To support the enhanced pressure and temperature requirements, a more robust titanium autoclave has been designed. Importantly, this new HPWL is intended to operate concurrently with the existing HTWL. This will effectively double the advanced fuel cladding irradiation capability at the MITR. While the design and fabrication of this innovative capability are progressing independently, the tasks in this project related to PWR conditions are strategically scheduled to leverage this capability as soon as it becomes operational. A comprehensive design for the

Figure 1. MITR HTWL schematic







Figure 3. HPWL component designs

The MIT Nuclear Reactor Lab sets a pioneering benchmark in irradiation experimentation, uniquely combining a versatile 6 MW reactor with capabilities such as the innovative High-Pressure Water Loop — all underpinned by high-fidelity simulations, an array of specialized irradiation vehicles, and adaptable in-situ electric heaters, acceleration nuclear fuel research and development

HPWL, bolstered by high-fidelity simulations, is already in place. The procurement of essential components and requisite engineering modifications are well underway. Construction for the HPWL is slated to commence in January 2024, and it is anticipated that the loop will be fully operational by October 2024. The HPWL also introduces an enhanced capability, incorporating in-situ electric heaters with a controllable heat flux up to 1.2 MW/ m² for emulating radial heat fluxes in standard cladding samples.

1.4 INTERNATIONAL COLLABORATIONS

Advanced Fuels Campaign International Engagement

Phyllis King

he Advanced Fuels Campaign (AFC) engages with international organizations and various countries to advance U.S. nuclear energy interests. Below are key interactions with global collaborators in fiscal year (FY) 23.

Nuclear Energy Agency (NEA)

Framework for Irradiation Experiments

Idaho Falls:

AFC hosted international members from the Framework for Irradiation Experiments (FIDES) October 14-21, 2022. FIDES is an international framework coordinated through the Nuclear Energy Agency (NEA) charged with conducting integral irradiation experiments of international importance. The FIDES membership consists of research and regulatory bodies from several NEA countries (Belgium, Czech Republic, Finland, France, Germany, Italy, Japan, Netherlands, Spain, Sweden, Switzerland, and United States). They discussed technical and business aspects of the program



Figure 1. FIDES Members from the following participating countries: Belgium, Czech Republic, Finland, France, Germany, Italy, Japan, Netherlands, Spain, Sweden, Switzerland, and United States.

including part of the program that takes place at Idaho National Laboratory (INL). INL is conducting irradiations in the Transient Reactor Test Facility (TREAT) reactor as part of this program and the results of those TREAT irradiations were highlighted at this meeting. Presentations were made on the status of the ongoing irradiation testing efforts and technical results were presented and reviewed.

A tour of the INL facilities related to irradiation testing also took place. The facilities included the Advanced Test Reactor (ATR), TREAT, Hot Fuels Examination Facility, and Irradiated Materials Characterization Laboratory.

Irradiation testing of nuclear fuels and materials is a key part of the Department of Energy (DOE) mission at INL. Hosting this group helped to establish DOE and INL international leadership in this important technical area.

Prague:

Members of the FIDES-II from 12 countries assembled in Prague, Czech Republic for meetings of its Technical Advisory Group and Governing Board. The NEA FIDES-II is a multilateral effort established to preserve and strengthen the global fuel and materials experimental capacity. This facilitates high-priority safety and operationsrelated experiments that benefit a broad community of users around the world.

During the meetings, the members received updates on the progress of four ongoing Joint Experimental Programmes (JEEP), including one that was added to the FIDES Programme of Work earlier in 2023.



The new JEEP, In-Core Real-Time Mechanical Testing of Structural Materials, expands the scope of technical areas covered under FIDES-II and re-establishes key capabilities lost with the closure of the Halden Reactor.

Discussions were held on the progress of the FIDES Strategic Plan, which provides direction to the future of the experimental program, with reviews of the status of ongoing work with the four existing FIDES projects and of the preliminary proposals being presented for the next triennium.

INL is the operating agent for DOE's project with NEA FIDES. The government benefits from this collaboration by contributing to the discussion and decisions being made at the technical advisory group and governing board meetings. Participation in-person ensures the DOE/ INL position on various aspects of the FIDES Strategic Plan (and thus the future direction of FIDES) is captured in the approved document. Additionally, participation in-person allows for DOE/INL to fully engage with other participants and allows for the NEA FIDES stakeholders to

Figure 2. Attendees of the French Alternative Energies and Atomic Energy Commission (CEA) Paris-Saclay visit.



Figure 3. (From left) Scott Holcombe, David Kamerman, Carolina Losin, Daniel Jadernas ask questions about the proposals and for the proposals to be highlighted during the discussions, thus strengthening key relationships in the group.

Boone Beausoleil participated in a meeting with the FIDES advanced fuels core group (related to Fission Accelerated Steady-state Testing-II experiments) and NEA where we discussed engagement with industry partners and how to bring them into the FIDES community. We also worked through the down selection of fuel types for advanced fuel testing in the FIDES-II proposal.

Studsvik Cladding Integrity Program

AFC staff attended and presented at the Studsvik Cladding Integrity Program (SCIP) IV Program meeting in Nykoping, Sweden. This meeting is a crucial part of aligning DOE high burnup LOCA activities helping to drive international efforts towards relevant testing. Discussions with Studsvik regarding ramp testing capabilities that were performed at the Halden and R2 reactors were held. The development of irradiation ramp testing capabilities is central to DOE's plans for irradiation testing of ATF. Discussions ensured the experimental capabilities and test plans developed at INL expand on the work done historically at Halden and R2. This interaction helped confirm that DOE's investment in irradiation ramp test capabilities at INL sees the maximum technical benefit.

Jason Schulthess and Robert Armstrong attended the SCIP-IV meeting and collaborative planning meeting. The SCIP project is an important source of data for outof-pile evaluation of high-burnup fuels and includes data on transient fission gas release, out-of-pile LOCA testing using furnaces and performs testing under a variety of conditions such as with small and standard size plenums to evaluate the effect of the amount of stored energy in the plenum on burst and dispersal behavior. Data from SCIP is being used extensively by both industry and NRC to support higher-burnup fuels and it is an important collaboration point regarding DOE's high burnup and LOCA testing program.

Jason Harp also attended the SCIP-IV program meeting. Harp is a designated ORNL representative to the SCIP. This project is complimentary to on-going DOE-NE sponsored work dedicated to understanding the fuel performance behavior of high burnup commercial LWR fuel. This meeting offers a chance for ORNL staff to interact with Studsvik staff to both learn the status of recent examinations and to also help shape the direction of future examinations based on traveler expertise and experience with performing similar examinations at ORNL hot cells. Supporting participation in this meeting will enhance U.S. high burnup and ATF expertise and visibility and allow the traveler to learn from the experience of other fuel performance experts from Studsvik who are performing complimentary experiments.

Working Group on Fuel Safety

Charles Folsom attended the NEA Working Group on Fuel Safety (WGFS) meeting in Boulogne-Billancourt, France March 6-11, 2023, to represent AFC's interest in the WGFS RIA database. Participation in the meeting allowed input of needs for the data being generated as part of the NEA FIDES High Burnup Experiments in Reactivity initiated Accidents Project.

Working Party for Fuels and Materials/Expert Groups on Fuel Materials

Paris, France: The NEA Working Party for Fuels and Materials (WPFM) was held in Paris, France. The WPFM of the NSC has been recently established with the objectives to connect modelling and simulation (M&S) analysis across different scales, to introduce machine learning approaches in the analysis, and to build closer links between M&S and experimental activities.

The WPFM and the two Expert Groups on Fuel Materials (EGFM) and on Structural Materials (EGSM), met on 23-25 May at the NEA for the first in-person WPFM week. More than 50 participants from 13 NEA member countries discussed the ongoing work and future activities. Joint sessions provided an opportunity to share best practices in nuclear materials microstructural characterization techniques, exchange views on cross-cutting activities within the WPFM, and further develop synergies with the NEA FIDES-II effort.

Other topics of interest to the group included materials ageing, materials acceleration platforms and highentropy alloys, artificial intelligence (AI) and machine learning (ML) for nuclear fuels, separate effect validation of fuel micro-mechanical models, thermodynamic data collection and gap analysis to identify needs for modelling and experiments on fuels and materials.

Dan Wachs, chair of the NEA WPFM, directed the meeting at NEA. As chair of the WPFM and co-chair of the FIDES Technical Advisory Group, Dr. Wachs participated and reported the status of efforts to the NEA NSC at NEA. Participation in the NEA working parties is critical to meeting U.S. nuclear technology development and deployment objectives.

Tarik Saleh attended the EGSM and associated meetings. He participated in a site visit at the French Alternative Energies and Atomic Energy Commission (CEA) Paris-Saclay to discuss the Matrix shipment. Saleh also presented an AFC-based advanced cladding roadmap at the EGSM.

Luca Capriotti attended the EGFM meeting (and ancillary meetings) as part of the newly formed Working Party on Materials Science Issues in Nuclear Fuels and Structural Materials. He attended the meeting as a U.S. representative. This group focused on collaborative projects/ activities with other NEA memberstates aimed at an effective partnership between experimental work and modelling efforts. Dr. Capriotti's presence was important to align the newly discussed activities with DOE interests in the different fields of nuclear fuel and modelling. Future data exchanged may be used to validate DOE fuel performance codes (e.g., MARMOT, BISON) in the future. This trip increased international cooperation and information exchange through several NEA working groups. Information exchanged in this meeting increased access to data from international fast reactor irradiations and access to codes used to simulate these experiments and new activities (such as, best practice in nuclear materials characterization). Data collected in this working group can also be used to benefit DOE simulation tools such as BISON by providing additional validation data sets.

Boone Beausoleil attended the EGSM and the EGFM. He presented on a joint US-Czech project to explore the use of additively manufactured high entropy alloys for nuclear applications to the EGFM. He presented a capabilities assessment on test reactors and PIE facilities available for fuel testing to the EGFM with a proposal to do a gap assessment on advanced fuels and testing. He also presented the advanced reactor fuels proposal for the FIDES second triennial to the EGFM and discussed what advanced fuel testing needs there are within the community.

Expert Group on Innovative Fuel Elements

Josh White participated in meetings in Paris, France March 27-28, 2023, as the U.S. technical expert on ceramic fuel properties. The focus of the Expert Group on Innovative Fuel Elements (EGIFE) meeting was for the group to work through the comments on the draft report on mixed oxide (MOX) and metallic fuel properties provided since April FY22 by a group of external reviewers. In addition to progressing towards finalization of the Properties Report, plans for extension of the Property group's work to other physical properties and additional fuel types were discussed. The EGIFE is a part of the Organisation for Economic Co-operation and Development (OECD)/NEA Nuclear Science Committee (NSC)'s Working Group on Scientific Issues of Advanced Fuel Cycles and brings together world leaders on the properties and performance of advanced nuclear fuels. There is significant benefit to the U.S. nuclear fuels programs through access to detailed property and performance information of advanced fuels well beyond what is available from historical U.S. efforts. This engagement was a very effective "multiplier" of U.S. investment in fuels for advanced fuel cycles especially considering that the U.S. funding for advance reactor fuels is currently low compared to past levels.

AFC staff attended the 6th OECD-NEA EGIFE at the OECD headquarters in Paris, France. The working group discussed reviewer comments for the draft report "Recommendations on fuel properties for fuel performance codes," which was completed and issued to the modeling round robin portion of the EGIFE. The working group also discussed the next topics of interest for the continuation of EGIFE and general areas of interest for each country involved.

International Atomic Energy Agency

Robert Armstrong attended the International Atomic Energy Agency (IAEA) Technical Meeting on Safety and Performance Aspects in the Development and Qualification of High Burnup Nuclear Fuels for Water-Cooled Reactors in Vienna, Austria, November 15-18, 2022. He presented the AFC Consensus TREAT/SATS LOCA Test Plan to an international audience from 21 member states and two international organizations. The presentation detailed the upcoming high burnup loss of coolant accident (LOCA) experiments to be performed at INL in TREAT and at Oak Ridge National Laboratory (ORNL) in Severe Accident Test Station (SATS), which aim to support light water reactor (LWR) fuel burnup extension needs by achieving an improved understanding of fuel fragmentation, relocation, and dispersal. This international collaboration proved to be incredibly valuable as it provided insights into research being performed in high burnup LWR fuel behavior all around the world.

AFC staff attended the IAEA Technical Working Group on Fuel Performance and Technology in Vienna,



Austria. Status and trends in nuclear power reactor fuel performance and technology were discussed. Other topics discussed were nuclear core materials research and development, fuel design, manufacturing and utilization, coolant chemistry, fuel performance analysis, and quality assurance issues. Due recognition was given to all relevant aspects including safety, economy, management systems, nuclear science, and nuclear power plant operations. This meeting provided opportunities to establish the DOEs leadership role in advanced fuel qualification (AFQ).

David Kamerman and Scott Holcombe traveled to Halden, Norway and Nykoping, Sweden for discussions on irradiation ramp testing capabilities that were performed at the Halden and R2 reactors.

The development of irradiation ramp testing capabilities is central to DOE's plans for irradiation testing of accident tolerant fuel (ATF). DiscusFigure 4. From left: Colby Jensen, Robert Hansen, Brian Durtschi, Shinichiro Yamashita (JAEA), David Kamerman, Yoshiyuki Kaji (JAEA), Daniel Wachs, Nicholas Woolstenhulme, Nozomu Murakami (Mitsubishi) sions ensured the experimental capabilities and test plans developed at INL expand on the work done historically at Halden and R2. This interaction helped confirm that DOE's investment in irradiation ramp test capabilities at INL sees the maximum technical benefit.

Japan Atomic Energy Agency

Individuals from the Japan Atomic Energy Agency (JAEA) and Mitsubishi Heavy Industries visited INL to discuss joint activities on irradiation testing and post irradiation examination (PIE) of ATFs. Visitors are project leads for developing and testing of ATF in Japan. They observed activities involving irradiation testing in the center flux trap of ATR and examinations of the test articles at the Materials and Fuels Complex. Hosting this group facilitates DOE and INL international leadership in these essential technical areas.

Bilateral Agreements

Department of Energy/ French Alternative Energies and Atomic Energy Commission

Tarik Saleh and Tsvetoslav Pavlov traveled to Bologna, Italy to attend the Plutonium Management for More Agility Workshop on Fuel Properties. This workshop fits into the French Alternative Energies and Atomic Energy Commission/ Department of Energy (CEA-DOE) bilateral scope and international fuel collaboration scope. Saleh is an expert on elastic properties of fuels and actinides and was invited by CEA colleagues to present at an international working group and workshop. The workshop hosted a mix of international expert and grad student talks covering techniques and results for property measurements of plutonium-bearing fuels. Dr. Saleh spoke on elastic moduli measurements of fuels via resonant ultrasound spectroscopy and presented results on moduli of mixed oxide (MOX) fuels. Dr. Pavlov spoke about thermophysical property measurements at INL including irradiated MOX data.

Civil Nuclear Energy Working Group

The 11th Meeting of the Advanced Fuels Area of the Civil Nuclear Energy Working Group (CNWG) Fuel Cycle and Waste Management (FCWM) sub-working group was hosted by the Japan Atomic Energy Agency (JAEA) in Mito, Japan. The meeting agenda included technical discussions on the status and path forward for the existing activity, facility tours, and discussions on potential expansion of the arrangement to include other fuel types of mutual interest.

This collaboration is an important part of the overall CNWG's efforts under the U.S.-Japan Bilateral Commission on Civil Nuclear Cooperation. There is significant benefit to the U.S. nuclear fuels programs through access to detailed property and performance information of advanced nuclear fuels well beyond what is available from historical U.S. efforts.

ADVANCED LWR FUELS / HIGH BURNUP

- 2.1 Fuel Fabrication and Properties
- 2.2 LWR Core Materials
- 2.3 LWR Irradiation Testing
- 2.4 LWR PIE
- 2.5 LWR Fuel Safety Testing
- 2.6 Performance Assessment
- 2.7 ATF Industry Advisory Committee
- 2.8 ATF Industry Teams

Thermal Properties of ATF Claddings

Principal Investigator: Tsvetoslav R. Pavlov Team Members/Collaborators: Lorenzo J. Vega Montoto, Michael C. Marshall

The new methods demonstrated for thermal expansion and total emissivity characterization of ATF cladding materials is critical for supporting the qualification of ATF cladding materials

hermal expansion and total hemispherical emissivity values are critical to enable licensing of new nuclear fuel and cladding materials. Thermal expansion is critical for establishing the dimensional changes of the cladding during irradiation while total emissivity dictates the radiative heat transfer at the external and internal cladding surfaces which influences the temperature profile across the fuel and the cladding. Both properties are critical for fuel performance codes to predict the thermo-mechanical behavior of novel fuel and cladding materials and to maintain necessary safety margins during reactor operation.

Project Description:

Capabilities at Idaho National Laboratory were further developed to enable the measurement of thermal expansion (using push-rod dilatometry) and total emissivity (using Laser Flash Analysis (LFA) and Fourier Transform Infrared (FTIR) Spectroscopy). For the first time, a new data analysis package was developed to enable the evaluation of total emissivity from laser flash data. Measurements were performed between room temperature and 1200C and the new methodologies were validated using available reference materials. Finally, the methodologies were applied to a range of accident tolerant fuel (ATF) cladding materials.

Accomplishments:

The current work could be summarized by the following goals:

- 1. Develop and establish methods for the measurement of:
 - a. thermal expansion of tubular cladding geometries in the radial and axial measurement directions from room temperature to 1200°C.
 - b. total emissivity from room temperature to 1200°C.
- 2. Deploy new methods/techniques for the characterization of ATF cladding specimen.

Thermal expansion of tubular samples was measured using a Netzsch push-rod dilatometer. A schematic of the equipment used is shown in Figure 1a and pictures of the post-test material (stainless steel type 316) are shown in Figure 1b. Stainless steel 316 tubes were used as surrogate cladding materials (instead of Zircaloy tubes). An example data set used to validate the methodology is shown in Figure 2. The modified thermal expansion method was subsequently applied to various ATF cladding tubes manufactured using different fabrication parameters. Temperature dependent thermal expansion results were collected in the axial and radial directions.

A laser flash setup (see Figure 3a) was used to determine the total emissivity as a function of tempera-



Figure 1. a) Schematic of the Netzsch push-rod dilatometer [1]; b) image of the post-test material (stainless steel type 316)



Figure 2. Thermal expansion of a stainless-steel type 316 tube as a function of temperature compared to the work of Kim [2]







Figure 4. Total hemispherical emissivity as a function of temperature measured using FTIR and LFA techniques. a) Comparison between measured values and reference range for carbon paint [4]; b) Comparison between measured values and reference range for oxidized INCONEL 600 alloy [5]

ture. A novel inverse method was developed to evaluate total emissivity from transient laser flash measurements (thermograms). The method consists of a least square minimization scheme between measured and theoretical thermograms (calculated via finite element analysis). The logic scheme of the method is described in Figure 3b. Another method known as FTIR was applied at lower temperatures (50°C -600°C). The equipment (see Figure 3c) consists of a Hyperion 3000, a high-resolution microscope interfaced with the FTIR Bruker VERTEX 70 spectrometer. Spectral emissivity was measured from 1.43 to 16.7 micrometers at every temperature step and subsequently integrated to obtain total hemispherical emissivity.

The measurements of total hemispherical emissivity via FTIR and LFA are shown in Figure 4. Figure 4a shows the results obtained from a carbon painted Pyroceram 9606 sample. The results are in very good agreement with the expected reference range [4]. The results on INCONEL 600 show good agreement with the wide reference range [5] shown in Figure 4b. The slightly higher results obtained using FTIR suggest the specimen used for the measurement was more oxidized compared to the sample measured via the LFA technique. Subsequently, measurements were performed on a series of ATF cladding specimens produced using different fabrication parameters.

References:

[1.] J. E. Daw, J. L. Rempe, D. L. Knudson, K. G. Condie and J. C. Crepeau, "Viability of pushrod dilatometry techniques for high temperature in-pile measurements," Idaho National Laboratory, Idaho Falls, 2008.

- [2.] C. S. Kim, "Thermophysical properties of stainless steels," Argonne National Laboraotry, Argonne, 1975.
- [3.] B. Cheng, B. Lane, J. Whiting and J. Chou, "A combined experimental-numerical method to evaluate powder thermal properties in laser powder bed fusion," Journal of Manufacturing Science and Engineering, vol. 140, pp. 1-8, 2018.
- [4.] R. Brandt, C. Bird and G. Neuer, "Emissivity reference paints for high temperature applications," Measurement, pp. 731-736, 2008.
- [5.] R. Shurtz, "Total hemispherical emissivity of metals applicable to radiant heat testing," Sandia National Laboratories, Albuquerque, 2018.

Summary of Laser Flash Analysis measurements on Cr-coated cladding

Principal Investigator: Weicheng Zhong

Team Members/Collaborators: Hsin Wang, Amy Godfrey, Nathan Capps, Andrew Nelson

ransfer of heat generated by fission within a nuclear reactor fuel to the coolant governs normal operation and thermal response to possible transients. Long term service of Cr coated Zircaloy in light water reactors (LWRs) leads to microstructure evolution of coating and the growth of Laves phase from the interaction of coating and substrate. The impact of such microstructure evolution and the Laves phase formation on the heat transport property were unknown and they are investigated in this project.

Project Description:

The goal of this project is to investigate the thermal transport property and to characterize the interaction of the coating/substrate in Cr coated Zircaloy system after the thermal aging condition. Nuclear fuel cladding with superior accident tolerant properties is demanded for current LWR fleets, and Cr coated Zircalov is the primary candidate of the coated cladding concepts. The impact on the thermal transport property from the Cr coating and the Laves phase laver must be examined and understood. This project investigates Cr coatings of two different deposition techniques: High Power Impulse Magnetron Sputtering (HiPIMS) and cold spray techniques. Comparing different coatings allow one to understand the effects of deposition methods on the microstructure and thermal diffusivity property. Thermal aging was performed on Cr coated Zircaloy at 750°C in inert environments. Thermal diffusivity was measured on the as received and thermally aged samples using the laser flash

Figure 1. Cross section of asdeposited a,c) HiPIMS and b,d) cold spray Cr coatings on Zircaloy



2023 AFC ACCOMPLISHMENTS
This project provides critical information for thermal transport property modeling by examining the thermal diffusivity and the microstructure of Cr coated Zircaloy after thermal aging condition.





analysis techniques. To better understand the thermal diffusivity results and the interaction of Cr coating with Zircaloy substrate, multi-scale microstructure characterization was performed for the microstructure of the coating and Laves phase formation, primarily using scanning electron microscopy, electron backscatter diffraction and transmission electron microscopy (SEM).

Accomplishments

The microstructure of the as-deposited Cr coatings was characterized



Figure 3. Cross section of HiPIMS and cold spray Cr coating after thermal aging at 750°C. Larger voids were observed at the interface or within the coatings using SEM, as shown in Figure 1. The HiPIMS Cr coating demonstrates a dense microstructure, with no observable porosity in the coatings. The cold spray Cr coating exhibits some larger pores (up to ~1 μ m) and crack-like defects, with some examples labelled in Figure 1b. Smaller pores in Cr coatings are also observed in higher magnification image in Figure 1d. Such porosity could be attributed to the high supersonic velocities of powder particles that was deposited on the substrate surface, which could cause imperfect bonding.

As-deposited Cr coatings has minor effects on the thermal diffusivity of Zr cladding. Figure 2 shows the temperature-dependent thermal diffusivity of as-deposited Cr coated Zircaloy up to 700°C, in comparison to uncoated Zircaloy. Three measurements were performed on the cold spray Cr coated Zircaloy, and the results are consistent, with the variation below ~2-5%. HiPIMS Cr coated Zircaloy have similar thermal diffusivity as uncoated Zircaloy, and cold spray Cr coated Zircaloy shows the reduced average thermal diffusivity by ~5-7% than the uncoated Zircaloy. The slight reduction of thermal diffusivity in cold spray Cr coated Zircaloy is not attributed to its thicker Cr coatings due to the higher thermal diffusivity of pure Cr than Zr but could be attributed to the porosity in the cold spray Cr coating as seen in Figure 1.

The microstructure of Cr coated Zircaloy after thermal aging at 750°C was characterized. Figure 3 shows the microstructure of the HiPIMS and cold spray Cr coated Zircaloy after the aging for 2800 hours at 750°C. HIPIMS Cr coatings has maintained the dense microstructure with no coating delamination. Voids are observed at the coating/substrate interface. The formation of such voids is likely due to the counter flow of vacancy in response to mass transport across an interface (in this case Cr diffusion to





Figure 4. Scanning transmission electron microscopy images of spray Cr-coated Zircaloy after thermal aging at 750°C

Zircaloy) based on the inverse Kirkendall effect. For cold spray Cr coatings, pores are more obvious inside the coating after the thermal aging, and they are preferentially distributed at grain boundaries of Cr coatings. In addition, a layer of the Laves phase formed because of the interaction of Cr coating and Zircaloy. Such Laves phase layer is consist of fine grain structure near the Cr coatings and the coarse grain near the Zr substrate, as the transmission electron microscopy image shown in Figure 4.

Importantly, the formation of Laves phase layer and the larger voids formation at the interface or within the coatings do not have significant impact on the thermal diffusivity. Figure 5 shows the thermal diffusivity of Cr coated Zircaloy after thermal ageing up to 2800 hours at 750°C, and very minor or no change of thermal diffusivity was observed after the thermal aging for both coatings.



Figure 5. Thermal diffusivity of as-deposited and aged (left) HiPIMS and (right) cold-spray Cr coated Zircaloy up to 700°C

Handbook on Accident Tolerant Fuel Doped UO₂ Properties

Principal Investigator: Adrien J. Terricabras

Team Members/Collaborators: Sean Drewry, Maria Kosmidou, Darrin Byler, Joshua T. White

oped UO_2 fuels are an industrial relevant Accident Tolerant Fuel (ATF) owing to the ability of this class of fuels to offset fission gas release into the plenum as well as purported thermomechanical benefits to minimize pellet clad mechanical interactions. Work this year completed efforts to summarize a suite of thermophysical properties and microstructural characterizations on Cr-doped UO₂. This work is summarized in a non-proprietary handbook of interest to the Nuclear Advanced Modeling and Simulation community as well as the Nuclear Regulatory Commission. Work described herein details the initial draft of a handbook on near term ATF doped UO₂ fuel starting with the fresh fuel characterization which is planned to be enhanced with follow on post irradiation examination (PIE) studies and transient/accident tests.

Project Description:

Due to the proprietary nature of ATF datasets, this project was initiated to create an open database for regulatory entities to verify models being developed by industry partners in support of their ATF qualification campaigns. For the purposes of this project, Cr-doped UO₂ was chosen as there is relevance to multiple industry teams. The Cr-dopant is added to promote grain growth in UO₂ which is purported to improve

the fission gas release behavior as well as enhance the creep behavior of UO₂. However, many of the thermophysical properties are sparse in the literature and have a wide scatter for what has been published. Furthermore, many of the published literature do not accurately report the Cr concentration of the as fabricated pellets, making comparison across multiple sources difficult to ascertain conclusions. This is further compounded by the proprietary nature of the fabrication routes from industry which have now been emulated at Los Alamos National Laboratory (LANL) to produce reasonable grain sizes like what is available in the literature to provide confidence in the structureprocessing-property-performance relations. The above datasets have been compiled into the initial draft handbook on doped UO₂ ATF, which is expected to be expanded upon in future FYs, summarizing on-going efforts within the Advanced Fuels Campaign (AFC) at PIE facilities as well as irradiations at test reactors.

Accomplishments:

The work supported under this work package has resulted in two milestone reports: M3FT-23LA0202010110 (submitted May 2023) and M2FT-23LA020201011 (submitted September 2023). The work accomplished here summarizes new and existing experimental



Figure 1. Thermodynamic equilibrium between metallic Cr and Cr oxides, stability domain of the CrO(I) composition and sintering conditions from literature

Figure 2. Measured Cr₂O₃ content in sintered pellets vs initial added amounts before sintering from literature. Drawn lines to guide the eye are representing 0 % volatility (filled line) and 50 % volatility (dashed line)



Figure 3. Coefficient of thermal expansion as a function of temperature in undoped and Cr-doped UO₂

results that have been generated in the AFC and compiles data published in peer-reviewed journals associated with the properties of Cr-doped UO₂. Results, direct comparisons with undoped UO₂, and recommendations were compiled in a non-proprietary handbook. Properties determination and comparisons throughout the handbook were made based on Cr_2O_3 doped UO₂ samples (750 ppm to 7800 ppm additive) sintered at LANL.

Sintering tests combined with solubility limit and chromium incorporation data in UO₂ allowed for recommendations to be made on the fabrication of Cr-doped UO₂. Sintering above 1700°C with appropriate oxygen potential for the sintering to occur in the CrO₍₁₎ phase prevented early Cr/Cr₂O₃ precipitation as shown in Figure 1. Inductively Coupled Plasma **Optical Emission Spectroscopy** measurements showed consistent Cr volatility of ~ 48 % during sintering, in agreement with recent literature reports. These results can be seen in Figure 2. The current proposed solubility limit of Cr in UO_2 is ~ 700 to 1200 ppm Cr in UO₂ (~ 1000 to 1750 ppm Cr₂O₃ equivalent).

Lattice parameter contracted with increasing Cr content up to the solubility limit, indicating an assimilation in the UO₂ lattice as opposed to a secondary phase formation. No significant variations were observed above. Sintering ATF handbooks, like the one prepared here, provide a critical source of non-proprietary data to evaluate fuel performance metrics and validate models being developed across the nuclear enterprise.

conditions affected the final grain sizes, especially when using CO_2 mixtures. This work reports a maximum average grain size of 52 ± 8 mm obtained after initially doping UO₂ with 4800 ppm Cr₂O₃ and using a wet Ar/H₂ atmosphere.

Thermophysical properties determination of Cr-doped UO₂, including specific heat capacity, linear thermal expansion, coefficient of thermal expansion, and melting temperature showed no appreciable differences compared to undoped UO₂. Coefficient of thermal expansion as a function of temperature can be seen in Figure 3. Thermal diffusivity (and thermal conductivity) presented a small decrease compared to undoped values above 2500 ppm Cr_2O_3 in UO_2 , which was associated with the higher defect concentration in the material generated by the Cr additive.

Discussion on the lack of published work on the mechanical properties as well as PIE on Cr-doped UO_2 showcased the current gap surrounding this topic and revealed future work opportunities.

2.2 LWR CORE MATERIALS

Initial Deployment of MiniFuel in the Removable Beryllium Reflector

Principal Investigator: Jacob Gorton

Team Members/Collaborators: Annabelle Le Coq, Zane Wallen, Christian Petrie, Joshua White, Nathan Capps, Kory Linton

he MiniFuel experimental platform at ORNL has been used for separate-effects nuclear fuel irradiations in the High Flux Isotope Reactor (HFIR) since 2017 [1]. The initial capability was qualified for deployment in HFIR's vertical experiment facilities (VXF), located roughly 20 cm from the outermost fuel element, which enabled irradiations with fuel specimen temperatures typically ranging from 500°C to ~700°C and up to ~1000°C in some cases [2]. MiniFuel experiments conducted in HFIR's removable beryllium (RB) reflector would enable fuel temperatures >1000°C to be achieved with faster fuel specimen burnup accumulation compared with VXF experiments because of the closer proximity to the driver fuel elements. These conditions are better suited to studying critical fuel performance phenomena, such as fission gas release (FGR). Design changes to the MiniFuel experimental architecture to enable deployment in the RB reflector were documented in a fiscal year (FY) 2021 AFC milestone report [3]. This report summarizes the FY 2023 accomplishments pertaining to RB MiniFuel, which include safety qualifying the experiment for HFIR insertion, executing

design calculations for the initial experiment containing Cr-doped UO₂ fuel samples, and insertion of the first RB MiniFuel experiment into HFIR on July 5, 2023.

Project Description:

The overarching objective of the RB MiniFuel irradiation capability is to study critical fuel performance phenomena at high temperatures and burnups. Cr-doped UO₂ was used in the first RB MiniFuel experiment, and the primary objective was to investigate the impact of Cr content and UO₂ grain size on FGR. Two UO₂ grain sizes and two Cr concentrations (750 ppm and 2500 ppm) were considered for the fuel specimens. The experiment was designed to provide time-average, volume-average (TAVA) fuel temperatures between 1075°C and 1100°C for each of the 12 fuel specimens included in the experiment. Desired fuel temperatures are met by sizing the outer diameter (OD) of each subcapsule (thus changing the gas gap between each subcapsule and the target inner wall) and by selecting the target fill gas mixture. Figure 1 shows a schematic of the RB MiniFuel platform along with component designations. The specimens in one MiniFuel target



Figure 1. Schematic of the RB MiniFuel experimental architecture

Figure 2. An example of timedependent fuel specimen temperatures during an RB MiniFuel experiment (markers indicate average temperature and error bars represent minimum-tomaximum temperature range)



Figure 3. Subcapsule components prior to assembly

containing 6 of the 12 specimens are expected to reach burnups between 45–50 MWd/kg-U and will remain in HFIR for 5 irradiation cycles or approximately 125 irradiation days. The second target containing the other half of specimens will remain in HFIR for 9 irradiation cycles, or roughly 225 irradiation days, allowing the fuel specimens to reach burnups between 86–96 MWd/kg-U. The two burnup ranges were selected to provide some insight to the relationship between FGR and burnup at approximately constant temperatures.

Accomplishments:

The initial RB MiniFuel experiment using Cr-doped UO₂ fuel specimens was inserted into HFIR in FY 2023. This effort included design of the subcapsule outer diameters and target fill gas compositions, performing extensive safety calculations to gain approval for HFIR insertion, experiment procurement and assembly, and installation. Design calculations were performed by first determining time-dependent heat generation rates (HGRs) in experimental components using neutronics tools and then importing



Figure 4. Layout of MiniFuel target assembly



Figure 5. MiniFuel irradiation vehicle assembly

those HGRs into a finite-element thermal model of the experiment. Adjustments to the subcapsule ODs and the target fill-gas were made iteratively until the desired TAVA temperatures were met in each subcapsule. Fuel temperatures during a single HFIR cycle are expected to vary by 150–200°C, although approximately 100-125°C of that variation occurs over the first two days of each cycle as HFIR's control drives are adjusted. An example of the time-dependent temperature predicted for one of the fuel specimens is shown in Figure 2.

Fuel specimens for the experiment were provided by Los Alamos National Laboratory (LANL). Once received at ORNL, each of the fueled subcapsules was assembled. Each subcapsule assembly comprised a fuel specimen, cup, heater tube, SiC thermometry, grafoil insulators, and a subcapsule holder and endcap. The cup, tube, holder, and holder endcap were all comprised of low-carbon arc-cast Mo. An example of the subcapsule components laid out prior to subcapsule assembly is shown in Figure 3. Once assembled, each subcapsule underwent an electron beam welding process followed by a tungsten inert gas seal welding process in a helium back-filled chamber. The bottom target endcap was orbital welded to the main housing tube, and then the stack

The RB MiniFuel platform enables high-temperature irradiation testing with rapid burnup accumulation, which will be key for separate-effects fuel performance experiments under reactor-relevant conditions.

of subcapsules, centering thimbles, and end springs were loaded into the target (Figure 4). Another orbital weld was performed in the desired fill gas environment to seal the target. Leak tests were performed at each seal welding step, and a final non-destructive radiograph was taken to ensure sufficient weld penetration. The targets were then loaded into the Al MiniFuel basket in their desired radial and axial positions, with Al dummy targets filling the remaining open positions (Figure 5). The entire irradiation vehicle was lowered into an RB facility on July 5, 2023, prior to the start of HFIR's 502nd irradiation cycle, which commenced on July 18, 2023.

References:

[1.] Petrie, C.M., Burns, J.R., Raftery, A.M., Nelson, A.T. and Terrani, K.A., 2019. Separate effects irradiation testing of miniature fuel specimens. Journal of Nuclear Materials, 526, p.151783.

- [2.] Gorton, J.P., Gallagher, R.C., Wallen, Z.G., Le Coq, A.G., Helmreich, G.W., Petrie, C.M., Linton, K.D., Latta, R. and Gerczak, T.J., 2023. Simulation of a TRISO MiniFuel irradiation experiment with data-informed uncertainty quantification. Nuclear Engineering and Design, 404, p.112177.
- [3.] Gorton, J.P., Wallen, Z.G. and Petrie, C.M., 2021. Modifications to MiniFuel Vehicle Needed to Enable Higher Temperature UO2 Irradiation Capabilities (Summarizing Issue Report) (No. ORNL/SPR-2021/2096). Oak Ridge National Lab (ORNL), Oak Ridge, TN (United States).

FeCrAl Optimization and Corrosion Results

Principal Investigator: Caleb Massey Team Members/Collaborators: Mackenzie Ridley, Yukinori Yamamoto, David Hoelzer

> his work involves optimization of FeCrAl alloys as nuclear reactor materials in regard to both corrosion and irradiation responses. Historic FeCrAl alloys contain high Cr contents, which leads to precipitation of an alpha prime phase during both thermal ageing and exposure to radiation, which can produce undesirable mechanical property effects. Oak Ridge National Laboratory (ORNL) manufactured multiple FeCrAl alloys with variations in Cr content from 12 wt. % to 17 wt. % while maintaining a constant Al content. The goal was to understand the minimum Cr content that would result in acceptable oxide thicknesses after long term exposure at prototypic reactor conditions, and how the varied Cr content impacted mechanical properties. Current progress shown here includes the initial oxidation behavior of FeCrAl alloys after exposure to hydrogen water chemistry under light water reactor (LWR) conditions. Hightemperature tensile testing was also performed to support FeCrAl chemistry optimization.

Project Description:

The technical objectives of this research include (1) the development of FeCrAl alloys with systematically varied chemistries, (2) mechanical testing to establish potential tradeoffs in alloy mechanical performance if compositional changes are pursued, and (3) autoclave testing to illustrate the effect of Cr content on the corrosion performance of FeCrAl alloys in LWR environments. These three goals are critical to the state-of-theart knowledge on the FeCrAl alloy system because the model alloy C26M (Fe-12Cr-6Al wt%) was down-selected primarily based on factors such as fabricability, high-temperature steam oxidation performance, and to minimize Fe-Cr phase separation during irradiation and/or thermal ageing. Thus, corrosion was not considered in this initial down-selection framework. Consequently, this work aims to fill this critical gap regarding the effect of Cr on the performance of the FeCrAl alloy system under normal operating conditions, thereby pushing forward this alloy concept as a potential drop-in replacement for existing Zr-based alloys and enhancing the safety of the existing LWR fleet.

Accomplishments

Multiple FeCrAl alloys were forged at ORNL to assess the impacts of varied Cr content on oxidation and hightemperature mechanical performances. The Al content in these alloys was maintained at 6 wt. % such that the impacts of varied Cr content could be best visualized during corrosion testing. Due to the dissolution of Cr in boiling water reactor chemistries,



pressurized water reactor conditions were chosen as an initial study to verify oxide formation kinetics, phase stabilities, and microstructural evolution of layered oxide systems. Oxidation studies in pressurized water at 330°C and 15.3 MPa have been performed up to 1500h of exposure, identifying clear chemistry impacts on the corrosion thickness of each alloy. Low Cr-content FeCrAl alloys showed formation of a continuous and dense Cr/Al mixed oxide with Fe oxide nodules on the outer surfaces after 500h of exposure. This was verified with energy dispersive Figure 1. Scanning transmission electron microscopy energy dispersion x-ray elemental mapping and elemental linescan through the corrosion layer of C36M4 after 500 hours of exposure to 330°C and 15.3 MPa with hydrogen water chemistry



Figure 2. Median oxide thickness as a function of time for each FeCrAl alloy during autoclave exposure in prototypic LWR conditions with hydrogen water chemistry

52

Optimization of FeCrAl chemistries will enable use of FeCrAl as an accident tolerant fuel cladding material to improve both reactor safety and cycle efficiencies.

spectroscopy utilizing a scanning tunnelling electron microscope (See Figure 1). High Cr content alloys did not form a continuous Cr/Fe mixed oxide layer until 1000h or 1500h of exposure, representing enhanced corrosion resistance with increasing Cr content. Additions of yttrium were shown to improve corrosion resistance in comparable alloy studies (See Figure 2). Fe oxide nodules were not as prevalent on outer surfaces of high Cr content alloys, likely indicating that the nodule formation is reliant on the formation of a stable Cr/Al mixed oxide scale underneath. Uniaxial tension testing showed that low Cr FeCrAl alloys had similar yield strengths as high Cr FeCrAl alloys up to 800°C. Yet, all FeCrAl variants that contained yttrium additions exhibited premature failure before reaching the ultimate tensile strength and ultimate elongation. While yttrium additions have been shown to enhance corrosion resistance, mechanical performance needs to

also be considered when optimizing alloy chemistry. The current oxidation results up to 1500h show that a Cr content greater than 13 wt. % is likely required for improved corrosion resistance with minimal impact on mechanical properties. This project will continue oxidation testing past 1 year of total exposure for a realistic understanding of corrosion kinetics.

Mechanical Properties of ATF Cladding

Principal Investigator: Benjamin Eftink and Robert Hansen

Team Members/Collaborators: Hyosim Kim, Nan Li, Philip Peterson, Katelyn Baird, Peter Beck, Mathew Hayne, Carl Cady, Tim Graening, Tyler Dabney (University of Wisconsin Madison: UW-Madison), Kumar Sridharan (UW-Madison), Tarik Saleh, Fabiola Cappia, Jake Stockwell, David Kamerman

This project has developed mechanical testing protocols for testing hoop direction properties at high temperatures and in hot cells, as well as evaluating coating adhesion on newly developed fuel cladding concepts.

ccident tolerant fuel (ATF) cladding concepts aim to improve high temperature steam corrosion resistance for instances of off-normal conditions in commercial light water reactors (LWR). One proposed concept is the addition of a Cr coating on Zr based cladding tubes. These materials are being studied for early implementation in LWRs. Understanding how the coatings respond to mechanical stresses and strains, particularly in the hoop direction of the tube, is necessary before wide implementation. Since the hoop direction is the direction that sees the greatest stress in real world situations, it is necessary to test the hoop direction directly. Additionally, determining the adherence of the coating is important to understand. Ouantifying mechanical properties of ATF cladding concepts at service relevant temperatures and in hot cell environments adds complications to testing and was the focus of FY23 efforts for this project.

Project Description:

This project had three main technical objectives: i) high temperature hoop direction strength of Cr coated Zircaloy, ii) implementation of hoop strength testing in hot cells, and iii) measure high temperature adherence of the Cr coating applied to Zircaloy tubes. Implementing accident tolerant cladding concepts in the next few years requires confidence in the cladding's mechanical performance. Coatings applied by different deposition parameters/ methods show different responses to applied stresses/strains. By mechanically testing these cladding tubes, we are understanding better how they will hold up in service conditions including high mechanical strain (due to internal gas pressure) accident scenarios. Extending capabilities to hot cells also allows testing of irradiated cladding, whether from test reactors or commercial fuel rods, directly enabling qualification of commercial ATF cladding concepts. This directly supports the Department of Energy objectives of safe, reliable, and thereby economical operation of the nation's current reactor fleet.

Accomplishments:

The first project goal of determining the high temperature strength of Cr coated Zircaloy was achieved by ring pull testing of Cr coated Zircaloy tubes at Los Alamos National Laboratory (LANL). The tubes were coated by different processing conditions as well as coating methods. Coating methods included high impulse magnetron sputtering (HiPIMS) coated tubes provided by Oak Ridge National Laboratory (ORNL) as well



Figure 1. (a) Ring pull test fixture. TZM was used for the material to withstand high temperature testing. The fixture has interchangeable mandrel inserts to facilitate testing of rings from 5 to 20 mm ID. (b) Ring pull high temperature test setup inside of a Centorr load frame vacuum furnace

as cold spray coated tubes from UW-Madison. Using ring pull testing at room and elevated temperature, differences in mechanical strength were compared between the samples.

This year, one of the focuses was to implement high temperature ring pull testing using a newly designed fixture. The fixture was machined out of molybdenum alloy titanium zirconium molybdenum (TZM) and has interchangeable mandrels for accommodating rings of different diameters. The fixture was designed to be compatible with multiple load frames, and ring pull testing can be performed at temperatures up to 800°C. Figure 1 shows (a) the test fixture and (b) setup in a load frame with vacuum furnace.

Ring pull tension testing at room temperature, 300 and 600°C was performed for three sample types: i) Zircaloy-4, ii) Zircaloy-4 with a Cr coating applied with HiPIMS Physical Vapor Deposition and iii) Zircaloy-4 with a Cr coating applied with cold spray deposition. The control material was from the same tube as the cold spray coated samples, while the HiPIMS coated Zircaloy-4 tubes were different. Figure 2 shows the effective stress-strain curves at (a) room temperature, (b) 300°C and (c) 600°C. Tests indicate that the Cr coating increases high temperature strength and has less of an impact on strength at room temperature.

The second goal focused on in-cell implementation of hoop direction

mechanical testing. Another version of the ring pull test setup was qualified in the hot cell at INL. The grip was made of 17-4 PH Stainless Steel, with interchangeable mandrels and design features to enhance remote handling and was used with the remote load frame in the hot cell as shown in Figure 3(a). Single-gauge



Figure 2. Effective stress-strain curves for ring pull testing at (a) room temperature (b) 300°C and (c) 600°C. The numbers after the sample names indicates the specific ring tested



Figure 3. (a) INL version of the ring pull test, placed in the hot cell and compatible with clamshell furnace (not pictured). (b) A single-gauge cladding ring after testing to failure. (c) Stress-strain curves of uncoated and cold-spray coated cladding at room and elevated temperature



Figure 4. (a) The load-displacement curves of five cantilever beam bending tests. (b) - (f) are the scanning electron microscopy images to show the plastic deformation of five beams, respectively

unirradiated rings from uncoated zircaloy and from cold-sprayed cladding were tested to failure, seen in Figure 3(b). Although these were unirradiated, in-cell milling was also qualified to support future mechanical tests of irradiated cladding. Testing was performed at both room temperature and 350°C, with results shown in Figure 3(c). The cold-spray cladding showed similar performance to the out-of-cell testing from LANL, with comparable ultimate tensile strengths.

The third goal of the project was to determine the mechanical integrity of the coatings at elevated temperatures. For this, micro-cantilever testing was performed at 300°C at LANL. For both the cold spray and HiPIMS applied Cr coatings, adhesion was robust. Significant plastic deformation was observed in the Zircaloy rather than the Cr coatings or interface between the two. Figure 4 shows for the 300°C micro-cantilever tests of the cold spray coated samples (a) load-displacement curves and (b)-(f) cantilever beams at the end of each test.

Advanced Characterization of SiC Subjected to Transient Irradiation

Principal Investigator: Fei Xu Team Members/Collaborators: Tiankai Yao, Mario D. Matos II, Jason L. Schulthess, Peng Xu

> This project aims to develop a comprehensive characterization dataset for silicon carbide (SiC) cladding before and after Transient Reactor Test Facility (TREAT) experiments to resolve multi-length scale defects such as porosity and cracks to inform modeling and simulations.

Project Description:

SiC ceramic matrix composite (CMC) cladding is currently being pursued as one of the leading candidates for accident-tolerant fuel cladding for light water reactor application. SiC CMCs have excellent high-temperature oxidation properties, superior irradiation resistance, inherent low activation, and other superior physical/chemical properties. However,



Figure 1. The proposed workflow. (a) An image from the fresh sample with field of view representing 3024×3064 with resolution 6µm/pixel; (b) Sub-image containing material annulus; (c)-(f) Images represent a spatially transformed coordinate system showing from top to bottom, a binary segmentation of the annulus, original grayscale reconstruction, segmented voids, and color-coded pore classifications. In (f), green denotes the voids/cracks connected directly to the outer surface of the annulus; blue identifies closed voids/cracks that do not touch the inner or outer surface of the annulus; yellow identifies voids/cracks touching the inner and outer surfaces. (g) Rewrapped image frame with corresponding classifications from (f)



SiC exhibits nonlinear damaging mechanical behavior governed by microcracking within the material under different conditions, such as the stress-state induced by irradiation, and pellet cladding mechanical interaction (PCMI) under accident conditions. Morphology of defects, including size, shape of voids, is one of the key factors which impact cladding performance and guarantees reactor safety. Therefore, quantification of defects' size, location, distribution, and leak paths are critical to determining SiC cladding in-core performance. This project provided qualitative insight into the defect's distribution under multi-scale characterization before and after different TREAT irradiation tests. A non-destructive evaluation technique based on 3D X-ray imaging help to assess critical microstructural defects from manufacturing or PCMI testing. Following the X-ray Computed Tomography (XCT) data analysis and visualization,

advanced microstructure characterizations using electron microscopy techniques such as Focused Ion Beam (FIB)/Scanning Electron Microscopy (SEM) and Transmission Electron Microscopy (TEM) were conducted. This complements the findings from the XCT data, in particular quantification of nano-porosity and microcracks. In this project, three SiC samples (two irradiated, noted as Sample 1 and Sample 2, and one unirradiated as Sample 3) provided by General Atomics are investigated. The automated workflow developed in this work enabled identification of the defects in the irradiated samples that were not present in the unirradiated sample. The statistics of the defects and characteristics can inform the mechanical responses of SiC cladding in accidents and can be later correlated to manufacturing for processing and performance improvement, as well as modeling and simulations.

Figure 2. 3D rendering result of Sample 1

The multiscale characterization dataset for defects in SiC CMCs and the statistical method enhance the predictable modeling capability of SiC fracture behavior under irradiation and dynamic loads.



Figure 3. Voids structure in Sample 1, Sample 2 and Sample 3

Accomplishments:

Multi-scale characterization of SiC porosity and cracks is accomplished, including 3D X-ray global-scale, FIB/ TEM local cross-sectional micro-scale and local nano-scale TEM 3D tomography. This work demonstrated the applicability of X-ray tomographic imaging capability in rapidly supporting post-irradiation examinations of non-fuel components. Critical quantitative global information of the material will be obtained to evaluate the material condition and performance and eventually assist the decision-maker to determine the material lifespan. The current

X-ray instrument capability at Idaho National Laboratory (INL) can provide the image quality with length scales from 10 cm to 10⁻² cm, however, the crack or pore less than $35 \,\mu m^2$ (5 pixels) cannot be detected using this capability. This study bridges multiscale characterization results by using 3D tomography Scanning Transmission Electron Microscopy (STEM) to validate 3D XCT results and provide more confident qualitative information under microstructure levels. This is the first work to fuse multi-scale characterization data of defects and cracks in SiC CMC cladding after transient safety testing at INL.

Engineering-scale Defect Characterization using 3D X-ray Imaging

In this task, we developed a fully automatic workflow to detect and analyze defects using image processing techniques on 3D X-ray images (Figure 1). Void/crack detection, visualization, and analysis are the three major components. The results were shown in Figures 2-3.

FIB/SEM Characterization

Higher resolution characterization was conducted using dual beam FEI Helios Plasma Focused Ion Beam on cross-sections of Sample 1 and



Figure 4. Defects verification from FIB characterization of Sample 1

2023 AFC ACCOMPLISHMENTS



Sample 2. As shown in Figures 4 and 5, multi-scale Backscattered Electron images were collected to confirm the leaking cracks' existing on Sample 1 and Sample 2. Moreover, radial, circumferential, and secondary cracks were observed from the two samples.

High Resolution TEM Characterization

3D STEM tomography was used to characterize nano-porosity and microcracks. One TEM sample $(3.26\mu$ m× 3.26μ m× 0.32μ m) was lifted out from Sample 1 and 3D STEM data was generated using Avizo software. The 3D STEM data was collected from in-situ TEM sample with tilting -50 degree to 50 degree. After acquiring the images, aligning and reconstruction steps were utilized to obtain the 3D tomography data. The defects were detected using image processing techniques, and results were shown in Figure 6. Figure 6(e) shows the rendering result of the defects. In this TEM region, the volume porosity fraction was determined to be 5.83 vol %, and the primary crack size was found to be 4.34μ m× 2.41μ m× 0.29μ m. A secondary crack was also observed.

Figure 5. Defects verification from FIB characterization of Sample 2



Figure 6. TEM Sample 3D tomography

Thermal Conductivity Measurement of Fresh SiC CMC

Principal Investigator: Cynthia Adkins

Team Members/Collaborators: Tsvetoslav Pavlov, Zilong Hua, Ethan Hisle, Michael C. Marshall, Peng Xu

his is a method development project for thermal conductivity measurement on an unfueled silicon carbide fiber, silicon carbide matrix (SiC-SiC) composite cladding system where surrogate fuel material is Tin-bonded within the cladding. This project scope focuses on as fabricated material, however the data collected, and analysis methods used will be directly applicable to the post-irradiated rodlets that are being irradiated in the Advanced Test Reactor. The goal of this testing is to capture important material properties and cladding component performance data on rodlets provided by General Atomics Electromagnetic Systems (GA-EMS). The data collected will be used to inform GA-EMS' SiC-SiC cladding design.

Project Description:

GA-EMS has put forth a SiC accident tolerant fuel concept where a SiC-SiC composite cladding tube is used to encapsulate UO_2 rodlets that are bonded to the inside of the cladding tube using liquid Sn. This fuel/cladding design needs comprehensive characterization of thermophysical and mechanical properties before and after irradiation testing. Accurate thermo-mechanical property data is of paramount importance for fuel performance codes to predict the fuel rod's behavior during normal and off-normal reactor operating conditions. This study focuses on the thermal conductivity of the cladding and Sn bonding layers of the total fuel design at room temperature through the melting temperature of Sn. It is important to determine any effect on thermal conductivity due to the presence of Sn both solid and liquid as well as the independent thermal conductivity of the SiC-SiC composite material, including the individual components of the composite. In this study the UO₂ fuel rodlet is replaced with a surrogate material, molybdenum metal.

Pulse laser flash analysis (LFA) and modulated thermoreflectance using a thermal conductivity microscope (TCM) are used to measure the thermal diffusivity radially across the cross-section of the SiC-SiC composite alone and the SiC-SiC/Sn/ Mo surrogate fuel system, in both the bulk and micro scale, respectively. Thermal diffusivity is directly related to thermal conductivity, an important parameter for characterizing and modeling heat transfer performance of materials. Further method development is needed to correct for the curvature of the cladding in the LFA and the surface topography within the SiC-SiC composite. Likewise, for the TCM method, further develop-



ment is needed at the SiC cladding / Sn interface to maintain accuracy and quantify uncertainty in the measurements, as well as to obtain measurements near the melting temperature of Sn.

Accomplishments:

Identification of the effect on thermal conductivity of the SiC-SiC composite from the Sn layer and possibly the presence of UO₂ firstly, requires an understanding of the thermal conductivity of the cladding tube material itself and secondly how the curvature of the cladding tube / fuel material impacts the thermal transport model used to calculate thermal diffusivity and conductivity from the instrument data due to the irregular geometry. Understanding of these effects is the technical goal of this study.

A specimen of the SiC-SiC cladding material was cut from the side of the cladding tube and placed perpendicular to the laser beam in the LFA instrument to measure thermal diffusivity across the radial direction of the tube (obtained by Pavlov and Marshall). Figure 1 shows the resulting bulk thermal diffusivity vs. temperature of the cladding material up to 1000°C. Additionally, Figure 1 also gives a graph of the relative Figure 1. (a) Thermal diffusivity(α) of SiC-SiC composite via LFA (b) relative error in α . Red curve indicates the laser heated the inside surface and blue curve indicates the laser heated the outside surface of the cladding



Figure 2. Cross-section area of SiC-SiC cladding where local thermal diffusivity measurements were collected. Composite features are labeled in red

error in thermal diffusivity between measurements taken on the inner and outer surfaces of the cladding curvature. Idaho National Laboratory's advanced data analysis (developed by Pavlov) allows for a correction due to the curved geometry and that the correction factors of curved samples are different depending upon the heating and measurement surfaces. This information will be used to correct the LFA measurements that will be conducted on the SiC-SiC cladding/Sn/Mo layered system in future work.

The TCM instrument measures thermal diffusivity at the surface of a sample on a meso-scale level which can be 20-100 µm in length to capture microstructural effect on the thermal transport of that area of a material. The SiC-SiC composite was cut across the tube and a portion was mounted into epoxy, polished and coated with a thin layer of gold for data collection with the TCM. Five area cross-sections of the SiC-SiC cladding tube were examined for thermal diffusivity. Each area was a combination of chemical vapor deposit (CVD) and chemical vapor infiltration (CVI) SiC and SiC fiber features. Figure 2 is an optical micrograph of one of the cross-sections that was examined with the different features labeled. The average thermal diffusivity of

CVD, CVI and fiber regions was 36.8, 15.7, and 6.4 mm²/s, respectively (obtained by Hua). This demonstrates that the TCM instrument is sensitive on the length scale of the heterogeneous features of the SiC-SiC composite.

A heating stage on the TCM was used to heat a SiC-SiC cladding / Sn / Mo cross-section from room temperature to 180°C to examine the effect on thermal diffusivity from the Sn bonding layer. Figure 3 shows the thermal diffusivity at various temperatures (obtained by Adkins and Hua) across the SiC-SiC cladding, the Sn layer and the Mo center region. It is noteworthy that the values for diffusivity in the SiC-SiC region are lower than the CVD and CVI values reported at room temperature up to 125°C where they begin to increase. The scatter in the values at all locations within a specific temperature is likely due to a high surface roughness at the location of the TCM measurement which can scatter the laser beams and introduce noise into the data. Above 180°C the integrity of the gold coating was compromised in the vacuum chamber during testing therefore scattering the lasers beyond the ability to measure diffusivity. The next step in this study will be to conduct measurements inside the heating stage under an



Figure 3. Mesoscale thermal diffusivity of the SiC-SiC/Sn/Mo surrogate fuel design at various temperatures. The optical microscopy inset is the approximate area of TCM measurements on the length given on the x-axis

argon purge to allow the sample to get to 300°C, well above the liquid Sn transition.

Future work on this study will include bulk thermal diffusivity measurement of the SiC-SiC/Sn/ Mo surrogate fuel design across the radius using the LFA method up to 300°C as well as the measurement of thermal diffusivity of liquid Sn. The state-of-the-art techniques developed for thermal conductivity measurement of composite/layered materials in curved geometries are critical to enable the postirradiation examination and qualification of novel cladding concepts such as SiC-SiC composites.

SiC/SiC Development Strategy and 5-Year Execution Plan

Principal Investigator: Takaaki Koyanagi Team Members/Collaborators: Yutai Katoh

> ilicon Carbide (SiC) fiber– reinforced SiC matrix (SiC/ SiC) composite-based accident tolerant fuel (ATF) cladding technology is an engineered cladding designed to withstand the severe thermomechanical and chemical environment in light-water reactors (LWRs). As a result, deployment of SiC/SiC in LWRs could offer enhanced performance in design and beyond design basis accidents, thereby removing economic restrictions compared with existing fuel cladding concepts. However, deployment of SiC/SiC LWR-specific SiC technology requires advancements in the foundational understanding of material behavior responses and bridging science and technology. The Advanced Fuel Campaign is supporting the vendors' commercialization program by evaluating the industry-specific SiC/SiC composite cladding concept and providing material property solutions based on in-depth understanding of general SiC/SiC composite material behavior under LWR-relevant environments.

To support the accident tolerant SiC-based cladding technology development, the development strategy and the 5-year execution plan were developed.

Project Description:

The development of the ATF SiC/SiC composite technologies is conducted based on a 5-year research and development (R&D) execution plan. The overarching goal of the R&D activities is to advance SiC cladding concepts to a maturity level suitable to support lead test rod (LTR) insertion into a commercial LWRs. To achieve this goal, the development plan is made to (1) obtain critical experimental data by integrated test to address the cladding performance issues (e.g., loss of hermeticity, hydrothermal corrosion behavior, failure behavior during simulated accident environments), (2) develop and experimentally validate an advanced fuel performance modeling tool to predict cladding behavior involving the interplay of



multiphysics phenomena, and (3) provide material property solutions to optimize cladding design and address feasibility issues.

These three development activities are connected, schematically shown in Figure 1. The advanced fuel performance capabilities will be used to assess the cladding performance to optimize the composite microstructure and fuel rod design. Integrated tests and material property handbook updates for the latest grade of cladding are required to keep the material models applicable. Integrated tests are defined as a type of test in which specimens are exposed to an LWR application-relevant environment involving the interplay of multiphysics phenomena. For example, stress corrosion cracking testing investigates the combined effects of corrosion and applied stress.

Accomplishments

Based on the technology gap analysis, the 5-year R&D execution plan was documented by Oak Ridge National Laboratory (Figure 2). The plain is aligned with the Systematic Technology Evaluation Program (STEP) for SiC/SiC composite-based ATF cladding and core structures, which Figure 1. A schematic illustration for high-level structure of the SiC development activities



Figure 2. SiC/SiC development strategy and 5-year execution plan document

guided early stages of the development activities and identified the critical feasibility and performance issues. The 5-year plan prioritizes experimental and modeling activities to address critical limitations of SiC-based cladding technologies. The demonstration of hermetic cladding and its end plug under normal operating environments has been identified as one of the most critical feasibility issues. Other related normal operation critical performance issues are hydrothermal corrosion and bowing of SiC/SiC composite cladding. Advanced materials testing under accident conditions is also required to obtain important but missing data needed to conduct accident analysis. Material property solutions will be delivered numerically and experimentally to mitigate the critical cladding performance issues. The various activities proposed in the document leverage unique capabilities and expertise available at the national laboratories, including the Advanced Test Reactor and Transient Reactor Test Facility at Idaho National Laboratory, and High Flux Isotope Reactor and Severe Accident Test Station at Oak Ridge National Laboratory.

During the 5-year project period, a transition into a technology integration and implementation phase is anticipated, as illustrated in Figure 3. The technology integration phase



addresses the mitigation technologies for the feasibility issues, advances critical composite performance (e.g., thermo-mechanical properties), and develop other enabling technologies including non-destructive evaluation method of the composite processes. In the implementation phase, the integrated material evaluation and modeling tools is used to address requirements for the lead test rod insertions. High-fidelity fuel performance modeling tools (e.g., BISON code) will facilitate integration of cladding performance data. A fuel performance evaluation may be needed to support an LTR program and likely will be needed to estimate SiC cladding benefits to operation and risk of failure. Figure 3. A schematic illustration of the path to a technology integration and implementation phase for SiC/SiC composite technologies for LWRs

ATF Rodlet Irradiation in ATR (ATF-2C)

Principal Investigator: David Kamerman

Team Members/Collaborators: Brian Durtschi, Travis-Laboossiere-Hickman, Matilda A. Aberg Lindell, Richard Skifton

The ATF-2C experiment representing the state of the art in irradiation test sophistication completed assembly and began irradiation in 2023.

ccident Tolerant Fuel (ATF)-2C is the latest iteration of the Advanced Fuels Campaign's flagship irradiation test located in the center flux trap of the Advanced Test Reactor (ATR). ATF-2C provides a fully prototypic, flowing, pressurized water test environment via the Loop-2A independent water loop. ATF-2C continues the irradiation of chrome coated cladding from the previous ATF-2B experiment and is the first ATF test to include novel SiC-SiC cladding materials. The pinnacle of the test is the irradiation of 6 longer (55cm) fuel pins, four of which contain in-situ instrumentation. Two of the pins contain centerline thermocouples (TCs) and two of the pins contain pressure bellows with linear variable differential transducer (LVDT) assemblies for measuring plenum pressure. The irradiation began with ATR cycle 171A on the 26th of April 2023 and is planned to be irradiated for a total of 4 ATR cycles (60 effective full power days per cycle) concluding in September 2024.

Project Description:

With the closure of the Halden nuclear reactor in Norway in 2018, Idaho National Laboratory's Loop-2A in the center flux trap of ATR is the only fully prototypic pressurized water loop available for civilian nuclear fuels and materials testing in the western world. The development of new fuels for light water reactors relies heavily on this capability. The ATF-2 experiment started in 2018 and successfully completed 7 cycles of irradiation on over 40 different test pins comprised of mostly evolutionary chrome coated zirconium alloy cladding concepts. In 2021 and 2022, the ATR underwent a massive core internal change out. During that time the ATF-2 experiment was redesigned and expanded to include revolutionary ATF concepts such as novel silicon carbide (SiC)-SiC cladding materials and the use of in-situ instrumentation.

The irradiation of SiC-SiC cladding materials started with empty cladding tubes and will next include the irradiation of cladding tubes with inert molybdenum gamma heaters replicating fuel pellets in cycle 173A. Full fueled irradiations of SiC-SiC cladding will then begin in cycle 175A. The long, instrumented fuel pins are part of a collaboration with the Japan Atomic Energy Agency (JAEA) and Mitsubishi Heavy Industries, where chrome coated zirconium alloy cladding provided by Mitsubishi are tested in conjunction with the in-situ sensors. Two of the instrumented pins will contain type N TCs. These pins will use Conax-type compression seals at the top to form a seal with the TC. These compression seals have been fabricated from Zircalloy-4 to



Figure 1. Loop-2A and ATF-2C instrument readouts from Cycle 171A



Figure 2. Long, instrumented fuel pins protruding from the top of ATF-2C tier 4 allow them to weld on to the fuel pin body. The upper assembly of the TC instrumented pins is shown in Figure 2. The accuracy of the type N TCs is \pm 0.75% the temperature reading; with drift due to thermoelement transmutation expected to be -0.45% at around 1021 nvt showing a good calibration over the life of the TC. The other two instrumented pins will contain pressure bellows and a LVDT measurement assemblies. Because this is a longduration irradiation, during which creep can be a significant concern, the bellows assemblies will have a weephole to allow precise selection of pressure at the beginning of life.

This will minimize the stress state and, thus, creep for the bellows for a large portion of irradiation while retaining reasonable resolution.

Accomplishments:

In FY23 the final assembly of the ATF-2C test train was completed and the test was installed into the reactor. The first cycle of irradiation concluded on June 19, 2023, with approximately 54 effective full power days of irradiation. Peak burnups for chrome coated cladding test pins carried over from ATF-2B reached 33 – 34 GWd/MTU of burnup after their 8th cycle of irradiation. The SiC-SiC test pins reached nearly 1 dpa of damage


which is where the irradiation damage for this material is expected to saturate. The longer JAEA test pins with in-situ instruments reached burnups of between 3-4 GWd/MTU of average burnup with peaking factors of about 1.2 pushing the peak burnup in the fuel pin to between 3.5-5 GWd/MTU. The TCs performed well during the first irradiation cycle recording expected fuel temperatures between 800°C and 900°C. The LVDTs measuring plenum pressure appeared to be drifting up slightly although little fission gas is released at the beginning of the irradiation. The LVDTs did come offline for several days

in the middle of the cycle before coming back online. The process of converting the measured displacements into meaningful plenum pressure measurements is ongoing. Overall, the irradiation experiment is achieving its goals of irradiating a variety of both evolutionary and revolutionary ATF fuel concepts in a fully prototypic environment. This world leading irradiation test brings together ATF developers both nationally and internationally and has achieved a new level of sophistication through the incorporation of in-situ instruments providing valuable real time data to ATF developers.

Figure 3. SiC-SiC Cladding Pins loaded into ATF-2C tier 3

2.4 LWR PIE

Examinations of BWR Lead Test Rods

Principal Investigator: Jason Harp

Team Members/Collaborators: Caleb Massey, Casey McKinney, Annabelle Le Coq, Nicholas Russell, Padhraic Mulligan, Jesse Werden

ak Ridge National Laboratory (ORNL) supported the General Electric (GE) Accident Tolerant (ATF) and High Burnup (HBu) program through several efforts in fiscal year (FY) 2023. Postirradiation examination (PIE) was performed on IronClad segments irradiated in the Hatch reactor. This included mechanical property testing, microstructure characterization, and activation analysis. Irradiation testing efforts were also initiated to test candidate FeCrAl alloys in the High Flux Isotope Reactor (HFIR). The irradiations will explore both traditional mechanical testing geometries and for the first time tube geometries that are representative of the diameters used in boiling water reactors (BWRs). Preparations were also made to receive irradiated fuel from the Clinton Nuclear Power Station to further support GE's HBu efforts.

Project Description:

Research in this area supports GE's continued development of ATF for future deployment in commercial reactors. Data generated by future exams on high burnup BWR material will support GE's efforts to extend the discharge burnup of BWR's and provide additional PIE regarding the performance of BWR fuel.

Accomplishments:

Mechanical tests and microscopy were conducted on an IronClad lead test rod irradiated in the Hatch ORNL's partnership with GE continues to provide PIE data necessary to support GE's mission of enhancing fuel materials for BWR commercial development in the near and medium term.

reactor. The lead test rod, comprised of multiple axial segments, was disassembled within the hot cells at the Irradiated Fuels Examination Facility at ORNL. Following alpha decontamination, sectioned tubes were shipped to the Low Activation Materials Development and Analysis Laboratory for mechanical testing and microscopy. In FY 2023, initial investigations prioritized the examination of the end cap. the upper segment, and the mid axial segment of the rod. Wire electrical discharge machining was used to section miniature tensile specimens from the end cap. The same technique was also used to extract ring pull and axial tube tensile specimens from the individual tube sections. Summarized mechanical testing data is shown for these three characteristic locations in Figure 1. In this figure, specimens tested at room temperature are shown in blue while specimens tested at operational temperature are shown in red. Room temperature specimens routinely exhibited brittle



fracture in the elastic regime of tensile deformation with no retained ductility. However, specimens tested at operational temperature exhibited some retained ductility, with uniform elongations measuring up to 3%. For ring-pull specimens (tested in the circumferential direction), the loads are not converted to equivalent stresses/strains due to the combined bending and axial tension loading conditions in this test.

The underlying microstructural features underpinning the postirradiation mechanical response of the IronClad segments are revealed Figure 1. Summary of the types of mechanical testing performed on Hatch-irradiated IronClad segments. Miniature tensile specimens were extracted from the (a) end cap, (b) upper segment, and (c) axial midplane segment. Specimens tested at room temperature showed minimal ductility, with failure prior to yielding. Specimens tested at operational temperature conversely showed some retained ductility



Figure 2. A scanning transmission electron microscopy bright field image of dislocations in commercially irradiated IronClad. Traced loops, including face- and edge-on a{001} loops and a/2{111} loops are shown in blue and red, respectively, while smaller black dot defects are shown in black. Dislocations deviating from characteristic loop morphologies are categorized as preexisting network dislocations in green

Figure 3. Atom probe tomography (APT) results reveal Cr-rich segregation in commercially irradiated IronClad segments. On the left, an APT reconstruction shows a high density of Cr-rich surfaces using 19% enriched iso-concentration surfaces, while the right image shows a proximity histogram quantifying the level of Crenrichment and (Fe,AI)-depletion within identified Cr-rich precipitates

> in Figures 2 and 3. Following irradiation, a high density of dislocation loops was imaged along characteristic habit planes in the bodycentered cubic crystal lattice. These loops are visible in Figure 2 taken from an upper segment of the rod. In addition, Cr-rich precipitation was also revealed using atom probe tomography. The high aluminum content, coupled with the relatively low irradiation temperature, resulted in core Cr concentrations reaching 40% Cr, as shown in the proximity histogram in Figure 3.

More data on the fundamental

changes to mechanical properties produced by neutron irradiation in FeCrAl candidate alloys are needed to fully understand the wide variety of processing parameters and alloy compositions that are currently being evaluated for use as BWR cladding. This requires irradiation of new candidate materials as well as historical alloys to understand how processing and alloying improvements impact properties following irradiation. Irradiation positions in HFIR are an ideal location for rapid screening of several different alloys using standardized irradiation test designs. A series of irradiations were designed, and



Figure 4. Irradiation test vehicles used to support the irradiation of alloys of interest for GE show both the new large diameter capsule above and the standardized tensile test geometry below. Both of these capsules will support the irradiation of candidate alloys in support of future iron based cladding deployments in current generation BWR and advanced reactor applications

fabrication was initiated to irradiate FeCrAl alloys that are of interest to GE and the Advanced Fuels Campaign. These tests will utilize standard geometry tensile specimens and standard geometry bend bar specimens. Additionally, a new irradiation capsule was designed to accommodate the larger diameter cladding utilized in BWR's. This irradiation will be the first irradiation of BWR geometry cladding in HFIR. Hydrodynamic testing of this new capsule was also completed as a necessary step to approve the use of this large diameter irradiation vehicle in HFIR. Sketches of both the standard tensile specimen irradiation vehicle and the large diameter cladding vehicle are included in Figure 4.

Planning is underway to receive irradiated fuel from the Clinton Nuclear Power Station to support the High Burnup licensing efforts of GE. There is a pressing need to

gather additional irradiated material data from production fuel rods with local exposures near the current regulatory limit to support upcoming licensing submittals. This shipment will allow ORNL to characterize high burnup BWR fuel including gadolinium doped uranium dioxide fuel. This will be the first examination of gadolinium doped fuel in the U.S. that was irradiated in a commercial reactor. This fuel will enable the study of uranium dioxide microstructure changes in BWR geometries for comparison against the formation of similar microstructures in pressurized water reactor conditions. The evaluation of hydrogen pick-up and the accident behavior of BWR fuel will also be studied.

Commercial Lead Test Rod Transportation and PIE

Principal Investigator: Edward Mai

Team Members/Collaborators: Eric Woolstenhulme, Susan Case, Heather Chichester, Ernesto Pitruzzella, Jason Harp

Establishing routine capability to receive full length ATF test rods from the fuel vendor teams and share segments from them between DOE complex laboratories is vital to the strategic ability to conduct integrated ATF testing, and support future industry LWR R&D needs long term. ddressing the strategic objective to efficiently plan, prepare, and execute transportation needs to support the Advanced Fuels Campaign (AFC) post-irradiation examination (PIE) mission, our group focused its efforts this year on establishing these critical shipping pathway tasks.

Project Description:

Three accomplishment areas were targeted objectives this year:

- Create a national PIE strategy, supported by centralized national planning and coordination of required transports, that meets the timeline and objectives of Both industry and Department of Energy, Office of Nuclear Energy (DOE-NE) stakeholders in as efficient a manner as possible.
- Develop and complete readiness, staging, and pre-shipment check-listing and Idaho National Laboratory (INL) tasks required to ship the BEA Research Reactor (BRR) cask from ORNL to INL with accident tolerant fuels (ATF)/high burnup (HBU) segments
- Develop and complete readiness, staging, and pre-shipment checklisting and INL tasks required to ship the Nuclear Assurance Corporation (NAC) Legal Weight Truck (LWT) cask from Bryon Nuclear Generating Station (NGS) to INL with ATF/HBU Lead Test Rods (LTR).

Accomplishments For the national PIE strategy formulation, a second round of ATF/HBU stake-

holder reviews, based on current year vendor team changes was conducted, and incorporated into a fiscal year (FY)23 ATF strategic transportation and PIE plan.

Stakeholders were polled in November 2022 for updates to defined strategic shipping events to support ATF PIE, and the draft ATF Program Transportation and PIE Plan was prepared and issued in January 2023 in protected form. Some updates and vendor directed changes identified in the second quarter were incorporated and presented to the ATF community at the April 4, 2023, Electric Power Research Institute (EPRI)/DOE Joint Combined Workshop on ATF and HBU. The second annual NE AFC PIE Capabilities and Transportation Workshop was held on August 8 and 9 at INL, with more than 55 participants in attendance.

For the BRR cask deployment effort, contract modifications began with ORANO Federal Services in October 2022 to prepare analysis and create the Certificate of Compliance (CoC) revision for the BRR cask and established by an award in January 2023. In the course of analysis, design of new inner containers and baskets was performed to optimize handling of encapsulated segments in and out of cask packaging and targeted hot cell handling needs. As analysis progressed, concern overdose on cask contact of the postulated loads required an iterative approach to creating a proposed contents loading for the Safety Analysis Report (SAR) CoC revision. Contract modifications were completed for Secure Transport Services (STS) to help develop a plan to ship ATF LTR segments from ORNL to INL, potentially using the STS dry transfer cask system, used by other NE programs for fuel shipments. Team reviews with INL, ORNL, and STS were initiated in January 2023 to begin to define a logistical process path forward and continued through FY23. Evaluations by the group and specifically ORNL hot cell management concluded that the STS dry transfer cask system is the preferred alternative for shipping ATF LTR segments between INL and ORNL. Drawing reviews for evaluating use of the STS transfer cask system began in April 2023, and Rod in Tube Canister hardware draft drawings were completed in July 2023. Draft source terms calculations for bounding ATF shipment loading were performed by Idaho State University staff and students in June 2023, with revisions performed in the fourth quarter of the year by INL staff to complete preparation and submittal of the CoC revision to the Nuclear Regulatory Commission in early FY24.

In preparation to receive the NAC-LWT cask system at INL from Byron NGS, INL transportation planning staff initiated a task release with NAC in November 2022 to perform a shipment of Byron LTRs to INL and awarded the task release for the Byron shipment in June 2023. In March, Byron operations staff postulated that it would support a December 2023 loadout of the ATF LWT basket and transport to INL via the NAC-LWT cask and confirmed it via Westinghouse Electric Corporation (WEC) in April 2023. On April 11, the IWTU started operations to treat radioactive liquid waste, and on May 1, 2023, Idaho Governor Brad Little notified DOE-NE that conditions stated in the 1995 Batt Agreement to allow the 25 rod Byron shipment were satisfied. INL transportation planning staff, in conjunction with WEC ATF project management, began to review lessons learned from previous ATF shipments conducted and prepared a detailed checklist of all tasks to evaluate and conduct to ensure efficient and timely operations and shipment. Bi-weekly integrated planning meetings between INL, WEC, Byron, and other operations support began in May 2023 and will continue into FY24 to orchestrate all tasks needed for the shipment, now anticipated for the work window of November 28 to December 6, 2023. WEC rod fabrication, loading, and power histories for all 25 rods were received at INL on August 19, 2023, to begin shipping and facility acceptance calculations and documentation. A draft report showing the source term calculation and results was provided to the Hot Fuels Examination Facility (HFEF) for a preliminary impact assessment on the facility's receipt criteria - along with other

required data. Weekly planning meetings have been conducted with HFEF operations, management, nuclear safety, criticality safety, and engineering to discuss tasks, schedule, completing Nuclear/ Irradiated Material Acceptance Form-2095, and status necessary to prepare HFEF for the shipment and obtain authorization to ship.

Project personnel have been working closely with the cask handling crew at HFEF to ensure cask configurations are documented, fuel loading maps are accurate, tools and equipment are ready, and fuel unloading plans are approved. The cask engineer performed the HFEF cask cart working load limit and seismic analysis. The engineering calculations and analysis report has been drafted to increase the cask cart capacity to support NAC-LWT shipment.

HFEF cask operations performed a leak check at the 1M penetration using the BRR cask to allow HFEF to unload the NAC-LWT with the fuel rods while in high Material At Risk mode if that is required. The steps for the leak check and the surveillance requirements are currently being implemented into procedures.

PIE Technique Development for SiC Claddings

Principal Investigator: Alexander J. Winston

Team Members/Collaborators: Jordan M. Argyle, Spencer H. Parker, Jerry A. Kahn, Katie A. Hawkins, Jake A. Stockwell, David Kamerman, Peng Xu



Figure 1. 3D printed Collet

This project is to conduct assessment and preparation to remote handle Silicon Carbide (SiC) rodlets in hotcell for non-destructive and destructive examinations.

Project Description:

In anticipation of the receipt of irradiated SiC rodlets from Advanced Test Reactor (ATR), the Hot Fuel Examination Facility (HFEF) has begun developing the capabilities for examining non-metallic cladding. Due to the nature of ceramics, special handling methods must be implemented to avoid mechanically stressing the cladding prior to destructive testing for remote handling.

Accomplishments:

The in-cell post irradiation examination (PIE) readiness preparation for SiC fuel rodlets can be summarized in two scopes: material handling and leak testing.

1. Material Handling:

Reduction in handling of the rodlet itself will be achieved via grasping of the end plug assembly in a specially designed, yet multi-use collet, that will be near-universally accepted by examination instrumentation in the HFEF hot cell. Metallic rodlet handling is normally done with a press fit style holder specific to each rodlet geometry. The collet (model on left, seen orange as 3D printed polymer in mock-up testing) seen below will be able to grasp rodlets securely while distributing the compressive force evenly around the most robust portion of the rodlet, the end cap.

This collet will allow mating to the E/M crane T-handle for cross cell movements, mating to the 3-jaw chuck used for rotational profilometry at the measurement bench, mating to the gripper for precision gamma scanning, and will travel as an assembly (collet and rodlet) for neutron radiography.

Additionally, the collet may be ideal for gripping when the rodlet is to be sectioned for mechanical testing and ceramatorgraphy purposes in the containment box.

2. Leak Testing:

In the past few years, many programs have shown in the capability of HFEF to re-encapsulate previously irradiated fuel in accident tolerant fuel (ATF) style cladding pins for re-irradiations at Transient Reactor Test Facility and ATR. The system that was devised to perform this exercise was the ICWUPS or "In-Cell Weld Under Pressure System", as shown in Figure 3.

During the evolution of re-encapsulation, the welding vessel is pressurized with helium as a weld cover gas and the end cap is mated to the body of the cladding. To verify the quality of the weld, the sealed rodlet is then transferred to a bell jar, where a vacuum is pulled for verification of hermiticity i.e., leak checked.



Figure 2. Robotic arm handling demonstration of SiC rodlet using 3D printed collet



Figure 3. ICWUPS or the "In-Cell Weld Under Pressure System" For the purposes of leak checking the SiC rodlets, the team has revised the ICWUPS testing procedure to include only the pressurization of the welding chamber and the He-leak checking in the bell chamber. This system will be able to meet the test requirements dictated by General Atomics. Location and function of the assembly has been verified and the system will be ready for use when the governing procedure has been reviewed for safety and entered as an official document, expected mid-September 2023.

Demonstration of incell remote handability and leak test capability for SiC rodlet is a critical step toward SiC ATF rod development.



DIC Developments for RHT and Coating Behavior Assessment

Principal Investigator: Robert Hansen

Team Members/Collaborators: Prasenjit Dewanjee, Philip Petersen, Fabiola Cappia, Jake Stockwell, David Kamerman



Figure 1. Diagrams of the two RTT configurations. The 'dogbone' (left) with a double gauge supported by a central insert, and the 'hemi' (right) with a single gauge supported by hemicylindrical inserts

n recent years, the Idaho National Laboratory (INL) Accident Tolerant Fuel (ATF) Post-irradiation Examination program has developed techniques for accurate measurement of cladding material properties. Anisotropic ATF cladding concepts being pursued require assessment with methods like the Ring Hoop Test (RHT), which consumes only small amounts of irradiated cladding material and interfaces easily with existing infrastructure at INL hot cells. Historically, several configurations have been used, each with unique advantages and shortcomings. Importantly, the various configurations can produce significantly different measured material properties, and the unique stress states of the configurations can interact with coated ATF claddings differently. To address these challenges and validate modeling studies of the test methods, digital image correlation (DIC) has been implemented for out-of-cell RHT experiments with a variety of ATF coated zircaloy cladding concepts. This work enables deployment and interpretation of robust in-cell testing on irradiated cladding samples.

Project Description:

Pairing DIC with RHT experiments offers two significant advancements in cladding mechanical testing. The first is experimental validation of computer modeling efforts. Previous RHT development utilized finite element modeling to understand the



Figure 2. Customized split-tube furnace on glide rails (left), and dualcamera images of speckled cladding gauges (right). The cameras are stacked one above the other with a stereo angle less than 20 degrees and focused through the window

advantages and drawbacks of two configurations, the 'dogbone' which features a dogbone-shaped central insert to support the two gauges of a ring specimen, and the 'hemi', which uses full hemicylindrical mandrels to support a single gauge. These configurations are shown in Figure 1. Modeling showed that the dogbone exhibits non-uniform stress due to bending, resulting in underpredictions of the strength and significant sensitivity to uncontrollable parameters. Conversely, finite element analysis showed the hemi produces uniform stress in the gauge and overpredicts strengths but can be corrected by accounting for ring friction. The hemi system was recently qualified for in-cell use at INL, joining the dogbone configuration, but neither had been characterized using DIC. By deploying DIC, full-field strain maps can be generated to validate modeling results and to interpret test outcomes.

The second key advantage of DICbased RHTs is the ability to assess ATF coating performance. With near-term ATF concepts centered on applying chrome coatings on existing zircaloy-based claddings, evaluating the effect and behavior of these coatings is critical. Full-field displacements and strains produced from DIC measurement can be used to detect coating cracks as they develop during mechanical loading of the cladding, helping to establish strain limits of specific coatings and provide understanding of coating failure behavior. When paired with load-displacement data, it can become a powerful tool for evaluating coated cladding performance.

To realize these benefits, a DIC setup was deployed for out of cell RHT experiments. Stereo DIC was necessary to measure the curved outer surface of the cladding where coating exists, requiring a dual camera system capable of



Figure 3. Load-displacement paired with DIC results for room temperature test of cold-spray coating. a) Loaddisplacement curve, b) strain in y-direction during linear deformation (140 s into test), c) strain in y-direction post-necking (485 s into test)

Figure 4. DIC results showing cracking for elevated temperature test of coldspray coating. a) Immediately after test begins (5 s into test), b) post-necking when cracks are detected (665 s into test), c) immediately after previous image (670 s into test). Closeups of region with cracking shown below

> imaging through a furnace window for elevated temperature tests. Optics, lighting, software, speckle patterning, and calibration techniques all needed to be optimized to provide robust DIC measurements for the RHT.

Accomplishments:

This DIC work was accomplished at the out-of-cell load frame located at INL's Materials and Fuels Complex. Prasenjit Dewanjee, a graduate student intern at INL from Utah State University, performed most of the DIC setup and image capturing, under the direction and with the assistance of Robert Hansen. Phil Petersen designed the grips and RHT experimental method, and also assisted in DIC setup. Programmatic direction and coordination of material specimens was provided by Fabiola Cappia, Jake Stockwell, and David Kammerman. For the testing, four different cladding types were tested: uncoated Zircaloy-4 provided by the Advanced Fuel Campaign (AFC), Cr-coatings produced by coldspray (CS) from Kumar Sridharan at the University of Wisconsin, and a physical vapor deposition Cr-coating and CrN coating, both from Martin Sevecek at Czech Technical University in Prague. Initial tests used the dogbone configuration.

A unique furnace setup was used, with a custom window added in the front allowing optical access for DIC during thermomechanical testing. The split-tube furnace was also customized to allow the halves to be fully separated, as seen in Figure 2, designed by Andrew Johansen (an intern from the University of Idaho). This facilitates stereo DIC calibration prior to testing. Speckle patterning was developed for the small length scale elevated temperatures required; sample images of the patterned gauges from the camera pair are shown in Figure 2. Open-source software was used both for image acquisition, with custom python scripts, and for image correlation, with digital image correlation engine from Sandia National Laboratory. Through optimization of lighting, imaging settings, stereo angles, and software settings to improve image quality and maximize calibration scores, successful DIC measurements were made at both room temperature and 350°C.

An example of pairing the loaddisplacement data collected from the load frame with the DIC results can be seen in Figure 3, which shows strain in the vertical (y) direction for a room temperature dogbone RHT with a CS cladding sample. Fullfield data sets such as these are very useful for confirmation of modeling results, allowing point-by-point comparison with finite element outputs for strains in multiple directions, across the range of test deformations. In this case, the results illustrate the non-uniform deformation in the gauge at a displacement which the load curve would indicate is undergoing uniform elastic deformation. This aligns well with modeling predictions.

The capability to assess coating behavior through cracking was also demonstrated with the DIC setup. In Figure 4, vertical strain contours are shown for an elevated temperature dogbone RHT with a CS cladding sample, with a portion of the gauge shown below. The contour just after test initiation (a) shows no evidence of cracking, as expected. However, as deformation increases, eventually localized strain bands appear in the contour (b). Closer inspection of the specimen image shows horizontal cracks coinciding with the strain bands, which grow large enough by the next image (c) to cause dropped subsets. These added DIC capabilities position INL and AFC for high quality, robust mechanical testing of ATF cladding.

Mechanical Testing – Methodology, Evaluation, Testing, and Analysis

Principal Investigator: Ben Garrison

Team Members/Collaborators: C. Massey, W. Ren, M. Gussev, T. Graening, R. Sitterson, N. Capps, K. Linton



Figure 1. Procedure for geometrybased mechanical property correlation. Different specimen geometries of "Identical" material were machined from the same plate of fully recrystallized Zircaloy-4, and the mechanical properties obtained from each geometry were compared to determine if there was any sensitivity to anisotropy and texture for a nuclear relevant alloy I n support of a multi-laboratory cladding mechanical properties database, Oak Ridge National Laboratory's cladding tube mechanical test geometries were manufactured out of several nuclear-relevant materials and tested. These geometries were optimized as mechanical test specimens that could be used on irradiated materials, including those from commercial reactors via in-cell machining.

Project Description:

The goal of this project is to establish a paradigm of mechanical testing methodology, evaluation, testing, and analysis (META) that produces precise and repeatable mechanical property data on nuclear relevant material. In this effort, mm-scale tube axial tension test (ATT) and ring tension test (RTT) mechanical test specimens that can measure the mechanical properties of a tube geometry in the axial and hoop directions were developed that (1) minimize nonuniform deformation in the tensile regions, which makes measured outcomes easier to relate to mechanical properties and (2) minimize the material consumed per test, which becomes highly valuable with irradiated materials. Following the geometry development, the techniques employed need to be documented in a thorough and organized manner so that material property information measured by the techniques are detailed and unambiguous.

Accomplishments

Historical data created correlations among SSJ3, ATT, and RTT specimen geometries with several materials machined out of plate material, but none of the materials were similar to the textured, anisotropic zirconiumbased alloys used in nuclear industry except for one Zircadyne rod. As a result, this fiscal year (FY) SSJ3, ATT, and RTT specimens were machined out of fully recrystallized Zircaloy-4 plate to obtain mechanical property correlation data on a nuclear-relevant alloy. Those correlations were consistent with the previous data, so it was found that there were not sensitivities associated with the Zircalov-4 material. Furthermore, the short gauge length RTT specimen was previously shown to reduce strain gradients across the gage length of the specimen during testing, so correlation data was generated for that new geometry.

To allow for higher burn-up times and reduce the severity of loss-ofcoolant accidents, Advanced Fuels Campaign is pursuing accident tolerant fuel materials that reduce oxidation rates. One of these materials being pursued is chromiumcoated zirconium alloys. This FY, as-received, Hybrid High Power Impulse Magnetron Sputtering Cr-coated, and Cold-spray Cr-coated Zircaloy-4 tubes were machined into ATT and RTT geometries via electric discharge machining and end mill machining to investigate The testing geometries developed in META are able to be tested in a hot cell environment, minimize strain gradients from testing tube components to clarify mechanical property measurements obtained from the test, and minimize the consumption of valuable irradiated material.

the effect of the coating methods and machining methods on the mechanical properties. It was found that the cold spray and -100V bias coated material exhibited increased strength properties and slightly reduced uniform and total elongations in the axial direction from ATT tests at room temperature and 300°C. However, no improvements were found in the hoop direction from RTT tests, but results at 300°C showed that the differences between the coated materials and as-received material in the hoop direction were diminished. Furthermore, for all materials it was found that end milled specimens exhibited significantly reduced stress properties, signifying that the mechanical properties obtained from the tubing specimens are highly sensitive to the machining process used to create the tensile geometry. As a result, it is recommended that end mill machining be used in all future testing to reduce the machiningrelated material differences.



Figure 3. ATT specimens were machined out of Zircaloy-4 tubing in the AR and Cr-coated conditions via electrical discharge machining (EDM) or end mill to determine the effect of machining method on mechanical properties. It was found that EDM machining significantly impacts the mechanical properties obtained from testing, so end milling is recommended for all future testing



Figure 2. ATT and RTT mechanical tests were performed with as-received as well as cold-spray, -100V HiPIMS, -150V HiPIMS, and -200V HiPIMS Cr-coated Zircaloy-4 to investigate the effect of the coating procedures on mechanical properties of the tubes. Marginal strength improvements were found in the axial direction for the cold spray and -100V bias coating methods

2.5 LWR FUEL SAFETY TESTING

SATS 2.0 Capability Demonstration

Principal Investigator: Kory Linton

Team Members/Collaborators: Mackenzie Ridley, Sam Bell, Adam Willoughby, Brandon Johnston, Nathan Capps



Figure 1. Image of SATS 2.0 furnace

The United States nuclear industry is seeking to establish a technical basis for extending burnup beyond a peak rod average of 62 GWd/tU. Fuel fragmentation relocation and dispersal (FFRD) has been identified as the key technical and safety hurdle inhibiting burnup extension. The Nuclear Regulatory Commission (NRC) Research Information Letter (RIL) summarized key elements associated with FFRD that could benefit from additional experimental data and better support the industry's safety basis. Research priorities have been determined by the Electric Power Research Institute's (EPRI) Collaborative Research on Advanced Fuel Technologies working group. These priorities are focused on facilitating material model development, code validation, and licensing of industry safety cases. This prioritization has been documented in a consensus loss-of-coolant accident (LOCA) test plan. Both the test plan and the NRC RIL underscore the need for detailed. in-situ capabilities to rapidly address existing data gaps. These needs encompass detailed temperature characterization, in-situ cladding balloon strain, transient fission gas release, and high heating rates. In order to provide a platform for addressing these data gaps, ORNL has developed a second-generation Severe Accident Test Station "SATS 2.0" which improves the available test conditions and enhances the data collection capabilities.

Project Description:

The objective of this project includes several key aspects: 1) deployment of

a new furnace capable of achieving heating rates on the order of 100°C/s. 2) generation of detailed temperature profiles within the furnace for applications related to fuel performance, 3) development and deployment digital image correlation techniques to capture in-situ strain deformation, and 4) development and deployment of transient fission gas release measurement capabilities. Many of these objectives are summarized in separate accomplishments, and therefore, this accomplishment will focus on the first objective. The successful deployment of the new SATS furnace addresses multiple capability gaps that limit the Department of Energy from meeting its objectives. The new furnace offers the ability to generate heating rates consistent with industry high burnup LOCA needs. It also provides the ability to generate prototypic heating rates to support time at temperature effects (also a major industry activity).

Accomplishments:

The original SATS system was developed out-of-cell to evaluate candidate accident-tolerant fuel cladding concepts under design basis and beyond design basis accident conditions. Leveraging this experience, the SATS system was replicated and modified for current hot-cell operation. The hot-cell system underwent rigorous commissioning exercises and has successfully conducted multiple experiments involving high burnup fuels. Throughout the design of the SATS system, careful consideration was given to the operational constraints of the hot cell. These constraints include spatial limitations imposed by the manipulators used for unit operation and restrictions on the volume and types of gases that could be released from the system. This careful planning ensured an efficient transition from the out-of-cell module to the in-cell capability. The SATS 2.0 system was envisioned to improve heating rates benefiting from a new furnace, depicted in Figure 1, with an improved power output of 24 kW from twelve lamps, as compared to the previous four lamps providing 8 kW. These enhancements are expected to generate both higher heating rates and more uniform axial and azimuthal temperature profiles, representing a significant advancement in SATS abilities to address research needs.



Figure 2. Summary of heating rates from an unpressurized tube with and without a filler rod



Figure 3. Post-test scans of balloon burst test at heating rates of 100°C/s

The second-generation Severe Accident Test Station (SATS 2.0) will simulate accident scenarios and evaluate fuel segment behavior under a wide range of conditions to support critical gaps, including tFGR, cladding performance (existing light water reactor and advanced reactor), high burnup LOCA scenarios, and fuel rod time at temperature evaluations.

The SATS 2.0 furnace has been successfully assembled and tested in an out-of-cell setting. Figure 2 shows thermocouple measurements from the two commissioning tests with unpressurized cladding specimen. In one of the tests, there was no standard alumina filler rod (used as a surrogate for fuel in out-of-cell testing), while in the second test, a filler rod was included. The resultant heating rate for the tube without an alumina filler rod was a consistent 120°C/s from 300 to 1200°C. The sample with an alumina surrogate also achieved a heating rate of ~120°C/s up to 800°C but the heating rate slowed significantly between 800 and 1200°C. This effect was likely a result of the increasing heat capacity of the alumina filler rod with temperature. Future work will investigate alternative surrogate materials to mitigate this effect. Four additional tests were conducted to

further assess the system's capabilities. Three of these tests followed the standard SATS LOCA burst test protocol at different pressures (6.2, 8.2, and 10.3 MPa), as depicted in Figure 3. Heating rates of approximately 100°C/s were successfully attained for these samples, and the recovered burst data aligns well with established empirical models from historical data. For the fourth test, the sample was cyclically heated at a rate of 70°C/s to a peak temperature of 800°C and then allowed to cool to 300°C between cycles. This test successfully demonstrated the capabilities of SATS 2.0 to simulate complex conditions observed in anticipated operation occurrences, while mitigating temperature overshoots.



tFGR Measurement and In-Cell Demonstration

Principal Investigator: Yong Yan

Team Members/Collaborators: Jason Harp, Chuck Baldwin, Bob Morris, Nathan Capps



Figure 1. Schematic diagram of SATS tFGR facility for realtime tFGR measurement

This new system will evaluate critical gaps in tFGR behavior for fuel segments experiencing simulated LOCA events.

he transient fission gas release (tFGR) during the temperature increase associated with a lossof-coolant accident (LOCA) in lightwater reactors (LWRs) is a significant contribution to the total pressure in a fuel rod and may cause an unexpected rod burst. The lack of data related to transient fission gas (tFGR) continues to be a key gap in understanding LWR cladding burst behavior under LOCA conditions [1–6]. To fully characterize this behavior. tFGR data must be collected from several different systems. The available data suggests that tFGR testing must be performed under fuel-pin-relevant pressure (~10 MPa) to be characteristic of in-pile conditions [2–6]. Oak Ridge National Laboratory (ORNL) has developed a system to measure the integral tFGR from irradiated fuel segments. The system design, testing capabilities, and construction were reported in

fiscal year (FY) 2022 [7]. This milestone report summarizes the tFGR deployment and the first in-cell tFGR test using the irradiated high burnup fuel.

Project Description:

The goal of the integral testing is to capture the onset temperature of tFGR and the amount (moles of gas) of tFGR for a fuel segment experiencing a representative LOCA transient. This release is measured while the fuel is under mechanical constraint by the cladding and under representative fuel pin internal pressure before the segment bursts. For the ORNL approach, the irradiated fuel will be cut into segments 15 to 30 cm long. These segments can be refabricated with end caps and connected to a pressurization system. The tFGR system is coupled with the ORNL Severe Accident Test Station (SATS) and builds upon previous ORNL experience with the Core Conduction Cooldown Test Furnace [8]. During the SATS LOCA test, the segment will be stepwise heated to a temperature at a specified ramp rate, and the content of the segment will be expanded into a larger volume and sent to the fission gas traps out-of-cell at atmospheric pressure. There, the content will be evaluated by counting the released krypton-85 content. The test will then proceed on to the next hold temperature until the final temperature is reached. Alternatively, the fuel segment could be heated under representative LOCA ramp and then capture the total amount of tFGR released by the sample.



Figure 2. SATS installed in ORNL hot cell

Accomplishments:

The design and construction of the tFGR system were completed in FY 2023. The system is designed to integrate with the existing SATS, as illustrated in Figure 1. The tFGR system consists of a sweep gas system to transport gases from the in-cell SATS apparatus (Figure 2) to an out-of-cell fission gas detection system composed of a series of cold traps to capture the off-gas from the heating tests. It also includes a gamma spectrometry system to detect and measure krypton-85. Initial system testing operations have been completed, and krypton-85 collection and measurement were verified along with the ability to detect stable inert gases. The new tFGR system has been set up in the ORNL Irradiation Fuel

Examination Laboratory, as shown in Figure 3.

During a tFGR test, helium sweep gas is metered through a mass flow controller. It then passes through the hot-cell wall and enters the bottom of the quartz tube that contains the test specimen. As fission gas (krypton-85) is released from the test specimen during heating, it mixes with the sweep gas and exits the quartz tube at the top. Then it passes back through the hot-cell wall and on to the moisture trap assembly located on the tFGR table. When the mixed gases pass through the moisture trap, any water vapor present in the sweep gas stream will be removed, and fission gases and helium will pass on to the cold-trap assembly. The cold-trap assembly contains a column of activated charcoal

Figure 3. The SATS tFGR system in the ORNL Irradiation Fuel Examination Laboratory



maintained at –194°C. As the fission gas (krypton-85) encounters the charcoal, it is absorbed and trapped in place while the helium sweep gas passes through and returns to the hot cell through an exit mass flow meter and integrated exhaust system. The released gases transferred to the tFGR table are analyzed online by a gamma spectrometry station under the cold trap cooled by liquid nitrogen.

Figure 4 shows the cumulative release measured during the LOCA-type transient performed on a 32 mm long high burnup sample irradiated up to about 70 GWd/MTU. This figure demonstrates the successful completion of an in-cell tFGR test with irradiated fuels. The data analysis and post-test examination are underway, and results will be reported in FY 2024.

References:

- [1.] Y. Pontillon, M.P. Ferroud-Plattet, D. Parrat, S. Ravel, G. Ducros, C. Struzik, I. Aubrun, G. Eminet, J. Lamontagne, J. Noirot, A. Harrer, Experimental and theoretical investigation of fission gas release from UO2 up to 70 GWd/t under simulated LOCA type conditions: The GASPARD program, Proc. 2004 Int. Meet. LWR Fuel Perform. (2004) 490–499.
- [2.] J. Rest, M.W.D. Cooper, J. Spino, J.A. Turnbull, P. Van Uffelen, C.T. Walker, Fission gas release from UO 2 nuclear fuel: A review, J. Nucl. Mater. 513 (2019) 310–345. https://doi.org/10.1016/j.jnucmat.2018.08.019.



Figure 4. Krypton-85 fission measured during the LOCA test

- [3.] G. Khvostov, A. Pautz, E. Kolstad, G. Ledergerber, Analysis of a Halden LOCA test with the BWR high burnup fuel, LWR Fuel Perform. Meet. Top Fuel 2013. 1 (2013) 644–651.
- [4.] G. Khvostov, Analytical criteria for fuel fragmentation and burst FGR during a LOCA, Nucl. Eng. Technol. 52 (2020) 2402–2409. https://doi.org/10.1016/J. NET.2020.03.009.
- [5.] H. Sonnenburg, W. Wiesenack, J. Karlsson, J. Noirot, V. Garat, N. Waeckel, F. Khattout, A. Cabrera-Salcedo, J. Zhang, G. Khvostov, A. Gorzel, V. Brankov, F. Nagase, P. Raynaud, M. Bales, T. Taurines, T. Nakajima, A. Alvestav, Report on Fuel Fragmentation, Relocation and Dispersal, NEA/CSNI/R(2016)16. Organisati (2016). https://www. oecd-nea.org/nsd/docs/2016/ csni-r2016-16.pdf.
- [6.] J.A. Turnbull, S.K. Yagnik, M. Hirai, D.M. Staicu, C.T. Walker, An Assessment of the Fuel Pulverization Threshold During LOCA-Type Temperature Transients, Nucl. Sci. Eng. 179 (2015) 477–485. https://doi. org/10.13182/NSE14-20.
- [7.] Y. Yan, C. Baldwin, J.M. Harp, A. J. James, K. D. Linton, and N.A. Capps, SATS Transient Fission Gas Release Capability and Design, ORNL/SPP-2022/2663 (2022).
- [8.] C.A. Baldwin, J.D. Hunn, R.N. Morris, F.C. Montgomery, C.M. Silva, P.A. Demkowicz, First elevated-temperature performance testing of coated particle fuel compacts from the AGR-1 irradiation experiment, Nucl. Eng. Des. 271 (2014) 131–141. https://doi.org/10.1016/j.nucengdes.2013.11.021.

Performance of the HERA PreHydrided Experiments

Principal Investigator: Charlie Folsom, Jason Schulthess, David Kamerman Team Members/Collaborators: J. Stinger, J. Burns, P. Petersen, M. Ramirez, S. Seo, C. Jensen, D. Wachs

> daho National Laboratory (INL) in collaboration with the Japan Atomic Energy Agency (JAEA), under the Nuclear Energy Agencies' (NEA) Framework for Irradiation Experiments (FIDES) initiative have tested four fuel rodlets made of UO₂ pellets in Zirconium-4 cladding with a pre hydrided microstructure to simulate high-burnup conditions. The experiments were performed in Transient Reactor Test (TREAT) Facility at INL and the Nuclear Safety Research Reactor (NSRR) at JAEA as part of a larger campaign to evaluate the effect of pulse-width during a reactivity-initiated accident (RIA). It is hypothesized that the pulse width in an RIA may affect rod failure by pellet-cladding mechanical interaction, specifically, longer pulse widths may be less severe than shorter pulse widths.

Project Description:

The RIAs are a postulated accident condition in light-water nuclear reactors. In these accidents, the rapid ejection of a control rod from the core results in a reactivity addition and a rapid rise of power. One potential consequence of this rapid rise of power is a condition, known as pellet-cladding mechanical interaction (PCMI), in which the rapid rise of power causes thermal expansion of the fuel pellets. The pellets then contact the still relatively cold cladding and cause failure of the cladding due to excessive strain. To evaluate this potential failure mode, a series of experiments were developed

called High-burnup Experiments in Reactivity-initiated Accidents (HERA). The purposes of this test series include evaluating PCMI presence and severity at different pulse widths on fuel rods specifically manufactured to be representative of high-burnup fuel rods, namely Zircaloy cladding with hydride rims and sized to reduce the fuel-cladding gap.

The HERA project will include multiple experiments in the TREAT reactor with a ~90 ms full-width-athalf-max (FWHM) pulse and NSRR with a ~7 ms FWHM pulse width. The experiments are targeting test conditions to achieve fuel radial average enthalpy increases during the transient around the current failure limit of 711 J/gUO₂.

A modeling and simulation (M&S) exercise was also organized under the HERA project. The M&S exercise included participants from over 20 organizations around the world using 14 different fuel performance codes. The goal of the M&S exercise was to provide pre-test blind predictions to help design the targets for the actual experiments. The M&S work will continue with post-test as-run results that will provide valuable validation data for codes.

Accomplishments:

Experiment design, fabrication and irradiation was completed for the first two TREAT tests in this series, which included fresh fuel pellets assembled into rodlets with precharged hydrogen in the clad-



ding [1]. The precharging of the hydrogen in the cladding resulted in rods that contained ~400 ppm hydrogen with Zirconium hydrides forming in the circumferential direction in the cladding (See Figure 1; [1]). Two rodlets were fabricated at INL and shipped to NSRR where they completed the first experiment in December 2022 and the second in September 2023. Following the two TREAT experiments, initial post-transient examinations were performed and identified that the rodlets were intact. The rodlets did experience some permanent strain deformation caused by the transient, but the amount of strain was insufficient to cause failure. Strain for the PreH-3 rodlet was an average of 1.2% while for PreH-4 was 0.7%. Figure 2 shows HERA-PreH-4 rodlet inspection Figure 1. Prehydrided cladding similar to that used to build the PreH test specimens, showing elongated circumferential hydrides and higher concentration of hydrides near the outer rim Figure 2. HERA-PreH-4 rodlet inspection during experiment disassembly. The arrows point to locations for the four thermocouples that were found still attached



during experiment disassembly. These two experiments targeted 700 J/gUO₂ fuel radial average enthalpy increases. The future tests planned in TREAT will target higher enthalpy targets to fail the fuel rods.

The first experiment at NSRR targeted 625 J/gUO₂ fuel radial average enthalpy increase and did not appear to cause failure. Detailed post-transient examinations are in progress for this specimen. The second NSRR experiment was just completed with a targeted enthalpy of 725 J/gUO₂.

There was a lot of participation in the modeling and simulation exercise. A wide array of participants and fuel performance codes used provides a good estimation of the variability of possible results during a specific experiment. The modeling and simulation exercise included 14 cases with variations in energy deposition, cladding hydrogen content, and fuel-cladding gap. The nominal case for NSRR focused on a 7.5 ms FWHM pulse targeting 650 J/gUO₂ peak fuel radial average enthalpy rise with 400 ppm hydrogen content in the cladding. The TREAT had identical conditions but with a 90 ms FWHM pulse. Figure 3 shows the fuel radial average enthalpy result for the NSRR and TREAT cases. The

legend shows all the participants and their respective code used along with the peak values predicted. All codes do a good job predicting the same peak radial average enthalpy value, but the evolutions vary mainly due to differences in thermal hydraulic models between codes and users. Figure 4 shows failure predictions for the NSRR and TREAT cases. A value greater than unity means the code predicts that failure would occur. Most codes predict that the NSRR case would fail whereas not many codes predicted the TREAT case would fail.

As-run simulations using as-built dimensions and conditions for the experiments will be run once post-transient examinations of the experiments are complete. This exercise will be a valuable tool for code validation.

References:

[1.] D. Kamerman (2023) "Formation and characterization of hydride rim structures in Zircaloy-4 nuclear fuel cladding" Journal of Nuclear Materials Vol 586. https://doi.org/10.1016/j. jnucmat.2023.154675 The HERA project is generating valuable data on reactivity-initiated accident experiments that is enhancing our knowledge of pulse width impacts on fuel performance and safety limits of light water reactor fuel rods.



Figure 3. Evolution of fuel radial average enthalpy during the transient for a case targeting 650 J/gUO2 fuel radial average enthalpy increase: (a) for NSRR pulse, (b) for TREAT pulse



Figure 4. Cladding failure prediction defined as the maximum failure parameter over the critical value for predicted failure for a case targeting 650 J/gUO₂ fuel radial average enthalpy increase: (a) for NSRR pulse, (b) for TREAT pulse

TWIST LOCA Device Assembly and Commissioning

Principal Investigator: Colby Jensen

Team Members/Collaborators: Klint Anderson, Cindy Fife, Connor Michelich, Todd Birch, Ashley Lambson, Charles Folsom, Robert Armstong, Clint Wilson, Kelly Ellis, Changhu Xing, Nicolas Woolstenhulme

The completed assembly and commissioning of the TWIST device provides the U.S. with a LOCA and HBu LWR transient test device to perform critical safety testing for advancing LWR fuel.

oss of coolant accident (LOCA) testing is critical to support ongoing efforts to extend the allowable fuel burnup limits in light water reactors (LWRs). This testing also allows continued research on fuel fragmentation relocation and dispersal (FFRD). To address industry needs, and to help eliminate capability gaps resulting from the closure of the Halden Boiling Water Reactor (HBWR), a LOCA testing device was developed for use in the Transient Reactor Test Facility (TREAT). The final design for this device has been completed, components have been fabricated, the first experiment has been assembled, and the commissioning of the device has commenced in TREAT.

Project Description:

The Transient Water Irradiation System in TREAT (TWIST) provides a controlled environment in which a blow-down event can be simulated by actuating a valve to release water from the upper capsule containing a fuel specimen into an expansion tank below. The approach will fill the in-pile LOCA testing gap left by HBWR and adds capability to study stored energy heating effects resulting in temperature ramp rate effects, never evaluated before on irradiated LWR fuels. These behaviors are expected to play important roles in FFRD, the primary challenge to extending the regulatory burnup limit beyond 62 GWd/t. The unique design of the TWIST device allows highly representative testing using nuclear

heating, with achievable rod powers and temperatures prototypic of LOCA events. Figure 1 shows the TWIST experiment assembly, including the capsule and expansion tank.

The TWIST capsule was designed to be the LWR testbed for TREAT experiments with pre-irradiated fuel and is planned to be used for LOCA and reactivity-initiated accident (RIA) experiments on High-Burnup (HBu) fuel. The TWIST capsule accommodates rodlets with fueled lengths ranging from 25 to 50 cm and features a stateof-the-art instrumentation package to collect relevant time-dependent data for post-test analysis of the experiment. The commissioning test instrumentation package includes an electroimpedance sensor to detect phase change events in the water and cladding radial dimension, a pyrometer, thermocouples to measure cladding and surrounding environment temperatures, optical pressure sensors, and an acoustic emission sensor to detect cladding rupture. The design also includes options to measure either the fuel centerline temperature or rodlet pressure.

The commissioning test series in TREAT will be executed with fresh fuel specimens to demonstrate and qualify the TWIST system in preparation for HBu experiments, confirm power coupling in all planned experiment configurations, and provide detailed thermal hydraulic validation. Testing of pre-irradiated fuels in TWIST for the High burnup Experi-



ments for Reactivity initiated Accidents project is planned to commence following completion of the TWIST commissioning series.

Accomplishments:

Following completion of the final design review in 2022, the project team planned the fabrication and assembly of the experiment modules to support the commissioning test series of the TWIST device. Due to the number of fabricated components in TWIST, parts were fabricated both at Idaho National Laboratory (INL) machine shops and by external vendors. After completion of fabrication, components were welded, examined, and tested at INL prior to performing assembly. Figure 1. TWIST experiment assembly



Figure 2. TWIST assembled fuel specimen

The module and instrumentation assembly for the first TWIST commissioning test was completed at INL where instrumentation could be tested and calibrated during assembly. Following instrumentation assembly, all components were shipped to the Materials & Fuels Complex where the fuel specimen was fabricated, welded, and assembled into the TWIST device. Figure 2 shows the TWIST components and assembled fuel specimen. Thermocouples were attached to the specimen cladding with clips welded to the cladding tube to measure the specimen cladding surface temperature response. Figure 3 shows the thermocouple attachment being performed. After thermocouple attachment the specimen was inserted into the



TWIST capsule as shown in Figure 4 to complete the TWIST module assembly for the first commissioning test.

The first commissioning test on the TWIST device was successfully completed in September 2023. This milestone transient represents the culmination of several years of work by a large multidisciplinary team and officially provides the United States with a testing capability for LOCA events. This capability allows critical safety testing for advancing LWR fuel and supports continued research on FFRD. It also provides the needed capability to test larger fuel segments than TREAT capsules used to date since restart.

The TWIST device was designed for hot cell assembly and disassembly to support HBu testing. The Figure 3. Attachment of the thermocouples to the TWIST specimen cladding





Figure 4. TWIST capsule assembly

successful fabrication, assembly, and commissioning of the TWIST device provides experience and expertise to the project team. This expertise can be leveraged for upcoming RIA and LOCA HBu testing. The renewed in-pile LOCA capability and HBu LWR testbed that TWIST provides is expected to fill a critical gap in testing to support ongoing efforts to advance LWR fuel for the U.S. nuclear energy fleet.



High Burnup Fuel Loss-of-Coolant Accident Experiment Design

Principal Investigator: Robert Armstrong

Team Members/Collaborators: Colby Jensen, Klint Anderson, Charles Folsom, David Kamerman

he U.S. nuclear industry is aiming to extend the licensed burnup limit of nuclear fuel in light water reactors (LWRs) from 62 GWd/t up to approximately 75 GWd/t. To increase regulatory burnup limits, a licensing basis is required to ensure all regulatory requirements are met. The highest priority research and development need to support burnup extension, as determined by the Collaborative Research on Advanced Fuel Technologies Fuel Performance and Testing Technical Experts Group, is the behavior of high burnup (HBu) fuel during a loss-of-coolant accident (LOCA) due to a phenomenon that has been termed fuel fragmentation, relocation, and dispersal (FFRD). While an impressive amount of integral-type and separate-effect experiments studying HBu fuel behavior under LOCA conditions have been performed to date, data gaps regarding FFRD remain. To address the needs required for burnup extension, integral LOCA experiments on HBu fuel to be performed in the Transient Reactor Test Facility (TREAT) at Idaho National Laboratory have been designed.

Project Description:

Enabled through the combination of increased fuel enrichment and extending the licensed fuel burnup limit, the U.S. nuclear industry aims to enhance the economic efficiency and safety of LWRs. Through the use of increased enrichment and higher burnup, fuel management strategies that were not previously viable, such as extending operating cycle length, are now attainable. Such a transition has both economic and safety benefits as less frequent outages keep reactors running and reduces the dose exposure to outage workers and potential for outage-related events.

As part of the licensing basis for burnup extension, the behavior of HBu fuel during a LOCA must be addressed. To investigate the behavior of HBu fuel during Design Basis Accidents including LOCA, the Advanced Fuels Campaign (AFC) has developed an experiment vehicle known as the Transient Water Irradiation System in TREAT (TWIST). After a series of fresh fuel commissioning tests, HBu LOCA experiments will be performed in TWIST. These tests are a part of the AFC Integral LOCA test plan which also includes furnace tests to be performed in the Severe Accident Test Station at Oak Ridge National Laboratory.

One of the main objectives of HBu LOCA testing in TWIST will be to assess FFRD under conditions representative of large break LOCAs (LBLOCAs). In a pressurized water reactor (PWR) LBLOCA, the coolant in the core rapidly begins to void following break initiation. This results in a drop in the ability to remove heat from the fuel rods. Core coolant voiding also causes the reactor to go
The TWIST experiment design enables a first-of-a-kind capability to test high burnup fuel under representative LBLOCA conditions.



Figure 1. Illustration of the two-SP design of the TWIST LOCA Device







Figure 3. Fuel radial temperature distribution during various times of the transient. Solid Lines: TWIST; Dashed Lines: PWR

subcritical, which leads to a significant reduction in heat generation in the fuel. This rapid loss of cooling and heat generation causes the stored energy within the fuel to redistribute radially, resulting in a rapid temperature decrease of ~100°C/s at the center of the fuel, while the temperature of the outer portion of the fuel and the cladding increase rapidly at approximately the same rate. This has been termed stored energy heatup (SEH).

To date, LOCA experiments driven by a large SEH have never been performed on HBu fuel. Currently, only experiments which are consistent with a decay energy heatup (DEH) have been performed on HBu fuel. In DEH conditions, the fuel is isothermally heated at a rate of ~5°C/s. The ability to simulate SEH conditions, representative of LBLOCA scenarios, within the TWIST design will provide extremely important insights as to how the thermomechanical response of the fuel and cladding impact FFRD behavior.

Accomplishments

As the primary goal of the TWIST HBu LOCA experiments is to investigate FFRD under representative LOCA conditions, these conditions needed to be determined. To do this, a 4-loop PWR LBLOCA was simulated using the TRACE thermal-hydraulic systems code. The model included a core-wide power and burnup distribution consistent with that of a PWR core at the end of a 24-month cycle. The TRACE model was tightly coupled to the BISON fuel performance code using the BlueCRAB MOOSE-wrapped application. This tight coupling enables detailed simulation of the thermomechanical response of a high burnup fuel rod during an LBLOCA.

These representative conditions were then used as a target for the TWIST HBu LOCA experiments. Through an iterative effort between the mechanical design modifications and neutronic, thermal-hydraulic, and fuel performance analysis, excellent agreement between the PWR LBLOCA and TWIST HBu LOCA simulations was achieved. To accomplish this, a two-state point (SP) approach is used. In State Point 1 (SP-1), the test fuel rod is brought up to temperatures consistent with PWR operating conditions. In State Point 2 (SP-2), a valve is opened, rapidly removing the water

surrounding the fuel. At the same time the power of TREAT is reduced to simulate decay heat within the fuel. An illustration of the two-SP design is shown in Figure 1.

Figure 2 shows the fuel centerline and outer cladding temperature histories for the PWR LBLOCA and TWIST HBu LOCA. Figure 3 compares the radial temperature distribution within the fuel at various times throughout the transient. In the case shown here, the fuel rod simulated had a burnup of approximately 70 GWd/t and was operating at a steady-state power of 23 kW/m prior to the LOCA; however, manipulation of TREAT reactor power and TWIST operating conditions, enables the ability to simulate a wide variety of temperature histories.

The design of the TWIST LOCA experiments will allow for first-ofa-kind testing on HBu fuel under representative LBLOCA conditions. This capability will provide insights into to the complex interactions governing FFRD behavior such as fuel stresses, transient fission gas release, and cladding ballooning and burst. The work performed here enabled the completion of the preliminary design of the TWIST experiment device for HBu fuel testing, a level 2 milestone. The progress made on the TWIST HBu experiment vehicle will enable irradiation of HBu fuel samples from the Byron Nuclear Generation Station fuel shipment which is scheduled to commence in the first half of fiscal year 2025.

High Temperature Cladding Creep Performance of Cr-coated Cladding

Principal Investigator: Mackenzie Ridley

Team Members/Collaborators: Samuel Bell, Danney Schappel, Dan Sweeney, Nathan Capps

r coated zirconium alloys are considered nuclear fuel I cladding candidates as an accident tolerant fuel cladding. This is primarily due to the low neutron absorption cross-section of Cr and Zr, the excellent oxidation resistance of Cr. and the improved mechanical performance during postulated accident conditions such as loss of coolant accidents (LOCA). Yet, the exact mechanism for the improvements witnessed during LOCA is not well understood. Developing a mechanistic understanding of cladding deformation during design basis accidents is critical for accelerating the qualification of accident tolerant cladding materials and for improvement of modelling efforts. This work involved modifications of the Oak Ridge National Laboratory (ORNL) Severe Accident Test Station (SATS) to enable capture of cladding deformation during simulated LOCA tests. Digital image correlation (DIC) was used to monitor strain in situ, and posttest analysis provided insights on the mechanisms for deformation of tubing under high pressures and rapid temperature transients.

Project Description:

This work involved establishing a reliable DIC framework at ORNL to investigate the reasons for the known benefits of Cr coatings during light water reactor design basis accidents. Specifically, this work focused on LOCA scenarios. where fuel claddings are subject to rapid heating rates in steam environments with a large pressure differential across the cladding. The objectives were to utilize in situ strain measurements in three dimensions to quantify temperature and stress-dependent parameters, relating to temperature and stress-dependent mechanisms for cladding deformation. Such parameter development will improve understanding on why Cr coatings, typically on the order of microns thick, show measurable delays in cladding rupture until higher temperatures. Additionally, this information will inform fuel cladding vendors on design space processes that are needed for development of alternative cladding concepts with enhanced rupture resistance during design basis accidents. Deformation parameters will be used to inform modelling efforts, such as BISON, so that qualification of accident tolerant fuel cladding candidates can occur more rapidly. Accelerated qualification of advanced nuclear materials will increase the competitiveness of nuclear power generation through enhanced safety margin, opportunities for increased fuel duty cycles, and opportunities for improved reactor



efficiencies. Research on accident tolerant fuel claddings inevitably improves safety margin for design basis accidents, but the additional cost savings through achieving higher fuel burnups is expected to

encourage development of nextgeneration reactors for the United States fleet. Figure 1. Zry-4 pressurized to 9.69 MPa and heated at 5°C/s, a. Digital image correlation camera image with hoop strain map overlaid, b. Hoop strain line profile as a function of temperature. Temperature was controlled via thermocouples and monitored with fiberoptic sensors design of the TWIST LOCA Device In situ strain measurements of Cr coated and uncoated Zircaloy-4 during simulated LOCA testing showed that creep and plasticity related deformation mechanisms can be quantified to support accelerated materials qualification for light water reactors.



Figure 2. Temperature dependence of Zircaloy-4 and Cr coated Zircaloy-4 from digital image correlation data, showcasing a change in deformation mechanism around 2%/s strain rate

Accomplishments

Uncoated and Cr coated zircaloy-4 cladding tubes were prepared for this work and were tested at ORNL's SATS. The Cr coating was deposited by an external vendor via high power impulse magnetron sputtering, resulting in a 7-micron thick Cr layer on the outer surface of the tubes. Cladding tubes were coated in a high-temperature paint for pixel tracking with DIC. LOCA tests were performed at a temperature ramp rate of 5°C/s, with varied internal pressurizations, until cladding failure. Temperature profiles were validated with fiberoptic sensors for select tests. Two cameras viewed the specimen during the tests through a viewport in the furnace wall. Time, temperature, and pressure data were then synced with in situ strain measurements from the camera system for DIC. The rapid temperature transient during LOCA testing, 5°C/s in this work, was utilized to calculate temperature dependences for the visible deformation processes. Additionally, a novel isothermal pressure-jump test was developed for gas-pressurized tubing such that the stress exponents could be measured rapidly from a single cladding segment. Mechanisms for Zr deformation, such as the activation energy for Zircaloy-4 creep and the stress exponent for Zircaloy-4 creep, were established in this work and were in agreement with literature values. Comparisons to Cr coated tubing showed that Cr coatings have similar mechanisms for cladding deformation as uncoated material, yet the rupture conditions were roughly 80°C higher with Cr coated Zircaloy-4. Cladding segment yield stress and ultimate tensile stress were also extracted from the in-situ strain data, showing that Cr coated tubing has roughly 80°C higher temperatures in mechanical performance or roughly a 40 MPa increase in yield stress compared to the uncoated baseline. Post-test characterization showed that the operating temperature governed the microstructure characteristics, and that the Cr coating did not impact the morphology of the Zircaloy-4 tubing. From such deductions, the performance benefits of Cr coatings on

zirconium alloy cladding tubes must be related to the hoop stress from the coating. Creep model parameters generated from this work were implemented into BISON for validation. From this work, research is now ongoing to determine the impacts of residual stress from coating deposition processes and verify the impacts of a residual compressive stress on the rupture properties.

HBu UO2 Transmission Electron Microscopy

Principal Investigator: Chad M. Parish Team Members/Collaborators: Nathan Capps, Lauryn Reyes, Casey McKinney

This project provides quantitative experimental data that will allow validation and benchmarking of predictive models for improving safety and efficiency of LWR fuel.

oss-of-coolant accident (LOCA) tests showed significant Ifine fragmentation in high burnup (HBu) fuel such that some of the fuel was reduced to a sandlike consistency. This HBu fuel fragmentation raised concerns of the fuel fragments relocating into the ballooned region of the cladding and possible dispersal from the cladding through the rupture opening, collectively known as fuel fragmentation relocation and dispersal. In this work, as-irradiated and LOCA-tested HBu fuel samples are examined in scanning/transmission electron microscopy (S/TEM) to correlate microstructural features. primarily fission gas bubbles (FGBs), to fragmentation behavior.

Project Description:

The goal of this project is to enable understanding and predicting how light-water reactor (LWR) fuel will behave under HBu and LOCA conditions. Here, we are obtaining quantitative microstructural data from as-irradiated and post-LOCA samples of HBu fuel. This microstructural data is essential input to modelling efforts to understand and predict the performance of HBu fuel during both normal and accident conditions. The HBu micro- and nanostructure is so complex (large grains, small grains, high- and low-angle grain boundaries, FGBs, five-metal particles [FMPs], etc.) that modelling the behavior under irradiation or LOCA conditions requires the quantitative insight to the structure obtained by these experiments.

Specifically, HBu fuel from North Anna 1 and North Anna 2 (NA1 and NA2) LWRs have been examined. Samples were prepared by focused ion beam (FIB) for S/TEM analysis and S/TEM was used to image grains, cavities, FGBs, and FMPs.

S/TEM methods were used to quantify the microstructures present in the as-irradiated and post-LOCA samples as a function of radial location. For different radial locations (where r/r0=1.0 is the fuel-clad interface and r/ro=0.0 is the center of the pellet) we examined grain structures, large pores (~microns) that were prior-Xe bubbles, the FMPs, and the nanometric FGBs. We are also developing and refining an analytical method to estimate the pressure of the Xe in the nanometric FGBs, because the pressurization of the FGBs is a vital parameter for modeling and predicting fuel behavior and experimental confirmation of the predicted pressures is vital to validate current models.

Accomplishments

The most important results from the present work are comparisons of as-irradiated and post-LOCA samples from approximately the same radial regions in NA1 samples, and secondarily, comparison to the as-irradiated NA2 samples. In all the samples, a combination of large and fine UO₂ grains are observed, intermixed with large (~microns) regions of free space that were large prior-xenon bubbles before the samples were thinned to electron



Figure 1. Comparisons of as-irradiated and post-LOCA tested samples at approximately the same radial distance, ~0.55 r/ r0. X-rays maps are computed as weight%, with 3×3 or 5×5 moving averaging prior to quantification. Regions of vacuum (i.e., panel f, top-right corner) are non-physically noisy due to zero count rates



Figure 2. Comparison of NA1 asirradiated and post-LOCA fission gas bubble sizes at similar radial positions. Vertical lines are the median

transparency. Examination using S/TEM X-ray spectrum imaging (XSI) using energy dispersive spectroscopy (EDS) finds the elemental distributions within these complex structures, and FMPs ranging from ~1-2 nm to a several hundred nm and found distributed through the structures, with small FMPs primarily intragranularly and larger FMPs scattered both intragranularly, on grain boundaries, and on the surfaces of the large open cavities. Further, small FGBs — which are pure Xe to the detectivity of EDS, ≈1 wt% — show a wide range of locations and sizes. Small (~1-5 nm) FGBs are present in the interiors of many grains, often in association with similarly sized FMPs; although, other grains show

few or no FGBs. Ongoing work is to determine if recrystallization / restructuring has occurred in these FGB-free grains. Medium-sized FGBs (5-40 nm) are present in grain interiors and on grain boundaries, usually in conjunction with similarly sized FMPs.

Figure 1, as an example, shows the 0.55~0.57 r/r0 (approximately 1800 μm from the cladding) region. This region is in the central restructured region of the pellet, near the border with the mid-radial region. In the top row of Figure 1(a)-(c), the general microstructures are seen. In the as-irradiated samples, bubbles are seen that were presumably filled with fission gas prior to FIB preparation. Grain boundary cracking is also apparent (red arrow). Several smaller (~half micron) grain boundary or triple point bubbles are also visible, i.e., yellow arrow in the NA1 as-irradiated figure. In the post-LOCA, similarities and differences are noted. The number of small high-aspect, lenticular bubbles appear higher, and they seem less associated with the grain boundaries and more intragranular; green arrow. EDS mapping, Figure 1(d)-(i), show the polydispersity of the FMPs (yellow/white) and the FGBs (red). Also of note, and in need of further study, in the medium sized lenticular bubbles post-LOCA sample show depletion of FGBs for ~100-200 nm around the lenticular bubbles. The denuded zone is interesting and in need of exploration. Other examined radial distances (~0.53, ~0.77 r/ro) showed



similarities to these. Quantification of the FGB sizes, Figure 2, indicate significant scatter within each dataset, and large region-to-region variations within the same nominal sample, so conclusions regarding FGB size distributions are in need

of further explanation. Finally, estimates of FGB bubble pressures Figure 3) for as-irradiated and post-LOCA samples have been made and compared, and indicate no obvious trends. Bubble pressure work is still actively ongoing. Figure 3. As-irradiated and post-LOCA bubble pressure estimates for the NA1 0.57 r/r0 conditions. Note the very high bubble pressures (very small diameters) in the maps are non-physical and removed from the scatterplots

Investigation of Degradation Mechanisms of Cr-coated Zirconium Alloy Cladding in Reactivity Initiated Accidents

Principal Investigator: WooHyun Jung, University of Wisconsin-Madison (UWM) Team Members/Collaborators: Hwasung Yeom (Pohang University of Science and Technology [PUST]), Kumar Sridharan (UWM), Brent Heuser (University of Illinois Urbana-Champaign [UIUC]), James Corson (Nuclear Regulatory Commission)

> he Cr-coated Zr-alloy is expected to provide additional safety margins by virtue of its excellent corrosion/ oxidation resistance at accident conditions as a near-term accident tolerant fuel (ATF) cladding design. Recently, there was a data gap reported for licensing activities and performance criteria including degradation of the coated cladding above the Cr-Zr eutectic temperature, post-quench ductility, and coating integrity during swelling/ rupture. This research investigates the thermal, mechanical, and irradiation response of Cr-coated cladding tubes under prototypical reactivity-initiated accident (RIA) conditions, compared to uncoated cladding. Two different Cr-coating methods have been investigated, the cold spray (CS) process and the physical vapor deposition (PVD) process. The RIA tests have been performed at the Transient Reactor Test Facility (TREAT) at Idaho National Laboratory (INL). The tests impose power excursions on the Cr-coated cladding/UO₂ fuel system followed by comprehensive postirradiation examination (PIE). This work will extend our knowledge of the Cr-coated Zr-alloy cladding under the design-basis accidents.

Project Description:

The objective of the research is to investigate the thermal, mechanical, and irradiation responses of Cr-coated Zr-alloy claddings under RIA conditions, in comparison to uncoated Zr-alloy cladding. The objective is achieved by a pulse-type nuclear heat deposition on the Cr-coated cladding/UO₂ fuel system followed by comprehensive PIE with two different Cr coating methods: the CS and PVD processes. The proposed TREAT experiment focuses on demonstration of various cladding failure modes at the later phase transient (post-departure from nucleate boiling), including ballooning/burst, severe high temperature oxidation, partial melting due to the Cr-Zr eutectic, or a combination thereof. In-situ monitoring during the testing includes water and cladding temperature, fuel pellet temperature, cladding internal pressure, and cladding elongation. Through this work, the hypothesis is tested that degradation or failure mechanism of the Cr-coated Zr-alloy cladding is different from that of uncoated conventional Zr-alloy cladding in the high temperature phases of RIA transients. For example, the CS process relies on severe plastic deformation of powder particles and substrate surface, resulting in compressive stress in both the coating and the underlying substrate. These compressive stresses may influence



stress evolution during rapid power transient. Possible alteration of the damage/failure phenomena of the coated cladding under RIAs due to the chemical interactions and phase transformations must be important in future licensing and commercialization. Success of the proposed research will generate valuable experimental data for the response of Cr-coated Zr-alloy cladding tubes under RIA conditions, which then can be compared with future experimental results of irradiated high burn-up Cr-coated Zr-alloy cladding tubes currently being irradiated at commercial reactors. The result of this project will be used for anticipated licensing applications to the U.S. Nuclear Regulatory Commission (NRC), thereby accelerating deployment of Cr-coated ATF concepts in U.S. commercial light water reactor (LWR) fleets.

Accomplishments:

To achieve the objective of this research, five primary task areas have been identified: (1) sample fabrication for TREAT testing, (2) irradiation capsule design and fabrication, (3) TREAT testing for Cr-coated cladding tubes, (4) material characterization and mechanical test after TREAT testing, and (5) analysis of data and modeling of failure mechanism of Cr-coated cladding tubes. In the first stage of the project, the test matrix for TREAT testing was established to demonstrate potential late phase high temperature RIA failures such as cladding ballooning and burst, oxidation embrittlement, and cladding melting by adjusting the target peak cladding temperature, total energy deposited, and cladding internal pressure. A total of six RIA tests are planned: two uncoated Zr-alloy cladding as a reference test, three CS Cr-coated Zr-alloy cladding, and one PVD Cr-coated Zr-alloy cladding. Sections of Zircaloy-4 cladding tubes were coated at the UWM using the cold spray deposition facility as Figure 1. The as-deposited CS Cr-coated cladding was polished to achieve the smooth surface roughness and it was confirmed using Scanning Electron Microscopy (SEM) photography as shown in Figure 2. Several uncoated Zircaloy-4

Figure 1. Photographs of (a) cold spray coating system used for the Cr coating deposition and (b) sections of Zircaloy-4 cladding tube before and after Cr coating deposition Figure 2. Cross-sectional SEM image of cold spray Cr-coated cladding; (a, b) longitudinal cross-sections and (c, d) radial cross-sections with different magnifications



Figure 3. Schematic illustration of test capsule design: (from left) Overall capsule assembly in buster, sectional view of capsule assembly in buster, High Burnup Experiments in Reactivity initiated Accidents (HERA) capsule assembly, HERA specimen holder overall assembly, and HERA specimen holder assembly





Figure 4. Measured reactor power pulse and the centerline temperature of uncoated Zircaloy-4 cladding sample during the transient imposed by the TREAT facility

and CS Cr-coated Zircaloy-4 tubes were delivered to the TREAT team at INL to assemble the test capsule. called Static Environment Rodlet **Transient Testing Apparatus** (SERTTA), for proposed RIA tests (Figure 3). A total of five test vehicles have been manufactured at INL and the test sample consists of a stack of ten pellets with eight UO₂ pellets and a zirconia insulator pellet on the top and bottom of the stack. The top three pellets have their centerlines drilled out to insert a high temperature type C thermocouple to measure the cladding centerline temperature during a transient experiment. The cladding encasing the pellets has four bare wire intrinsic junction thermocouples spot welded to its surface at two axial elevations. The test capsule is also equipped with two pyrometricbased water pressure sensors, a pair of capacitive boiling detectors, and a rodlet pressure detector. The PVD Cr-coated cladding tube, currently under preparation by project collaborator UIUC, will have a test capsule manufactured in late

The result of prototypical RIA experiments for Crcoated Zr-alloy designs in TREAT facility will be used for anticipated licensing applications to the U.S. NRC, thereby accelerating deployment of Cr-coated ATF concepts in U.S. commercial LWR fleets

2023. Two RIA tests with uncoated, reference Zircaloy-4 samples were conducted in June 2023. Figure 4 shows the reactor power transient and measured centerline temperature of a fuel rod sample. The estimated energy deposition was ~ 1000 J/gUO₂ and Full Width at Half Maximum pulse duration was ~ 90 ms. PIE will be performed in early 2024 including visual inspection, ring compression test, SEM fractography, and cross-sectional SEM. The rest of the four test capsules will be tested by early 2024.

Analysis of Fuel Utilization for Microreactors

Principal Investigator: Arantxa Cuadra Team Members/Collaborators: Cihang Lu

> n assessment of the viability and potential attractiveness of advanced nuclear fuels must include consideration of how proposed concepts will impact the nominal reactor performance and safety characteristics. This assessment includes neutronics analyses to evaluate the impact on performance parameters (e.g., cycle length/burnup, power distributions, etc.), safety-related characteristics (e.g., reactivity and control coefficients/worths, kinetics parameters, etc.), and performance in a broad spectrum of transient and accident scenarios. Over the past decade, Brookhaven National Laboratory (BNL) has developed and benchmarked a scoping methodology to perform an initial screening of the performance and safety of candidate advanced fuel concepts, in support of the Advanced Fuels Campaign and its predecessors, recognizing that the fuel and reactor are an intimately linked system.

Project Description:

Advanced fuels, including ceramic fuels, have the potential to fulfill the needs of multiple reactor designs. This study assesses the impact of using uranium nitride (UN) fuel on the neutronic performance of a thermal-spectrum microreactor, specifically the Heat-Pipe Microreactor (HP-MR) designed by Argonne National Laboratory. Cooled by heat pipes and moderated by yttriumhydride, HP-MR consists of 30 hexagonal fuel blocks surrounded by hexagonal beryllium reflector blocks and 12 control drums. Employing traditional uranium oxycarbide (UCO) Tristructural Isotropic (TRISO) fuel compacts with a U-235 enrichment of 19.75 wt.% and a fuel to compact volume ratio of 4.9%, the baseline HP-MR is capable of generating approximately 2 MW thermal for ~8 years, with a discharge burnup of ~13 GWd/tU. Additionally, eight UN cases, with solid fuel pellets and TRISO fuel compacts, and different levels of nitrogen-15 (N-15) enrichment, were modeled to better understand the trade-off between U-235 enrichment and N-15 enrichment. It is important to assess whether the reduction in U-235 that is anticipated from the use of UN in pellet form would counteract the disadvantages of N-15 enrichment, which is a costly process at the moment. The U-235 enrichment of the eight UN fuel options was adjusted such that the fuel cycle length remained unchanged (8-year lifetime).

Accomplishments:

The impact of retrofitting the baseline HP-MR with eight UN fuel options was assessed, compared with the baseline UCO TRISO fuel. Due to the higher mass density of UN (13.46 g/cm3 with a uranium mass density of ~ 12.7 g/cm3) than UCO (10.52 g/cm3, with a uranium mass density of ~ 9.4 g/cm3), the UN TRISO options required a lower



Figure 1. Cross-sectional view of the HP-MR, showing its power distribution (hot color) and neutron flux distribution (hot color)

Figure 2. Enlarged cross-sectional view of the active core, where TRISO fuel pellets, moderator rods, and heat pipes are distinguishable. The grey circles with dispersed red dots are the fuel pellets, where the red dots represent the TRISO fuel particles. The violet circles are the Yttrium-hydride moderator rods. The cyan circles are the heat pipes, where the inner circle represents the vapor region, and the outer ring represents the liquid region of the heat pipes



Figure 3. Change in keff of the reactor with different fuels (the reference UCO fuel and eight UN fuel options). It is seen that the keff of the UN-pelletfueled HP-MR options decreases more slowly throughout the cycle length and their initial excess reactivity is significantly lower, due to their increased U/U-235 contents. Note that a similar amount of U-235 is consumed during the same time interval regardless of the fuel option

> This neutronics screening of high-density ceramics fuels in a thermal-spectrum microreactor examines the trade-offs between total mass of uranium, U-235 enrichment, and N-15 enrichment, and can be used as the basis for a detailed cost/ benefit analysis.

U-235 enrichment (~20% less on average) than the baseline UCO fuel. Because the fuel/compact volume ratio of the TRISO fuel options was smaller than 5%, the fuel material volume in the UN solid fuel pellets was ~20 times higher, which resulted in the ~20 times lower discharge burnup. This also resulted in the ~6 times higher U-235 mass in the UN solid pellet fuels, despite their ~3 times lower U-235 enrichment. As the reactor power remained constant, a similar amount of U-235 was consumed during the same time interval regardless of the fuel option. Therefore, a smaller percentage of U-235 was consumed in the UN-pelletfueled HP-MR than the TRISO-fueled



HP-MR at the same burnup. Because of the neutron capture reaction of N-14 through the 14N(n,p)14C reaction, the N-15 enrichment and the U-235 enrichment of the UN fuel options can be negatively (and linearly) correlated.

The use of the different fuel options had a minor impact on the fuel reactivity temperature coefficients which remained ~ -3 pcm/K throughout the fuel cycle. Due to the high thermal conductivity of the TRISO compacts and UN fuel pellets, temperature differentials are small, and higher operating temperatures could easily be implemented without significant neutronic penalty.

A cost benefit analysis should assess the trade-off between U-235 and N-15 enrichments and consider whether the cost of U-235 enrichment to HALEU levels but with a small mass requirement (UN TRISO) is more costly than low U-235 enrichment (on the order of commercial level ~5%) but with significant mass requirements (UN pellets). Figure 4. BNL microreactor analysis team (Arantxa Cuadra and Cihang Lu)

Accident Tolerant Fuel Industry Advisory Committee 2023 Summary

Committee Chair: Bill Gassmann, Constellation Team Members/Collaborators: Daniel Wachs, Ed Mai, and Phyllis King

The Advanced Light Water Reactor Fuel Advisory Committee was established in 2012 to advise the Advanced Fuel Campaign's (AFC) National Technical Director on the direction, development, and execution of the campaign's activities related to accident tolerant fuels (ATF) and other advanced fuels for commercial light water reactors (LWRs). The industry advisory committee (IAC) is comprised of recognized leaders from diverse sectors of the commercial LWR industry. They represent the major suppliers of nuclear steam supply systems, owners/operators of U.S. nuclear power plants, fuel vendors, advanced reactor representatives, the Electric Power Research Institute (EPRI), and the Nuclear Energy Institute (NEI). Members are invited to participate on the committee based on their technical knowledge of nuclear plant and fuel performance issues as well as their decision-making authority in their respective institutions.

During the past year the committee provided important industry input to Department of Energy (DOE) regarding utility and fuel vendor perspectives on the potential



benefits of ATF and extending the burnup limit and licensed enrichment of current fuels; continued efforts in testing, evaluation and examination of new ATFs, especially relative to the lead test assemblies operating in numerous commercial plants and focused on the 'near term' doped UO₂ and coated clad concepts; steady state/transient testing infrastructure needs and gaps, in particular at Idaho National Laboratory and Oak Ridge National Laboratory and focused on test reactor and post-irradiation examination capability; used fuel transportation, packaging and receipt issues; future plans for metallic fuel testing and qualification; coordination between DOE and industry groups such as EPRI and NEI; and DOE assistance with industry responses to proposed Nuclear Regulatory Commission rulemaking.

The IAC meets monthly via teleconference and is currently chaired by William Gassmann of Constellation Nuclear. Additional members represent Framatome, Global Nuclear Fuels, Westinghouse, General Atomics, TerraPower, BWXT Nuclear, Dominion, Duke Energy, Southern Nuclear, EPRI, and NEI.



ATF Industry Teams Overview

Ed Mai

ndustry team activities continue across the nation to support the accident tolerant fuel mission. Commercial and irradiations occurred at Vogle, Calvert Cliffs, Byron, Hatch, and Clinton. Some of the lead test assemblies completed 24-month irradiations this year. Research reactor irradiations continued at the High Flux Isotope Reactor at Oak Ridge National Laboratory (ORNL), and the Advanced Test Reactor at Idaho National Laboratory (INL). Preparation for the irradiation of new cladding variants continued for both new short term (coatings) and long term (Silicon Carbide variants) technologies. New irradiations include those in both commercial and research facilities. Multiple teams are deploying concepts to reactors this and next year.

Post irradiation examinations on mid-life test pin performance and properties were completed at INL and ORNL. Industry teams also continued their efforts to prepare for a transition to high burnup and longer fuel life. Topical reports were submitted for performance code development, burnup limit extensions, and low enriched uranium+ implementation. License amendments to start upgrading production facilities for both cladding and enrichment increases; scaling up production capacity and infrastructure to manufacture industrial quantities of CR-coated cladding also occurred.



Accident Tolerant Fuel (ATF) Industry Teams – Westinghouse Electric Company FY23 Accomplishments

Principal Investigator: Ed J. Lahoda

Team Members/Collaborators: Westinghouse Electric Company LLC (Westinghouse); General Atomics (GA); Bangor University (BU); Idaho National Laboratory (INL); Los Alamos National Laboratory (LANL); Oak Ridge National Laboratory (ORNL); University of Wisconsin (UW); University of South Carolina (USC); North Carolina State University (NCSU); University of Virginia (UVa); University of Bristol (UBr); Constellation Energy Corporation (EC) and Southern Nuclear Company (SNC); Air Liquide (AL), Karlsruhe Institute of Technology (KIT); Royal Institute of Technology (KTH)

> estinghouse is working to commercialize accident tolerant EnCore®* fuel (ATF) designs which include advanced Cr coatings using nitrogen cold spray (NCS) on zirconium alloy cladding and SiGA®* silicon carbide (SiC) cladding with the capability of using higher 235U enriched ADOPTTM* fuel (doped UO₂) and U15N fuel (UN) to achieve burnups of around 75 MWd/kgU and/or substantial power uprates.

Project Description:

The Westinghouse ATF program is deploying lead test assemblies (LTAs) of Cr coated cladding with ADOPT and UO₂ pellets with greater than 5% enriched 235U higher with increased oxidation resistance to steam and air at design basis and beyond design basis accident conditions. The oxidation resistant cladding along with the greater than 5% 235U fuel and U15N fuel provide utilities with fuel capable of supporting economic 24-month cycles and/or significant uprates. This capability significantly lowers the cost of operation of nuclear plants while providing increased accident tolerance. GA is developing SiC cladding (SiGA) that provides high-temperature fuel pin integrity in severe accident conditions. Previous SiGA evaluation has demonstrated minimal corrosion at 1600°C in a steam environment steam oxidation ultra-high temperature (UHT) test at Westinghouse Churchill.

Accomplishments:

A topical report for fuel enrichments above 5% 235U was submitted to the Nuclear Regulatory Commission (NRC) June 28, 2023. The Vogtle ATF License Amendment Request for the >5% 235U LTA program was approved by the NRC on August 1, 2023. Several lots of Cr coated tubes have now been manufactured for the Vogtle LTA. Final 'Request for Additional Information' responses for the Incremental Burnup Topical Report were submitted to NRC and testing to support the lead test rod (LTR) program for Électricité de France is continuing. Data is being



Figure 1. Interface between uncoated and Cr coated portion of the cladding for Byron-2 Rod 47I

Figure 2. Second cycle Cr coated cladding pristine with excellent coating adherence and little indication of crud

obtained from the ORNL post irradiation examinations on the Byron-2 first cycle LTRs (Figure 1). Pool-side examinations of the second cycle Doel-4 and Byron LTRs were very positive with little or no crud accumulation (Figure 2). Re-insertion in both Doel-4 and Byron-2 for a third cycle was approved. Preparations for shipment of 25 irradiated rods from Byron to INL in December 2023 are continuing.

Departure from Nucleate Boiling testing was completed on NCS Cr coated cladding on both AXIOM and Optimized ZIRLO[™] substrates. The critical heat flux for the coated



Figure 3. Cr coating Optimized ZIRLO dramatically reduced the burst opening area between 700°C and 1150°C

Internal Pressure



cladding was comparable to uncoated cladding. NCSU completed validation of the Westinghouse thermalhydraulic codes for High Burnup Higher Enrichment fuel.

To validate the accident tolerant benefit of Cr coated cladding, Westinghouse has supplied Cr coated Zr cladding for a second set of bundle tests to be performed at KIT. The second test will reach peak temperature of ~1500°C. These test results will be used to validate the beyond design basis accident performance models in the MAAP5 and MELCOR codes. The results of these codes and probability safety analyses which take advantage of FLEX capabilities, will be used as the basis for the NRC to allow downgrading of some equipment classifica-

Figure 4. Temperature and pressure sensor assemblies prior to installation at ORNL HFIR Westinghouse EnCore fuel is a gamechanger for the nuclear industry because of significantly increased safety margins in severe accident scenarios, longer fuel cycles and/or large uprates, lower operating costs, and increased flexibility for fuel management.

tions, resulting in economic benefits for the utility. Most recent burst test data for Cr-coated Optimized ZIRLO showed reduced burst opening area during increasing temperature burst tests, where bursting occurred between 700°C and 1150°C (Figure 3).

Lower length scale modeling (ASM) is being used to develop ADOPT and UN fuel properties. U15N has a higher 235U content than UO₂ because of its 35% higher U density and a higher thermal conductivity that aids in reducing or eliminating fuel fragmentation, relocation, and dispersal. This work is being conducted by the USC, LANL, BU and KTH. UN samples were generated by LANL for testing in the BR-2 reactor in 2024. Work on the economic production of enriched 15N is being carried out by Air Liquide. Potential processes have now been identified and are being tested. Successful tests were carried out at the High Flux Isotope Reactor at ORNL (Figure 4) on the in-rod sensors being developed at Westinghouse. These in-rod sensors will enable real-time validation of the property data predicted by use ASM and enable significant time and cost savings in the licensing of new fuels.

Experimental work on the behavior and mechanical properties of Cr

coated cladding is being carried out by the UVa, UBr, UW, and the USC. This work indicates that nitrogen cold sprayed samples perform about the same as helium cold sprayed samples.

A set of six GAs' unfueled SiGA rods are in test within the Advanced Test Reactor (ATR), cycle 171A/B at INL. The rodlets are planned to be removed near the end of September 2023 after completing 120 days of irradiation. In parallel, GA-EMS has delivered six SiGA rods with surrogate fuel (molybdenum pellets) for irradiations in ATR cycles 173A/B. Equivalent SiGA rods are currently in autoclave testing at Westinghouse to demonstrate autoclave robustness of this configuration through 120+ days.

Ongoing irradiation test of fueled SiGA rods at INL and ORNL will be completed by GA with results reported under the SiC Cladding Development program (DE-NE0009235). Progress and results of future autoclave testing, UHT tests, and WALT (DNB and critical heat flux) tests of SiGA rods will be reported through GA collaboration with Westinghouse under the ATF Phase 2C program.

Accident Tolerant Fuel (ATF) Industry Team - Framatome

Principal Investigator: Matthieu Aumand

Team Members/Collaborators: Idaho National Laboratory (INL), Oak Ridge National Laboratory (ORNL), Kernkraftwerk Gösgen-Däniken. Constellation Energy, French Alternative Energies and Atomic Energy Commission (CEA), Paul-Scherrer Institute (PSI), Electricité de France (EDF).

> Framatome's Accident Tolerant Fuel (ATF) strategy relies on a two-phased approach to balance benefits with the anticipated timeline for fullcore deployment. PROtect Cr-Cr is Framatome evolutionary solution which brings incremental benefits compared to standard Zr-UO₂ system. The goal is to deploy this product in commercials reactors by mid-2020s. PROtect SiC is Framatome's breakthrough solution which offers significant benefits during beyond design basis accidents. A progressive implementation is envisioned for SiC-based solution with an initial goal to demonstrate proof-of-concept in test reactors by the mid-2020s.

> In response to Department of Energy (DOE) direction, Framatome further expanded the Enhanced Accident Tolerant Fuel (E-ATF) program to include high burnup and increased enrichment with the objective of increasing energy production and reducing the outage costs by minimizing the number of refueling outages.

Project Description:

The goal of DOE's E-ATF program is to develop an economical and more robust nuclear fuel design that will reduce or mitigate the consequences of reactor accidents while maintaining or improving existing performance and reliability levels in daily operations. After extensive testing, evaluation, and down selection, Framatome's technical approach addresses three focus areas: (i) Coatings for pressurized water reactor (PWR) and boiling water reactor (BWR) claddings, (ii) Chromia-doped and Chrome-variant UO₂ fuel pellets, and (iii) Silicon carbide (SiC) composite materials.

A dense Cr-coating on a zirconiumbased cladding substrate has the potential for improved high temperature steam oxidation resistance and high temperature creep performance, as well as improved wear behavior. Over the course of the E-ATF program, extensive processing and testing activities are being carried out on Cr-coated M5Framatome cladding in support of batch implementation by the mid-2020s.

Chromia-doped UO₂ pellets can improve pellet wash-out behavior after cladding breach and reduce fission gas release. The performance of this fuel has been extensively studied in out-of-pile and in-pile test programs. Chromia-doped fuel topical report for PWR application is currently under Nuclear Regulatory Commission (NRC) review. Building on both the experience gained and scientific knowledge achieved from the PWR Cr-coated cladding development, a coating material that is suitable for BWR application has been developed. Several out-of-piles tests were performed to ensure adher-



ence and coating quality. Currently, the lead test rods with BWR coating segments are being irradiated in Monticello reactor and 1st cycle hot cell post-irradiation examination (PIE) was planned in 2023.

Framatome's E-ATF pellet development is also focused on improving thermal-mechanical properties, especially thermal conductivity. Variants of Cr-doped UO₂ fuel pellets have been developed and out-of-pile testing showed a significant increase in thermal conductivity compared to UO₂. As part of the proof-of-concept, investigation has been focused on understanding evolution of thermal conductivity and microstructure under irradiation. Framatome fabricated and shipped rodlets containing chromium variant pellets for testing in ATF-1 experiment, which started in 2023.

Framatome is also developing a composite cladding comprising a silicon carbide fiber in a silicon carbide matrix (SiCf/SiC) for revo-

lutionary performance improvements. The objective is to develop a system which does not suffer from the same rapid oxidation kinetics of zirconium-based cladding while having attractive operating features such as reduced neutron absorption cross-section and higher mechanical strength at accident temperatures. Currently, Framatome fabricated test rodlets made of SiC-based composite for irradiation in Massachusetts Institute of Technology (MIT) and Advanced Test Reactor (ATR) test reactors. The MIT irradiation has started late 2022. The ATR irradiation is scheduled to start in 2024.

Framatome's innovative SiC-based cladding design solves critical feasibility issues and NRC concerns such as hydrothermal corrosion, fission gas retention and end plug sealing.

Accomplishments:

A significant amount of data from irradiation campaigns in both commercial and test reactors was generated. At INL, the Figure 1. Fractography analysis on the 15GWd/MtU mechanical testing specimens with a) post-failure ring tensile test specimen; b) close-up on a fracture surface; c) high-strain area near the fracture surface demonstrating cracks in the coating do not propagate into the substrate (credit: INL) An E-ATF LTA fully loaded with Cr-coated cladding and Chromia dopped pellet has achieved a 24 months irradiation cycle, a world first. With this achievement, and the implementation of its growing irradiation program, Framatome demonstrates the performance of its E-ATF technology. By pursuing licensing & industrialization in parallel, Framatome is bringing to the market a solution able to enhance fuel performance and safety.



Figure 2. The CCL ATF LTA Top- large field view of the CCL ATF LTA; Bottom – closeup on the endcaps

atc

PIE was completed on Cr-coated cladding irradiated at 30GWd/ MtU, providing data on all the key parameters required for licensing purposes. Mechanical testing on material irradiated in the ATR demonstrated the reliability of the Cr-coated M5Framatome design which exhibits similar properties as M5Framatome. Fracture surface observations on 15GWd/ MtU specimens demonstrated that cracks opening in the Cr at very large local strains stop propagating once reaching the Cr/Zr interface, leaving the Zr substrate to drive the deformation of the cladding without degradation originating from the Cr.

At ORNL, all irradiations in the High Flux Isotope Reactor were completed, and mechanical testing on Cr-coated cladding irradiated trough end of life doses was initiated. ORNL completed the first set of mechanical tests with axial tension testing at temperature, demonstrating similar properties between coated and non-coated cladding. At the PSI, the PIE on IMAGO specimens irradiated in the Gösgen nuclear power plant (NPP) in Switzerland collected after two and three cycles of irradiation demonstrated no hydrogen pickup is occurring beyond the 1st cycle of irradiation.

Irradiations on NPPs is ongoing with lead test rods (LTRs) in the United States (U.S.) and Europe, and a fuel assembly in the US. The E-ATF ATF LTA (Lead Test Assembly) inserted in Calvert Cliffs Unit 2 in 2021 completed its first cycle of irradiation (24 months) in February 2023. This LTA contains 176 Fuel rods manufactured with chromium-coated M5Framatome and chromia-doped UO_2 fuel pellets, making it the first 100% ATF fuel assembly. Framatome completed a detailed visual fuel rod inspection which reported excellent fuel performances and no degradation of the product. The fuel rods exhibited a golden color, which is attributed to very low corrosion kinetics. In addition, the fuel assembly was measured for fuel assembly growth with no unexpected results. The LTA was reinserted for its second cycle of irradiation. In Europe, Framatome expanded its irradiation program with the delivery of 36 lead test rod to an EDF NPP for irradiation.

Loss of Coolant Accident (LOCA) testing was carried out in CEA's EDGAR testing device on Cr-coated cladding. This test campaign demonstrated benefits of the Cr-coated cladding when compared to baseline material with a smaller balloon size and a smaller rupture opening size, especially at high temperature. This data will support the development of a LOCA swelling and rupture model.

Framatome is actively preparing for the commercial opportunities the Cr-coated cladding will bring. The design and purchase of an industrial pilot physical vapor deposition coating machine has been approved for deployment on the cladding production site of Paimbœuf, France, and will enable to produce large quantities of Cr-coated cladding. Licensing efforts are ongoing to enable Framatome codes and methods to support higher burnup and higher enrichments for optimal cycle operations. Framatome's U.S. fuel production site in Richland, WA is currently undergoing upgrades to allow for the production of high enrichment fuels above 5w% U-235.



Figure 3. EDF LTRs manufactured in 2023 at Framatome's Lingen site (Germany)

General Electric Progress in Developing Accident Tolerant Fuels

Principal Investigator: Rajnikant (RK) V. Umretiya

Team Members/Collaborators: Evan J. Dolley, Raul B. Rebak, Russ M. Fawcett, Allan Jaworski, Sarah Desilva, Rich Augi, David Barrientos, Dan Lutz, Tyler Schweitzer, Jason Harp, David Kamerman, Fabiola Cappia, Scarlett Widgeon Paisner

General Electric (GE) has been actively engaged in the research and development of accident tolerant fuels (ATFs) since October 2012. Their efforts are focused on various approaches to create a more resilient and cost-effective fuel that can endure accident scenarios, reduce hydrogen production, and maintain a coolable fuel structure for extended periods. These approaches encompass:

1. Exploring two cladding options, namely ARMOR or coated Zircaloy-2, and monolithic FeCrAl.

2. Enhancing the thermal and mechanical characteristics of the fuel.

3. Investigating fuels with higher fissile uranium enrichment levels exceeding 5% and burn-up limits greater than 62 GWd/MtU. (Figure 1)

Project Description:

The primary technical goals of the GE-led ATF Project are to develop materials that enhance the reliability of fuel rods in light water reactors (LWRs), improving both safety and cost-effectiveness. These advanced materials have the potential to extend the operational lifespan of existing reactors, preventing premature decommissioning. With nuclear energy expected to play a crucial role in mitigating climate change, via decarbonization, in the coming decades, these innovative materials, although untested in LWRs,

require comprehensive evaluation throughout the entire fuel cycle, from cost-effective fabrication to the final disposal of used fuel. GE is actively involved in assessing ATF fuel materials across the entire LWR fuel cycle (Figure 2). Collaboration with utility partners such as Southern Nuclear and Constellation. as well as national laboratories like Idaho National Laboratory (INL), Los Alamos National Laboratory, and Oak Ridge National Laboratory (ORNL), along with industrial partners like Alleima AB (formerly Sandvik Materials Technology), is ongoing. Importantly, the insights gained from studying materials like IronClad or FeCrAl cladding can also be applied to future reactor generations (i.e., Advanced Small Modular Reactors), as these alloys exhibit exceptional strength and oxidation resistance at temperatures exceeding 600°C. Additionally, this project is playing a pivotal role in training the next generation of nuclear materials scientists, ensuring the continuity of nuclear energy as a reliable source of clean power.

Accomplishments:

The accomplishments for GE can be summarized in the following areas:

Reported by Sarah Desilva and Allan Jaworski

During the fiscal year of 2023, coating cladding work continued research and development activities for an ARMOR 1.0 replacement. The ARMOR 1.5 initiative was suspended due to poor performance in the boiling water reactor (BWR) stability screening tests. Both the ARMOR 2.0 initiative and screening test methodology were advanced. A large number of screening tests were conducted on multiple ARMOR 2.0 concepts with mixed results. A down selection for ARMOR 2.0 was planned for Q1 2023, however, none of the concepts were mature enough to merit a single selection for optimization. ARMOR 2.0 concepts tested in FY2023 were a mix of new concepts and optimization of existing concepts. The down selection of ARMOR 2.0 will be revisited in the 2024 fiscal year.

The advancements in the characterization of IronClad (FeCrAl) continued during the 2023 fiscal year and most of the results were published in more than 10 articles. More manuscripts will appear in the following period.

Reported by Dan Lutz

Hotcell examinations of ARMOR and wrought C26M (IronClad) rodlets irradiated in ATR (ATF-2) have continued in 2023 at INL. Hotcell examinations of a wrought C26M lead test assembly (LTA) segmented rod irradiated in Hatch #1 have also continued in 2023 at ORNL. Notably, the hotcell examinations have focused on sample General Electric is actively working on enhancing fuel materials for light water reactors, aiming to improve their durability for both near-term (~5 years) and longer-term (~10 years) commercial deployment.



preparation, mechanical testing, and microstructural characterization of the earliest generations of the wrought C26M alloy class to establish the extent and cause of irradiation embrittlement. Figure 1. Technologies considered under GE-led ATF Program

Competitive Fabrication of Cladding Tubes	 FeCrAl uses inexpensive abundant metals Powder metallurgy technology, proven fabrication methods.
Normal Operation Conditions	 Boiling Water Reactors, 288°C, water + hydrogen or oxidants Pressurized Water Reactor, 330°C, water + hydrogen
Severe Accident Conditions	 Loss of Coolant Accident Steam or steam + air at T > 1200°C
Fuel Storage in Cooling Pools	 Localized corrosion behavior in water up to 60°C General corrosion in water with dissolved air up to 60°C.
Fuel Reprocessing	Dissolution behavior of fuel components in standard mineral acids.
Nuclear Waste Storage	 Dry Cask and Geological Repository Used fuel long term degradation performance.

Figure 2. The entire fuel cycle needs to be considered when evaluating newer materials, such as FeCrAl, for the ATF Program Preliminary ORNL results indicate that wrought C26M maintains a measure of residual ductility in the irradiated state at BWR operating temperature and has a microstructure that contains alpha' and loop and dot irradiation damage. A single Hatch IronClad lead test rod continues to operate in its third cycle in Hatch and a fourth and final cycle of operation is being planned to start in 2024, while one of three IronClad assemblies continued to operate in its 2nd cycle during 2023. Preparations were made in 2023 at Clinton #1 for poolside inspection of IronClad rods from three LTAs and transportation of seven IronClad segmented fuel rods that are planned for delivery to ORNL in early 2024.

Reported by Tyler Schweitzer

Interactions with the Nuclear Regulatory Commission (NRC) continue to be constant and productive. In August 2023, Global Nuclear Fuel (GNF) received the final Safety Evaluation Report from the NRC on the three topical reports for Low Enriched Uranium (LEU+): Lattice Neutronic Characteristics Evaluation & Research Code (LANCR) License Topical Report (LTR), LANCR Model Description LTR, and LANCR/ PANAC Application LTR. GNF has responded to Requests for Additional Information (RAIs) for the SNM-1097 License Amendment to the NRC to enable the GNF-A fuel fabrication facility to handle LEU+ and the approval is expected later

in 2023. The LANCR implementation LTR was submitted late in 2022 and the NRC has indicated that there will be no RAIs on the topical report. Once the Audit of the LANCR Implementation LTR work in 2024. GNF will be mostly licensed to handle LEU+ from an engineering perspective. GNF received the approval from the NRC for the RAI-II fresh fuel container to ship LEU+ bundles in July 2023. GNF updated the NRC in August 2023 with the overall LEU+/high burnup (HBU) licensing plan and program trajectory.

GNF continues to progress on updating methodologies to enable LEU+ and HBU. GNF is working to prepare an update to the PRIME thermal mechanical code and topical report. This update is needed in order to enable HBU and LEU+. GNF is working to prepare a separate topical report independent of PRIME to cover all other methods needed for HBU licensing and will address the issues raised in NRC RIL-2021-13 as well as hydrogen for higher burnup.

Reported by David Barrientos and Rich Augi

GE has had extensive positive interaction with the utilities, mainly Southern Company and Constellation, which have been fully supportive on GE's current direction. The collaborative approach working with these two companies is on the current LTAs (Clinton and Hatch), and the HBU Lead Use Assembly (Limerick) to make sure GE can obtain as much valuable information as possible. The utilities have also contributed to conversations with Department of Energy, as we continue progressing through the funding reduction and product trajectory. Deployment for ARMOR and LEU+/ HBU will be closer to the end of the decade. IronClad has an estimated deployment date closer to mid-2030s. GE continues working with our utility partners and national laboratories getting as much information from LTAs and advancing our methods to support the data that is needed for all the licensing topical reports to ultimately deploy each technology.

ATF Industry Teams: General Atomics – Electromagnetic Systems (GA-EMS) Accomplishments

Principal Investigator: Ryan Hon

Team Members/Collaborators: Structural Integrity Associates Inc., Constellation Energy Generation

The Silicon Carbide (SiC) Commercial Irradiation program will accelerate commercialization of GA-EMS SiGA® accident tolerant fuel cladding by facilitating a first-of-a-kind commercial irradiation of SiGA® cladding in a commercial nuclear reactor. The Silicon Carbide (SiC) Commercial Irradiation program will facilitate a firstof-a-kind irradiation of GA-EMS' SiGA® silicon carbide composite (SiC-SiC) cladding in a commercial nuclear reactor. The program will complete the design, manufacturing, and preliminary licensing of an irradiation vehicle that will be inserted into a commercial pressurized water reactor (PWR) to provide SiC performance data that will drive SiGA® cladding development and commercialization.

Project Description:

The SiC Commercial Irradiation program's objective is to complete all of the necessary steps for a follow-on irradiation of SiGA® cladding in a commercial PWR. After completion of the program, SiGA[®] test articles will undergo one cycle of irradiation before being removed and sent for post irradiation examination (PIE). The PIE will provide data on SiGA[®] performance under prototypical PWR conditions including both corrosion and irradiation effects. This data, along with the lessons learned while designing, licensing, and implementing the irradiation, will provide a basis for future insertion of lead test rods. lead tests assemblies, and commercialization of SiGA[®] cladding. To facilitate the irradiation, GA-EMS is working with Constellation to target a Spring 2025 reactor insertion.

As lead on the program, GA-EMS is designing, manufacturing, and characterizing the irradiation vehicle and SiC test articles. GA-EMS is subcontracting with Structural Integrity, who is providing modeling and licensing support. Constellation, as a partner, will provide a commercial reactor to host the irradiation as well as provide licensing and analysis support. During the execution of the program, a fuel vendor will be brought in to facilitate the physical insertion of the irradiation vehicle into the reactor after program completion.

This project is part of a larger SiC-SiC cladding development effort at GA-EMS that includes funding via the Accident Tolerant Fuel (ATF) program, the Industry Opportunities for Advanced Nuclear Technology Development Program, and internally funded work. Data obtained following this project will be combined and correlated with data obtained through irradiation of SiGA[®] in test reactors to support follow on commercial reactor irradiations. The project accelerates the development of SiC cladding which has the potential to increase the safety and economics of the current reactor fleet due to its high-temperature strength and irradiation stability.
Accomplishments:

In the last year, GA-EMS has completed the conceptual design of the irradiation. A trade study was performed, in consultation with utilities and fuel vendors, to identify the optimal irradiation vehicle approach. The trade study identified modification of the guide tube thimble plug assemblies currently used in some commercial pressurized water reactors as the best solution. Housing the SiGA® test articles within the thimble plug assembly, located at the top of the core, has been shown through neutronics calculations to have insignificant impact on reactor flux and power levels. Preliminary drawings have been completed for two separate capsule designs which will replace the guide tube thimble plug fingers. The capsules have minimal change to the form, fit, and function of the guide tube thimble plug assembly. The capsules will be inserted into the reactor for one cycle of irradiation. To simplify licensing (through a 10 CFR 50.59 process) and implementation, the SiGA[®] test articles will not contain fuel. GA-EMS has designed the first capsule as a pressure vessel meeting American Society of Engineers Boiler and Pressure Vessel Code standards to insulate SiC test articles from the reactor coolant chemistry and pressure to investigate irradia-



Figure 1. Conceptual layout of the irradiation vehicles with SiC test articles to test mechanical performance and corrosion of SiC after commercial PWR irradiation

tion effects on the SiC test articles. The second capsule has the same dimensions as the first capsule with small penetrations to allow for interior coolant flow to investigate SiC corrosion under commercial PWR conditions. The conceptual layout of the irradiation capsules with SiC test articles is shown in Figure 1.

Preliminary drawings have also been completed for five different SiC test articles. GA-EMS down selected these test articles based on data needs and post irradiation examination planning developed in the past year. The first capsule will have four different variations of interior SiC test articles: 1) a closed SiC rod for hermeticity and endcap joint strength testing, 2) two open SiC tubes for axial strength testing, 3) five open SiC tubes for hoop strength testing, and 4) GA-EMS produced SiC fiber in a carbon holder. The second capsule will include: 5) two Chemical Vapor Deposition SiC cylinders with representative corrosion coatings (SiC/ Cr) for corrosion testing. The test articles are held securely within the capsules with a spring and are separated from the capsule and other test articles with spacers. Previous work in year one of the program had demonstrated the ability to manufacture SiC test articles to fit within the thimble plug assembly location while maintaining representative geometry, strength, and leak tightness when compared to the slightly larger ATF cladding geometry. These performance results give confidence that PIE on the SiC test articles will provide relevant data for the development and commercial implementation of ATF SiGA® cladding.

To support the irradiation modeling effort, SI has incorporated SiC property models for composite and monolithic SiC into their finite element fuel performance code PEGASUS. The SiC models have been validated through test case comparison and work has begun to develop a 3D model of the irradiation experiment. Accurate modeling of the capsules and SiC test articles will provide data to support licensing of the irradiation, inform follow-on PIE, and provide a modeling framework for facilitation of future commercial irradiations.

The GA-EMS team is continuing to progress the irradiation design and will complete design, manufacturing, and preliminary licensing over the next year.

ADVANCED REACTOR FUELS

- 3.1 Fuel Fabrication and Properties
- 3.2 AR Core Materials
- 3.3 AR Irradiation Testing and PIE Techniques
- 3.4 AR Fuel Safety Testing
- 3.5 AR Performance Assessment

Nuclear Energy University Project (NEUP) Award Chemical Interaction and Compatibility of Uranium Nitride Fuels with Liquid Pb and Alumina-forming Austenitic Alloys

Principal Investigator: Jie Lian, Rensselaer Polytechnic University (RPI)

Team Members/Collaborators: Bruce Pint, Jiheon Jun and Jason Harp, (Oak Ridge National Laboratory; ORNL), Erofili Kardoulaki, (Los Alamos National Laboratory; LANL), Michael R. Ickes (Westinghouse Electric Company)

n this project, the chemical interaction behaviors of uranium nitride (UN) and alumina forming austenitic alloys (AFAs), and UN and liquid lead (Pb) were investigated through the use of diffusion couple and liquid metal immersion experiments, respectively. Experiments were conducted at 550 and 750°C for 500 and 1000 hours in inert environments. Both preoxidized AFA and as-cast AFA were tested. Monolithic uranium nitride and uranium mononitride with <3 wt% of secondary phases including metallic uranium, uranium dioxide, and uranium nitride (U₂N₃) were used in order to study the effects of secondary phases on the chemical interaction behavior among UN-AFA-Pb.

Project Description:

UN is considered as a primary fuel form for lead-cooled fast reactors (LFRs) due to its enhanced fissile element density, higher thermal conductivity, better breeding ratios and neutronic economics as compared with oxide fuels; however, knowledge gap exists in the fuel-cladding interaction with the presence of liquid metal. This project targets the critical issue of chemical interaction and compatibility among UN fuels, AFAs and liquid Pb. The technical objectives of this research project include: (1) to testify the hypothesis of good compatibility of UN with liquid metal Pb and AFAs; (2) to identify any possible interaction and compatibility issues of UN fuels and optimized AFAs with the presence of liquid metals at elevated temperatures; (3) to understand how different fuel chemistry, impurities, phases, and coolant chemistry (specifically oxygen control) impact fuel-cladding interaction and compatibility with liquid Pb coolant/sublayer; and (4) to achieve a comprehensive understanding of the interaction and compatibility of this fuel-coolant-cladding system and recommendation of operation temperature for practical applications. Chemical interaction and compatibility of UN fuels with liquid Pb and AFAs are critical for the long-term performance evaluation of the fuel, coolant, and structural materials for LFRs. The critical obtained on fuel-coolant-cladding chemical interaction and compat-



Figure 1. Scanning electron microscopy (SEM) and energy dispersive spectroscopy (EDS) of a cross section of as-cast AFA with UN+U (2.5 wt%) annealed at 750°C-1000 hr. UN pellets were delaminated from the diffusion couple post annealing. Aluminum Nitride (AIN) formed along the interface of as-cast AFA and UN pellet

ibility will be useful for evaluating and defining key parameters of operation temperature, fuel impurities and coolant chemistry relevant to the deployment of this fuelcoolant-cladding system for LFRs.

Accomplishments:

The technical goals of the research are to characterize the chemical interaction and compatibility among the fuel-cladding-coolant system for the operation of LFRs at 550 and 750°C. Fuel-cladding diffusion couple experiments were conducted between UN, UN + U (2.5 wt%), and UN + UO₂ (<3 wt%) with both preoxidized AFAs and as-cast AFAs for 500 and 1000 hrs. Static liquid metal immersion experiments have been completed between UN, UN+U(2 wt%), UN + U₂N₃ (<3 wt%), and UN + UO₂(<3 wt%) with liquid lead at 550°C and 750°C at 500 and 1000 hrs. UN powders were provided by LANL, and the AFA coupons were provided by ORNL. Fabrication of uranium nitride pellets and experimentation and characterization was performed at RPI.

Across all diffusion couple tests, interdiffusion of major constituent elements can be observed between UN and as-cast AFAs when tested Figure 2. SEM and EDS of a cross section of as-cast AFA with UN+UO₂ annealed at 750°C-1000 hr. No AIN formation is observed, and an alumina-enriched scale formed at the interface with the presence of oxygen from the secondary phase of UO₂ in the fuel matrix



at 550°C for 500 and 1000 hours. At 750°C, a secondary phase of AlN also formed along the interface intermittently for the UN or UN+U pellets. In the inert environment of the diffusion couple experiment, the aluminum can readily form nitride at 750°C. The formation of AlN is also thermodynamically possible at 550°C, but the lower temperature likely limits diffusion of aluminum to the surface and diffusion of nitrogen out of the UN pellet. In addition, no AlN formation was observed in the UN + UO_2 (<3 wt%) and as-cast AFA when tested at 750°C for 1000 hours, and instead an aluminum-enriched surface scale

formed with the presence of oxygen due to the secondary phase oxide in the UN matrix. No chemical interaction and interdiffusion occur upon the interaction of UN with preoxidized AFAs. AFAs are designed with aluminum diffusing to the surface during normal operation, and an oxide scale forms in the preoxidized samples, preventing possible interaction and elemental interdiffusion. These results highlight the important impacts of the protective alumina scale, temperature, and secondary phases on the chemical interaction and interface reaction between UN and AFA.

The chemical compatibility of uranium nitride with aluminum-forming austenitic alloys and liquid lead makes this proposed fuel-cladding-coolant combination a viable candidate for Generation IV lead-cooled fast reactors.

The immerse experiments of UN in liquid metal also indicate a good compatibility between UN and liquid lead. The characterization of the interaction behavior is ongoing. Our initial results show the precipitations of Pb and PbO (<1.5 micron) observed on the surface of the UN pellets for the sample tested at 550°C; however only limited penetration of liquid Pb into the UN matrix occurred through interconnected pores. The most extreme test (750°C-1000 hrs) has yet to be characterized.

In general, UN display a good compatibility with AFA and liquid Pb when tested at 550 and 750°C with limited chemical interaction and liquid Pb penetration, and the fuel-cladding-coolant system could be viable for LFR applications.



Figure 3. SEM of a top surface of $UN+UO_2$ with static Pb for 550°C-500 hr. Spherical precipitations of lead and lead oxide can be observed

Advanced Reactor Core Materials

Principal Investigator: Tarik Saleh

Team Members/Collaborators: Caleb Massey, Stuart Maloy, Mychailo Toloczko, Dalong Zhang, Jens Darsell, Ben Eftink, Ramprashad Prabhakaran, Aditya Shivprasad, Laurent Capolungo, Andre Ruybalid, Hi Tin Vo, David Hoelzer



Figure 1. 14YWT NFA2 production for campaign testing. In (a) the general process for 14YWT fabrication is shown, including the mechanical alloying of precursor powders to their eventual consolidation via extrusion. In (b), comparisons of new NFA2 powder (2022) is compared with historical 14YWT heats, showing improvements in impurity control

dvanced Reactor Cladding Research encompassed two major thrusts in the past fiscal year: International high dose fast reactor irradiations and Data Gaps/ Legacy Sample inventory informing modeling efforts. Through the U.S. Department of Energy - Japanese Atomic Energy Association Civilian Nuclear Working Group (DOE-JAEA CNWG) preparations are being made for inserting HT9 and 14YWT in a JOYO irradiation in CY 2026. Through the French Alternative Energy and Atomic Energy Commission (CEA)-DOE bilateral agreement

work is ongoing to retrieve U.S. and French samples irradiated in the Phenix reactor as part of the MATRIX irradiations. In addition, through a Cooperative Research and Development Agreement with Terrapower, we are working to return the high dose irradiated HT9 specimens to Pacific Northwest National Laboratory (PNNL) after irradiation in BOR-60.

Project Description:

The goal of the Advanced Reactor Cladding area is to integrate our experimental and modeling capabilities to provide a path to qualifying future advance reactor cladding. This is done by high dose testing of irradiated cladding material and identifying legacy material available to test, as well as identifying and fabricating next generation cladding materials, along with supply data and identifying data gaps that help our predictive multiscale modeling efforts. These efforts will all combine to provide data and materials that are available for the next generation of advanced reactors.

Accomplishments:

Level 2 milestone was completed: Current research and technology gaps related to candidate fast reactor cladding materials HT9 and cold worked D9; A summary of HT9 and CWD9 cladding properties, design equations, and data gaps, led by Aditya Shivprasad and Tarik Saleh



at Los Alamos National Laboratory (LANL). This built on a large Level 3 milestone completed earlier this year, Current research and technology gaps related to candidate cladding materials HT9 and D9, led by Ramprashad Prabhakaran and Stuart Maloy at PNNL. These reports set the stage for future experimental and modeling work on these materials and will set up a final report due in FY24.

Additionally significant progress was made on fabricating a new largebatch heat of optimized 14YWT for inclusion in the JOYO irradiation, along with providing stock for future domestic irradiations. Alloy production uses best-practices developed over the past 20 years to reduce impurity contents and provide an alloy with high strength and fracture toughness, and follows the process used in Figure 1. Over the past two decades, improvements in milling atmosphere and consolidation methods have decreased deleterious N and C levels, while maintaining suitably high O to form the complex oxides necessary to give 14YWT its irradiation resistance and high temperature strength. This work was led by Caleb Massey and David Hoelzer at Oak Ridge National

Laboratory (ORNL), with final alloy consolidation and characterization schedule for FY24. In parallel to advancements in conventional fabrication techniques, Principal Investigator (PI) Stuart Maloy and colleagues at PNNL have demonstrated the ability to produce 14YWT, for the first time, using friction-extrusion methods as shown in Figure 2.

A L4 milestone, FFTF-MOTA Ferritic- Martensitic Steel Specimen Inventory at PNNL, was completed by Mychailo Toloczko, PNNL, which provides the available legacy high dose irradiated samples from the Fast Flux Test Facility's Materials Open Test Assembly.

Tarik Saleh, LANL, participated in the DOE-CEA Bilateral agreements, making progress toward finalizing the implementation agreement with CEA to bring the Phenix irradiated MATRIX samples back to the US for testing. Additionally, he participated in the DOE-JAEA CNWG Advanced Fuel Meeting, leading cladding discussions and planning for the JOYO irradiation. Figure 2. Image showing friction extruded rod of 14YWT produced from larger solid rod of 14YWT. Diameter is ~7 mm, length is ~35 mm, work done at PNNL by Dalong Zhang and Jens Darsell under PI Stuart Maloy

> The Advanced Reactor cladding thrust is targeted multi-lab integrated experimental and modeling effort to provide a framework for qualifying advanced reactor materials.

HT9 Modeling

Principal Investigator: Laurent Capolungo Team Members/Collaborators: Andre Ruybalid



Figure 1. Mean square relative error maps for the output response of the accumulated strain ɛss in the transient (top) and the steady-state (bottom) regimes, for the surrogate model with emphasis on the transient regime This task aims at developing models for the mechanical response of HT9 steels as a function of its microstructure.

Project Description:

Materials selection and qualification for nuclear energy applications entails that one can quantify the envelope of performance of components over their lifetime. Given the complexity of harsh environments to be encountered by the material, it is unlikely that decisions can be made as to the viability of specific materials based on experiments/ tests alone. Therefore, models for the mechanical performance of materials are needed. These predictors must (i) be sufficiently general to be applicable for arbitrary loading history, (2) quantify the effects of varying microstructure on the statistics of performance (e.g., yield strength, minimum creep rate, swelling rate etc.)

Empirical models fit to limited experimental data are unlikely to meet these requirements such that mechanistic constitutive laws are preferred. Further, from a pragmatic standpoint, constitutive models must be numerically efficient such as to enable utilization with finite element solvers (e.g., MOOSE). Unfortunately, while the literature offers a few advanced constitutive models for metals -and only one for HT9-, these cannot be seamlessly introduced in MOOSE due to their computational burden. To address this limitation, this project aims to develop and integrate a surrogate constitutive model for the mechanical response of HT9. The general idea is to generate an extensive synthetic database mapping the response of the material to its microstructure fingerprint and then to mine the database to derive surrogate laws. The synthetic database is generated via the use of an advanced constitutive model previously derived by the principal investigator (PI) and collaborators.

The development of these surrogates and successful integration in finite element solvers will enable for a quantification of the entire distributions of likely performance/ response as a function of the fine details of the microstructure thereby facilitating decision making (e.g., material acceptance, design of new experiments).

Accomplishments:

The starting point was a proof-ofconcept creep surrogate model (Los Alamos Reduced Order Model for Advanced Non-linear Constitutive Equations [LAROMance]) for HT9. This model needed significant improvements in terms of stability, range of utilization (i.e., stress, temperatures) and overall accuracy. In FY23, significant changes were made to the surrogate model to address the aforementioned needs. Specifically, a new version of the creep surrogate model (LARO-Mance) for HT9 has been calibrated from an extended synthetic database of the materials response. The database comprises long-term mean-field polycrystal simulations. As an improvement to the first surrogate model, the database has been extended in the stress domain, temperature domain, as well as the temporal domain (longer simulations). A total of ~100 million observations are now included in the calibration process of the surrogate model.

The HT9 surrogate model form has been revised, so that only Legendre polynomials up to the linear degree have been included in the calibration process, and the input/output transformation mappings have been optimized for This project will allow for a rigorous assessment of the performance of components under complex loading scenarios as a function of microstructure of metals. It could pave the way for optimal design of experiments.

each tile separately. Furthermore, a weighted least squares regression method has been implemented to allow for emphasizing certain parts of the model. In this work, the transient and steady-state regimes are autonomously detected by the algorithm, which allows to adjust the weight of either regime during the calibration process. Further, the metrics for model quality in terms have also been revisited and distinguish between the transient and the steady state regime of the simulated creep response. This enables a more thorough analysis of the surrogate model, which is useful for further developments.

These additions to the framework allow for a more tuneable surrogate model to be easily adjusted to the specific problem at hand. Overall, the updated surrogate model is better-behaved, in particular for longer-term simulations, both in terms of continuity/smoothness and accuracy, than its predecessor. The surrogate model shall be made available in the framework of MOOSE as part of the LAROMance family of creep surrogate models. Further improvements to the model are part of the future work and development of LAROMance in general and shall transfer to a possible next version of the HT9 surrogate model. Indeed, some regions of the model seem to overestimate the creep response, which may yield to conservatism in decision making related to safety criteria when this model is used for lifetime predictions of engineering structures.

Nuclear Energy University Project (NEUP) Award

Linear and Nonlinear Guided Ultrasonic Waves to Characterize Cladding of Accident Tolerant Fuel

Principal Investigator: Laurence J. Jacobs, Georgia Institute of Technology

Team Members/Collaborators: Jin-Yeon Kim (Georgia Tech), Jianmin Qu (Stevens Institute of Technology), Remi Dingreville (Sandia National Laboratory), Matthieu Aumand (Framatome), Maximillian Schmitz (Georgia Tech), Charles (Nate) Tenorio (Georgia Tech), Junzhen Wang (Stevens Institute of Technology)



Figure 1. Photo of experimental instrumentation

hile nondestructive evaluation (NDE) techniques are currently available for uncoated, zirconium-based alloys, the use of coated accident tolerant fuel (ATF) fuels adds a new set of challenges for the characterization of ATF. The associated NDE needs are: 1) verify the thickness and uniformity of the coating layers, 2) identify areas of missing coating layers, and 3) confirm the quality of the coating-substrate bond.

Project Description:

The proposed research project has four specific tasks: 1) Develop guided linear ultrasound techniques to verify the thickness and uniformity of cladding layers, 2) Develop guided nonlinear ultrasound techniques to characterize cladding-substrate bond quality 3) Materials modeling to enable reverse-engineering of cladding-substrate bond properties from the measured nonlinear ultrasonic wave results, and 4) Develop methodologies to quantify uncertainties in the proposed NDEbased approaches for quality control and quality assurance (QC/QA). The major deliverable will be the development of a combined linear and nonlinear guided wave ultrasonic approach, this project will address all three needs: (1) verify the thickness and uniformity of the coating layers, (2) identify areas of missing coating layers, and (3) confirm coating-substrate bond quality. Taken together, by addressing these three needs, our project will accelerate the qualification and licensing.

Accomplishments:

The goals of this research are to: 1) develop a combination of linear and nonlinear guided ultrasonic waves techniques applicable for ATF, 2) integrate these ultrasonic wave techniques with newly established material behavior models to develop a reliable NDE approach for ATF, and 3) develop QC/QA procedure for the cladding thickness, uniformity and bond strength. For Objective 1, used a machine and deep learning approach to nondestructively characterize the This research developed an integrated procedure to characterize critical cladding features of ATF using guided ultrasonic waves and machine learning.

thickness and uniformity of a coating in a layered system. A finite element analysis model is first used to computationally model transient, guided Lamb waves propagating in coated specimens with different coating thicknesses. These time-domain signals are then processed with a two-dimensional Fourier transform to obtain the corresponding frequency-wave number relation, which are the dispersion maps of the coated specimen. Dispersion maps are characteristic and depend on both the coating thickness and uniformity, plus its elastic properties (which are taken to be constant). Computationally simulated dispersion maps for a variety of coating properties are obtained and then further processed to extract a feature representation for each dispersion curve. Those extracted features are fed into machine learning classifiers which allow a thickness classification. This machine learning procedure is shown to be effective in classifying the thickness of a uniform coating. However, if the coating thickness is

nonuniform, deep learning, specifically a convolutional neural network architecture, is used for classification. The network is evaluated, tested, and recommendations on its use are given. For Objective 2, dispersion curves of guided waves in a coated plate can be measured by conducting guided wave tests. Since these dispersion curves are influenced by the quality of interfacial bond between the coating and the substrate, analysis of the dispersion curves may reveal the bond quality.

However, solving this inverse problem is extremely difficult. In this paper, we will develop an inverse method based on the use of deep-learning neural networks. By using a set of experimentally measured dispersion curves, this inverse method will yield the quality of the interfacial bond. We assume that the coating-substrate interface is modeled as a linear spring layer of zero thickness. The mechanical behavior of the spring layer is characterized by the spring compliance. Both tangential and normal



Figure 2. Schematic and theoretical dispersion curves

spring compliances are introduced to characterize the bond quality. First, forward computations are conducted for a wide range of the compliance of the spring layer. This will generate a training set of dispersion curves. To account for uncertainty and noise in the experimental data, Gaussian random noise will be introduced in the forward computation of dispersion curves. Second, a convolution neural network (CNN) architecture will be developed and trained using dispersion curves obtained through the forward computations. Once trained, the neural network is ready to solve the inverse problem, i.e., using a set of experimentally obtained dispersion curves to obtain the corresponding compliance of the spring layer. Finally, the performance of the network is assessed based on loss and accuracy.



Figure 3. Analytical dispersion curves for a uniform 200micron coating (red dashed) and a 1 mm plate (green dashed). The curves for the coating and the plate are modeled in a vacuum, respectively



Figure 4. Feature plot of 180 data points with uniform coating for dispersion inversion with fitted function of shape, and defined classifier

Examination of FAST-1 Accelerated Fission Rate Metallic Fuel Test

Principal Investigator: Boone Beausoleil, Luca Capriotti Team Members/Collaborators: S. Patnaik, Todd Jacobsen, Katelyn Baird, Julia Carter

> he Fission Accelerated Steadystate Test (FAST) irradiation test was initiated in 2018 with the first irradiation experiments inserted into the Advanced Test Reactor (ATR) in 2020. The first pins were removed after a single cycle and were awaiting destructive postirradiation examination (PIE) for several years for equipment modifications within the hot-cell. The PIE performed this year includes fission gas release analysis through the updated Gas Assay Sample & Recharge (GASR) system, sectioning of the fuel pins, and metallography analysis with an optical microscope.

Project Description:

The FAST experiments are designed to interrogate critical fuel performance behaviors of advanced reactor fuels in an accelerated manner through geometric scaling of the fuel rods. The first iteration of this test was completed within ATR in 2021 and the pins have begun destructive PIE activities this fiscal year. Modifications were completed to the GASR equipment to support the collection of fission gas from the FAST rodlets. This helps provide a basis for using FAST experiments to design and test advanced nuclear fuels for use in advanced reactors.

Accomplishments:

The Advanced Fuel Campaign (AFC) FAST experiments demonstrate an exceptional opportunity for advanced fuel research and developThis PIE data is the first destructive data showing the results of the FAST irradiation work and captures fuel behavior that contradicts many of the expectations of conventional behavior of metallic fuels.

ment by showing the feasibility of accelerated irradiation testing. The FAST tests differentiate themselves from other accelerated methods in that they use actual in-pile neutron irradiation (vs ion irradiation) and capture integral irradiation effects across prototypic temperature ranges (vs MiniFuel tests, https://doi. org/10.1016/j.jnucmat.2019.151783). These behaviors include fuel constituent redistribution, fuel porosity interconnectivity, bulk fission gas release/retention, fuelcladding chemical interaction, and fuel-cladding mechanical interaction (FCMI). The first FAST pins to undergo PIE includes an annular, 75% smear density (SD), helium bonded pin (FAST-007) and an otherwise equivalent solid, sodium bonded fuel pin (FAST-008) control pin. Both were irradiated to 4.8 % fissions of initial metal atoms, or at% and 3.9% with maximum irradiated cladding temperatures of 470°C

and 400°C, respectively. The results of the GASR collection estimated to be 60-70% gas release. Both pins were sectioned and mounted for microscopy (Figures 1 & 2) to show radial and transverse microstructure. Optical imaging revealed an unexpected closure of the annulus in FAST-007 and the more unexpected redistribution rings within the fuel. These results are extremely notable as conventional wisdom suggests that the redistribution is driven by an $\alpha \rightarrow \beta$ transition. The irradiation centerline temperatures (Figure 3) of both fuels are approximately 550°C and 485°C for FAST-007 and FAST-008, respectively. These temperatures are below the $\alpha \rightarrow \beta$ transition temperature of 660°C and so the redistribution of Zr and U within the fuel pins is unexpected. This work casts further doubt on the accuracy of the phase diagram and the possible misrepresentation of the β phase after previous work within the campaign failed to observe the β phase as the U-Zr samples cooled from $\mathbb{Y} \rightarrow \alpha$. A further observation of the FAST pins was the closure of the annulus of the FAST-007 pin. Previous irradiations under AFC on prototypic geometry pins, albeit with lower smear densities, failed to show a full closure of the inner annulus. The failure for previous AFC designs to close the annulus may be in part due to theorized swelling limitations that prompted the original metal fuel design to 75% SD as the fission gas pores driving the swelling would have reached connectivity and thus the driving force for fuel swelling diminished. Within the FAST-007 pin the annulus represented the idealized 33% swelling fraction and



Figure 1. Transverse metallography of the FAST-007 pin across the center of the pin. Note the coloration difference caused by minor oxidation after sectioning that is indicative of compositional differences (i.e., alloy redistribution). Secondly, tearing within the fuel can be seen along the fuel-cladding interface



Figure 2. Transverse metallography of the FAST-008 pin across the center of the pin. The redistribution within this pin, which is a conventional solid pin, is notably different than that of FAST-007 which is an annular geometry. This pin also shows similar tearing to that of FAST-007 thus the annulus was fully closed. Lastly, the FCMI behavior of the pin was appreciably different to that of historic pins where the profilometry data revealed approximately zero strain across the fuel region. The radiography images provide an indication of the cladding constricting across the fuel region. The optical graphs provide some confirmation of this as the fuel appears to fracture along the fuel-cladding interface region while leaving a thin layer of fuel adhered to the cladding. This suggests that during cooldown the fuel may have thermally contracted, and the tension of the contraction was sufficient enough to pull the cladding inward and eventually fracture the fuel. This will be the subject of modelling exercises in the following year.



Figure 3. A plot of the peak inner cladding and centerline temperatures for FAST-007 and FAST-008. The temperatures are plotted against irradiation time

Microstructure and Micromechanical Characterization of Cr Diffusion Barrier in ATR Irradiated U-10Zr Metallic Fuel

Principal Investigator: Yachun Wang, Luca Capriotti

Team Members/Collaborators: Cameron B. Howard, Fei Xu, Daniele Salvato, Kausttubh K. Bawane, Daniel Murray, David M. Frazer, Scott T. Anderson, Tiankai Yao, Sunghwan Yeo, June-Hyung Kim, Byoung-Oon Lee, Jun Hwan Kim, Randall S. Fielding

ngoing U.S. research and development (R&D) has been devoted to exploring the possibility of ultra-high burnup (30-40 % FIMA) metallic fuels to increase uranium (U) resource utilization in sodium-cooled fast reactors (SFRs). However, the fuel/cladding chemical interaction (FCCI) remains a major challenge limiting the deployment of ultra-high burnup metallic fuels. One possible solution to mitigate/prevent FCCI is to introduce a diffusion barrier. like Cr, between the fuel and cladding. To investigate the irradiation performance of the Cr diffusion barrier, a U-10Zr (wt%) solid fuel sodium bonded with HT-9 cladding with an internal Cr diffusion barrier was irradiated in the Advanced Test Reactor (ATR) to 8.7 %HM burnup at an averaged peak inner cladding temperature of 540-550°C. For the first time, in-depth post-irradiation examination on the Cr diffusion barrier in this fuel was performed at Idaho National Laboratory (INL).

Project Description:

This research aimed to 1) gain insight into the potential microstructural and compositional changes in the Cr diffusion barrier, to 2) characterize potential interactions, and to 3) evaluate the micromechanical properties of the Cr diffusion barrier and nearby The Cr diffusion barrier provided a good integrity to prevent FCCI between the U-10Zr fuel and HT-9 cladding upon irradiation in ATR to 8.7 %HM burnup at an averaged peak inner cladding temperature (PICT) of 540-550°C.

HT-9 cladding through a collection of characterization techniques, including Scanning Electron Microscopy (SEM), in-situ SEM micro-tensile testing, and Transmission Electron Microscopy (TEM), at the Irradiated Materials Characterization Laboratory within the Materials and Fuels Complex at INL. The result of this study provides in-depth insight into the microstructural and compositional stability of the electroplated Cr diffusion barrier during in reactor irradiation. Furthermore, this investigation sheds light on a promising solution to mitigate/prevent the FCCI in HT-9 clad U-10Zr fuels, which can greatly benefit the adoption of metallic fuel for SFRs at high or ultra-high burnup.



Figure 1. A montage backscattered electron (BSE) image shows the entire cross section of the examined fuel sample (a). High magnification BSE images of observed interaction (as red arrows denoted) between the Cr diffusion barrier and fuel periphery (b)-(f). A BSE image shows the preserved Cr diffusion barrier (g). The green and red lines in (g) show the locations for thickness measurements of the preserved Cr diffusion barrier using ImageJ software, with results tabulated in (h)



Figure 2. TEM characterization results of fuel-Cr diffusion barrier interacting interface. A STEM HAADF (a) and a high magnification BF (b) image showing the overall microstructure near the fuel-Cr diffusion barrier interacting interface. Red arrows in (a) highlight the interacted Cr grain boundaries. For the blue box highlighted region in (a), the STEM BF and HAADF and BF images is displayed in (b) and (c), respectively, the elemental mapping of Cr, Zr, U, Nd, and Ce is in (d), (e), (f), (g), and (h), respectively. (i) A TEM image of the nano crystalline Cr/Zr grains for collecting TEM selected area diffraction patterns as shown in (j) and (k)



Figure 3. STEM BF (a) and HAADF (b) images show the overall microstructure of the Cr diffusion barrier after micro-tensile testing. The gauge section shows several Cr crystals, with sizes ranging from hundreds of nanometers to around 2 µm (a). The STEM images (c) – (d) and EDS elemental maps (e)-(f) of a selected area highlighted in green rectangle in (a). The STEM BF image (d) shows a high density of cavities which appear to be closely associated with the Fe element in (e). (g) and (h) TEM selected area diffraction patterns taken from the top shoulder denoted by the as red circle in (a)

2023 AFC ACCOMPLISHMENTS



Figure 4. Engineering stress-strain curves for in-situ SEM micro-tensile tests. The inserted table summarizes the measured yield strength (oy-DIC), ultimate tensile strength (omax), and ductility (stotal-DIC) for each test, together with the average values for the HT-9 specimens

Accomplishments:

SEM characterization on the fuel cross-section (Figure 1a) confirmed the occurrence of interaction between the Cr diffusion barrier and the U-10Zr (wt%) fuel (Figure 1b-1g). At the most interactive location, more than half (~18.4 μ m, #6 in Figure 1g-1h) of the total thickness (~27 μ m) of the Cr barrier has been consumed. Despite the interaction between Cr and Zr (and some U), neither microcracks were detected in the preserved Cr diffusion barrier nor was the HT-9 cladding penetrated by fuel/fission products (Figure 1a). Therefore, the Cr diffusion barrier provided a good barrier integrity against interactions (interdiffusion) between the U-10Zr fuel and HT-9 cladding under the studied irradiation testing condition.

TEM characterization the interacting interface revealed interaction in some Cr grain boundaries (GBs, highlighted by red arrows in Figure 2a). The TEM compositional analysis on a selected region (Figure 2b-2c) found Cr (Figure 2d), Zr (Figure 2e), and some U (Figure2f) on the interacted Cr GBs. This suggests that the Cr grain boundaries have served as favorable pathways diffusing Zr (and some U) into the barrier. TEM characterization found some Nd and Ce-rich particles (Figures 2g-2h) and nano crystalline formation (Figure 2i) in the interaction zone. Indexing the collected diffraction patterns (Figure 2j and 2k which were taken from #1 and #2 spot, respectively, in Figure 2i) confirmed that the nanocrystalline is intermetallic α ZrCr2 Laves phase. The formation of α ZrCr2 is resulted by the reactive diffusion of Zr in the Cr diffusion barrier. driving the growth of interaction layer in the Cr diffusion barrier. The brittle nature of the α ZrCr2 phase explained why microcracks formed in the interacted zone (Figure 1f). Further out-of-pile diffusion couple studies may be necessary to properly evaluate the growth kinetics of the α ZrCr2 interaction layer, which is helpful to better understand the feasibility of the Cr diffusion barrier to prevent/mitigate FCCI.

Figures 3a-3b show the TEM images of the tested Cr diffusion barrier micro specimen. Several Cr grains, with sizes ranging from hundreds of nanometers to around 2μ m, were observed in the gauge area (Figure 2a-2b). Figure 3c-3d are the high magnification TEM images of a selected area in the gauge section (highlighted in green rectangle in

Figure 3). A high density of cavities (white dots in Figure 3d) was observed. The cavities were closely associated with Fe (Figure 3e) which was possibly from the HT-9 cladding by diffusion. It is not clear yet if and how the cavities would affect the barrier integrity if higher burnup was achieved. Indexing the collected diffraction patterns in Figure 3g and 3h confirmed that the Cr grain has a face centered cubic (FCC) crystal structure, while the as-electroplated Cr diffusion barrier has a body centered cubic (BCC) crystal structure. This suggests the occurrence of phase transformation in the Cr diffusion barrier, which may be induced by neutron irradiation.

Comparing to the HT-9 specimens (#2-#6 in Figure 4), the higher yield strength (1231 MPa) of the Cr diffusion barrier (#1 in Figure 4) suggests its higher hardness. Among the tested HT-9 specimens, bar #2 and #3 showed much lower yield strength (451 and 284 MPa, respectively) and ultimate tensile strength (754 and 606 MPa, respectively) than bars #4-6 (799-961MPa yield strength and 984-1296 MPa ultimate tensile strength). This may suggest the occurrence of mechanical softening in the HT-9 nearing the Cr barrier.

Nuclear Energy University Project (NEUP) Award

Femtosecond Laser Ablation Machining & Examination - Center for Active Materials Processing (FLAME-CAMP)

Principal Investigator: Peter Hosemann, University of California-Berkeley Team Members/Collaborators: Chaitanya Peddeti, Sebastian Lam, J. Barefield, J. Gigax, S.A. Maloy, D. Weisz, P. Chou



Figure 1. The LA-ICP-M set up at UCB. The current efforts are directed towards integrating the femtosecond laser tool with the mass spec to fulfil the work scope of the project. (A) Mass spec. (B) Femtosecond laser The project develops laser ablation mass spectroscopy and materials cutting and processing and explores other in-situ diagnostics for radioactive materials without ever touching the highly radioactive material directly.

Project Description:

Post-irradiation examination (PIE) of fuel and cladding materials are the bottle neck in obtaining reliable and meaningful data from irradiated materials, especially fuel rods. Specimen preparation, material property The development of a Laser Ablation Mass Spectroscopy system at UC Berkeley provides the pathway to analysis and process highly dangerous material in a non-contact fashion enhancing worker safety, data output and reducing costs to obtain data on nuclear materials.

evaluation (including thermal properties), isotope distribution, swelling, mechanical properties, and microstructural changes to assess performance in service are difficult, time consuming, and costly tasks. It is the aim of this proposal to develop and demonstrate femtosecond laser ablation-based diagnostic and processing of materials in a rapid throughput and low-cost fashion, while developing new materials sampling capabilities not deployed in the nuclear materials research field presently. This unique system focuses on irradiated fuel rod investigation and, therefore, be designed for hot cell applications and radioactive specimens. The precision and power of modern lasers enables an accurate and high throughput specimen evaluation. The combination of non-contact (reduced contamination), rapid removal rates, and direct measurement of localized elemental composition and thermal



conductivity make this new capability extremely valuable for PIE on irradiated fuel rods. This system will enable a fast and cost-effective fuel rod evaluation, accelerating nuclear fuel cycle research. This approach will enable fast and effective PIE fuel rodlet examination, thereby reducing costs and accelerating research. Furthermore, this work will pave the way to a completely new set of tools deployable for any radioactive sample of interest.

Figure 2. One of the students, (Chai Peddeti), working with the ICP MS installed at UC Berkeley for the FLAME-CAMP program

Figure 4. A 3D printer laser ablation chamber fitted with a glass viewport for radioactive materials processing. Two holes on opposing sides were designed for Ar carrier gas flow to flow through the samber and carry ablation debris to the mass spec. This way, the cleanup process is simplified after laser processing, saving time and cost in performing the isotopic measurements on hot samples



Accomplishments

To develop a laser ablation Inductively Coupled Plasma - Mass Spec (ICP MS) system to process hot samples at University of California Berkeley (UCB), there were several tasks that needed to be addressed. Firstly, we had to design a chamber that could house hot samples to not only shield users from radiation, but also contain all the radioactive debris produced from laser processing with the femtosecond laser. Furthermore, the containment chamber must incorporate a viewing port to allow for the laser to be visible to the sample in the chamber. With these considerations in mind, we designed the containment chamber to hold the rod in place and rotate it to expose different sides of the rod to our femtosecond laser. The viewing port is made of glass and the box is to allow for visibility in the containment chamber during material analysis, and aluminum to support

the various components we plan on using inside of the containment chamber (e.g., micromanipulators, etc.). The containment chamber is incorporated into our femtosecond laser tool and mass spectrometer to perform various analysis on the test rod. Additionally, we are required to clear the administrative hurdle of performing laser processing at the university level. This required rigorous planning and experimental data to show that we can contain laser debris in a closed environment and are maintaining the safety of the user. Thus, we have successfully shown the feasibility of performing laser processing on radioactive samples at UCB by satisfying constraints set by UCB's Environment, Health & Safety Division before we can work with test rods. We created a test experiment to gauge how effective our setup is at containing all active debris formed via laser processing, paving the way for working with higher activity



Figure 3. Plot showing a signal for 234U with optimized LA-ICPMS airflow parameters. 234U has an abundance of 0.005%, so seeing signal means trace analysis is possible with LA-ICPMS. The data shows that around 1 L/min of carrier Ar flow is optimal for high sensitivity signals

material in the future. Thirdly, we are tasked to develop a method to remove laser processed samples for further evaluation. We have designed a lift out device and developed a methodology to perform a lift out procedure to quickly extract laser processed samples. Finally, we must purchase and install an ICP MS system at UCB to fit within the current infrastructure. We were able to obtain a mass spec at UCB for use in the FLAME-CAMP program and collected initial mass spectrometry data from a fuel pellet. The mass spec has been installed and is functional at UCB. We were able to successfully get spectra and isotope data for SS304 with both the

Laser Ablation- Inductively Coupled Plasma - Mass Spec (LA-ICP-MS) set up and the light-induced breakdown spectroscopy setup to compare the data and verify that the components were working correctly as well as detect Kr from the surrogate rod post laser ablation.

Post-Transient Characterization of the Metal Fuel THOR-C-2 Experiment

Principal Investigator: Jason Schulthess, Colby Jensen Team Members/Collaborators: Aaron Craft, William Chuirazzi, Allison Probert

The THOR-C-2 test demonstrated a new capability to test fast reactor fuels to failure and provides an ideal demonstration of advanced nondestructive capability to evaluate fuel behavior for new and valuable data streams.

	THOR-C-2
Composition	U-10Zr
Fuel length (nominal)	34.3 cm
Element length (nominal)	74.7 cm
Fuel Diameter	4.3 mm
Cladding	HT-9
Cladding OD	5.8 mm
Cladding thickness	0.45 mm
Plenum / Fuel vol	1.45
Smear density	75%
Plenum gas	Не
Initial Fill Gas Pressure	0.1 MPa at RT
Heat Sink Material	Ti64

Table 1. Geometry and specifications for Mk-IV fuel pin used in THOR-C-2

The Transient Heatsink Overpower Response (THOR) irradiation capsule was developed to measure time dependent thermal behavior of advanced reactor fuels by subjecting them to accident conditions in Transient Reactor Test Facility (TREAT). The THOR-C-2 test was the first test in this capsule to test a fuel pin to failure and is part of a commissioning series of tests planned. Advanced non-destructive examination was performed on the material after the transient using neutron computed tomography (nCT).

Project Description:

To address needs to continue nuclear heated-safety research and transient testing of modern reactor fuels, the TREAT facility officially restarted in 2017 and planned to revive the planned M-8 experiment to be housed in a newly designed sodium-environment module, the THOR capsule. The THOR capsule was developed to measure time-dependent thermal behavior of sodium fast reactor (SFR) and light water reactor fuel pins subjected to accident conditions within TREAT. The THOR-commissioning tests were devised to establish the performance of the test device and explore fresh metallic fuel performance. These tests will determine the capsule's capacity to measure and detect the behavior of fresh SFR fuel during severe accident conditions, including heat-sink temperature near



Figure 1. Measured reactor power and energy for the THOR-C-2 experiment

the fuel pin and time of clad rupture. THOR-C-2 was irradiated in TREAT in August 2022 and contained an unirradiated experimental breeder reactor (EBR)-II Mk-IV fuel pin that experienced severe transient over power conditions, establishing both a tie-back case to historical testing and power-coupling measurements for calorimetric calibration factors for metallic fuels.

Accomplishments:

The experiment was assembled with an unirradiated EBR-II Mk-IV fuel pin, assembled into the THOR capsule, and irradiated in TREAT. The specifics of the fuel pin are contained in Table 1 with the target transient conditions shown in Figure 1. After irradiation, the experiment was transported to the Neutron Radiography Reactor where both standard radiography and digital nCT were completed. The standard radiography identified two regions of interest in the fuel pin with





Figure 2. 3D nCT projection of the lower region of interest for THOR-C-2 (a), individual nCT slices showing three thermocouple placements within the Ti heatsink at each of the axial positions (b) 4.1 cm, (c) 8.3 cm, and (d) 12.5 cm, and accompanying heat sink temperatures measured from TCs at axial positions (e) 4.1 cm, (f) 8.3 cm, and (g) 12.5 cm

Figure 3. 3D nCT projection of the upper region of interest for THOR-C-2, (a) individual nCT slices showing three thermocouple placements within the Ti heatsink at each of the axial positions (b) 29.7 cm, (c) 25.4 cm, and (d) 23.9 cm, and accompanying heat sink temperatures measured from the TCs at axial positions (e) 29.7 cm, (f) 25.4 cm, and (g) 21.1 cm

one being near the top of the fuel column height, and the other near the bottom. The nCT was completed on both regions of interest (Figure 2 and Figure 3) and the areas were reconstructed in a 3D representation using FIJI image analysis software. From the nCT, features identified include cladding deformation, fuel melting and relocation, cladding breach, and the formation of a central void in the lower region.

Cladding burst is evident in the nCT slices from approximately 29.2 cm above the bottom of the fuel column to the lower boundary of the radiograph at approximately 23.9 cm. Melted fuel was expelled from the pin, and it collected along the side of the cladding (an event known as candling) and in the bottom of the capsule. The candling of the relocated fuel can be seen primarily from 2.2 cm above the bottom of the fuel column until the bound of the lower region of interest (ROI) radiograph at approximately 13.5 cm above the bottom of the fuel column. Fuel also collected at the bottom of the capsule's fuel-pin chamber from about 2.1 to 0.9 cm below the bottom of the fuel column. A redistribution of fuel along one side of the heatsink wall from 0.9 cm below the bottom of the fuel column extending approximately 4.6 cm, axially. The relocated fuel left behind a central void within the fuel pin, evident from approximately 4.9 to 5.9 cm from the bottom of the fuel stack. The temperature measurements from thermocouples located in the heat sink along the length of the fuel pin are also shown in Figures 2 and 3.

Cladding diameter along the radiographed regions was initially measured using the image processor, FIJI. As was seen in results from preand post-transient analysis, cladding strain was greatest along the lower region of the fuel pin. Difficulties in analyzing the nCT radiographs for cladding diameter arose in differentiating the outer cladding surface from the relocated fuel deposited on the clad exterior along a large portion of the fuel pin and establishing an accurate scale, given limited features of known length. Results can be compared with future dimensional analysis from Design Engineers following disassembly of the capsule for accuracy. Planned work includes conducting systematic cladding strain and mass fraction measurements along the entirety of the imaged ROIs using machine learning and segmentation techniques. Quantifying these behaviors is essential to understanding the transient behavior of fuel and compiling an adequate safety envelope for fuel-system qualification.

First Irradiated-Fuel Fast Reactor Experiments in TREAT: Assembly and Performance

Principal Investigator: Klint Anderson

Team Members/Collaborators: Jason Schulthess, Trevor Smuin, Robert Armstrong, Jordan Argyle, Clayton Turner, Philip Petersen, Justin Yarrington, Randall Fielding, Sarah Khan, Todd Birch, Ashley Lambson, Colby Jensen

he Advanced Reactor Experiments for Sodium Fast Reactor Fuels (ARES) project is a collaboration between the U.S. Idaho National Laboratory (INL) and the Japanese Atomic Energy Agency (JAEA) to investigate the transient fuel performance of irradiated advanced metallic and mixed oxide (MOX) fuel designs. As part of the ARES project, a new sodium capsule for use in the Transient Reactor Test Facility (TREAT) has been developed to test fresh and pre-irradiated fuel designs from the Experimental Breeder Reactor (EBR)-II experiment programs. Testing completed in this new sodium capsule has provided fuel performance data of Sodium Fast Reactor (SFR) fuels to support continued improvement in fuel performance and economics.

Project Description:

The Temperature Heat Sink Overpower Response (THOR) capsule is a static sodium capsule designed for irradiation in TREAT. The capsule is sized to house a single EBR-II style fuel pin which is sodium bonded to an internal heat sink. The capsule includes a cable heater which is wrapped around the heat sink and provides pre-transient specimen heating. THOR features an advanced instrumentation package which is customizable for measuring different phenomena of interest depending on the individual experiment objectives. The capsule is heavily instrumented, especially for temperature, and includes numerous thermocouples monitoring the specimen temperature at various axial elevations. Additional instrumentation included in THOR is a linear variable differential transformer to measure either pressure or fuel elongation and an acoustic emission sensor for cladding rupture detection. Figure 1 shows an overview of the THOR capsule.

The THOR capsule was designed to facilitate remote handling and loading inside the Hot Fuel Examination Facility (HFEF) to provide fuel performance data on pre-irradiated metallic and MOX fuel pins. Three fresh fuel commissioning tests were performed leading up to fiscal year 2023. These commissioning tests, along with the design, demonstration, and deployment of remote handling equipment in HFEF, paved the way for the first TREAT fast reactor experiments with pre-irradiated fuel and second overall preirradiated fuel test since TREAT restart.



Figure 1. Overview of the THOR capsule design



Figure 2. Sodium Loading of the THOR-MOXTOP-1 capsule in the pryochemistry glovebox



Accomplishments:

After completion of the fresh fuel commissioning tests and deployment of remote handling equipment in HFEF, THOR capsule components were fabricated and assembled in preparation for the first preirradiated experiments in TREAT. Modules were assembled to support the first THOR-Metallic (THOR-M) and THOR-Mixed Oxide Transient OverPower (MOXTOP) experiments. The THOR experiment module features a hinged design, as seen in Figure 1, which allows the experiment components above the capsule to rotate out of the way to facilitate sodium and fuel loading. This feature allows the THOR capsule to be fully instrumented prior to insertion into HFEF. Assembly of the modules and instrumentation for the pre-irradiated THOR experiments was completed at the Figure 3. Fuel specimen loading of the THOR-MOXTOP-1 capsule



Figure 4. Instrumentation checkout of the THOR-MOXTOP-1 experiment

Measurement Science Laboratory at INL, where the instrumentation package was tested and calibrated.

Following the module and instrumentation assemblies the experiment hardware was shipped to the Materials & Fuels Complex for sodium loading in the pyrochemistry glovebox. In the inert atmosphere of the glovebox, solid sodium was rolled into cylinders and loaded into the capsule. Using the internal cable heater, the heat sink temperature was raised until the sodium melted. The capsule was then allowed to cool, the sodium level in the heat sink was verified, and the capsules were sealed. Figure 2 shows a series of photographs of the sodium loading of the THOR-MOXTOP-1 capsule.
The completion of the first fast reactor experiment in TREAT with previously irradiated fuel established unique capability and matured a diverse transient testing team to complete the JAEA-INL ARES project on high burnup fast reactor fuels.

The experiment modules were then transferred to HFEF for fuel loading and sodium wetting. Once inside HFEF, the hinged portion of the experiment module was rotated, the capsule top was opened, and the pre-irradiated fuel pin was inserted into the capsule and internal heat sink until fully seated. After fuel pin loading, the sodium was bonded to the specimen cladding, the capsule was closed, and instrumentation was checked prior to shipment to TREAT for irradiation. Figure 3 shows a photograph of the preirradiated fuel pin loading into the THOR-MOXTOP-1 capsule.

The THOR-M experiment was the first THOR capsule to go through fuel loading and assembly inside HFEF, with the intent of providing assembly experience to the THOR project team. The THOR-M experiment was successfully loaded and assembled in March 2023 and further proved the design of the remote handling equipment developed for assembly. The THOR-M experiment provided expertise to HFEF operations and the THOR project team which can be leveraged for future THOR experiments with pre-irradiated fuel specimens.

Following assembly of THOR-M, the THOR-MOXTOP-1 capsule was successfully loaded in HFEF and delivered to TREAT for irradiation. The THOR-MOXTOP-1 transient occurred in TREAT in July 2023, successfully completing the first irradiation of previously irradiated fuel pins in the THOR capsule. Figure 4 shows a photograph of the instrumentation checkout which occurred on the THOR-MOXTOP-1 experiment prior to shipment to TREAT. The completion of the first previously irradiated fast reactor experiment in TREAT paves the way for additional THOR experiments with pre-irradiated SFR fuels and has significantly advanced preparations for the following water capsule and sodium loop experiments with high burnup fuels.

Nuclear Energy University Project (NEUP) Award

NEUP 20-19374: Maintaining and Building upon the Halden Legacy of In-situ Diagnostics

Principal Investigator: Michael Corradini, University of Wisconsin-Madison (UW) Co-PIs: M. Anderson (UW), W. Marcum (Oregon State University, OSU) Team Members/Collaborators: H.J. Jo (Pohang Institute of Science & Technology, Pohang, South Korea (POSTECH) Senior Personnel: T. Moreira (UW), T. Howard (OSU) Graduate Students: A.B. Luna (UW), M. Moussaoui (OSU)

> iven the expanding mission of test reactors for accident tolerant fuel and advanced fuels, it is crucial to maintain and improve upon Halden Reactor's legacy of in-situ instrumentation and sensor capabilities. In particular, Halden has been a significant source of experimental data to support the nuclear industry's better understanding of fuel-cladding response during light water reactor (LWR) transients and accidents: e.g., Lossof-Coolant-Accident (LOCA). Specifically, this type of LOCA testing is important given the interest in advanced LWR fuels that seek to extend burnup beyond current regulatory limits: i.e., using doped fuel or coated claddings that can minimize operational corrosion and oxidation under accident conditions.

Project Description:

This project seeks to provide realtime in-situ measurements during testing. Our focus is on developing in-situ measurements of distributed temperature measurements and local strain measurements for a simulated fuel rod (electrically heated test rod) in out-of-pile test loops at UW (steady-state) and OSU (LOCA transient). This project objectives are listed below with the current accomplishments provided.

Accomplishments:

Objective 1: Review the current fiber optic sensor designs that would be suitable for the nuclear environment (i.e., appropriate radiation, pressure, and temperature conditions) and identify and acquire the fiber optic sensors to be installed in heated test rods.

Accomplishment

Our team led by UW investigators (A.B Luna, M. Anderson) reviewed a number of past research efforts on fiber-optic sensor (FOS) development and submitted a Milestone Report in 2021. In particular, the 'fiber embedment method' was found to be most appropriate. Researchers measured temperature profiles by embedding the FOS sensors in grooved channels of stainless pipe heated specimens. An ultrasonic additive manufacturing process was used to embed the FOS along the heater surface. For this work, distributed temperatures were measured along the rod surface using these embedded optical fibers. The UW approach was to incorporate this approach in the heated rod design.

Objective 2: Develop the heated test rod design that can accommodate the fiber optic sensors for distributed temperature measurements and local strain measurements to be used in the transient LOCA testing. Qualify this fabricated heated test rod for use in transient testing.

Accomplishment

The UW team (A.B. Luna, T. Moriera, M. Anderson) in collaboration with the OSU team (T. Howard, W. Marcum) designed two electrically heated rods to be used in the LOCA experiments in the OSU LOCA facility. These heater rods were fabricated by Stern Labs based on our design specifications. Delays in fabrication were encountered due to supply chain issues during the COVID pandemic. The heater rods have identical geometry, six internal thermocouples along the rod length, along with three stainless steel capillary tubes for insertion of FOS within the rod and identical power and heat flux profiles. The key difference is one of the heater rods will have helical grooving on the exterior heater cladding wall to accommodate for incorporation of FOS pairs along the surface. The temperature FOS is bare and sheathed in a steel capillary attached to the rod to allow strain-free measurement of the distributed surface temperature. The FOS for strain measurement incorporates a FOS fixed in the groove of the rod surface. By cross-comparison of these sensors, the mechanical strain can be measured. This technique was proven in UW steady-state tests for a heated rod under various pressures, temperatures and temperature gradients. These instrumented heater rods are to be shipped to OSU this quarter.



Objective 3: Design and modify the current OSU transient LOCA test facility to allow for transient LOCA testing. The modified facility will be qualified by performing scoping LOCA tests and then installing the instrumented heated test rod into the OSU facility.

Accomplishment

The OSU team (M. Moussaoui, T. Howard, W. Marcum) has made major improvements in the design and operation of the OSU LOCA test facility. The facility supports testing of the heater rod with in-situ diagnostics using FOS for distributed temperature and strain measurements. Improvements include [1] a redesigned and fabricated heat exchanger for heat removal from the heater rod test rig, which allows for greater heat transfer and larger natural circulation flow in the Figure 1. Dimensions are in mm unless noted otherwise. (a) Schematic of a 10.25 mm diameter and 254 mm long simulated nuclear fuel pin with 6 helical V-grooves machined on the surface of the fuel cladding, 120 degrees apart, 0.30 mm deep. (a & c) Strain and temperature fiber optical sensors were embedded side-by-side in the helical V-grooves and will be used to monitor surface temperature, cladding elongation, and bending. The temperature fiber data will be used as reference to decouple the strain and temperature data of the strain fiber. (b) Cross-Sectional view showing 6 locations for type-K thermocouples and 3 stainlesssteel capillaries for temperature fibers inside the cladding wall



Figure 2. (Left-to-Right) Dr. Tiago Moreira, Dr. Michael Corradini, Alastair Luna, Dr. Mark Anderson pictured holding a prototypic simulated fuel rod with temperature and strain fiber optical sensors embedded in the cladding for in-situ diagnostics

Figure 3. (Left to right) Dr. Trevor Howard, Musa Moussaoui pictured in front of OSU O-SERTTA facility

test capsule; [2] a LOCA ball valve assembly that allows for reproducible initiation of the LOCA transient; [3] replacement gland fittings and capillaries to accommodate heater rod with FOS instrumentation. The improvements have allowed for a series of LOCA tests to be performed at prototypical pressurized water reactor conditions using a heater rod with standard instrumentation. These have successfully exercised the data acquisition system and associated instrumentation as the OSU team awaits delivery of the FOS instrumented rod this quarter.

Objective 4: Perform the LOCA transient tests in OSU LOCA facility with fiber optic sensors installed.

Objective 5: Analyze LOCA test results using state-of-the-art safety analysis codes: e.g., RELAP5.

Accomplishments

Transient LOCA tests are to be performed this quarter after the installation of the heater rod with FOSs. In anticipation of this, a RELAP5 model of the OSU facility was developed, and simulations have been run and compared to existing transient test data and these have shown good agreement.

Summary

This research project has developed advanced In-Situ Fiber Optic Instrumentation to measure Distributed Temperatures and Mechanical Strain for prototypic LWR heated simulant fuel rods for LOCA Transients. These diagnostic techniques can be transferrable to advanced fuel being tested at Department of Energy laboratories.

References

- T.K. Howard, M. Moussaoui, J. Miller, A. Weiss, W. Marcum, "Overivew of the Out-of-Pile Transient, Blowdown, Reflood, and CHF Experiment at OSU", Proceedings of NURETH-19, March 2022
- [2.] A.B. Luna, M. Anderson, M.Corradini, "Deconvolving of Strain from Fiber Optic Sensor Measurements at Elevated Temperatures", Trans. of ANS Winter Meeting, Phoenix AZ, Nov. 2022
- [3.] M. Moussaoui, J.Miller, T.K. Howard, W. Marcum, "Out-of-Pile CHF LOCA Experiment", Transactions of the ANS Winter Meeting, Phoenix AZ, Nov. 2022
- [4.] J.Miller, T.K. Howard, W. Marcum, "Thermal Hydraulic Analysis of Annular Natural Circulation", Transactions of ANS Winter Meeting, Phoenix AZ, Nov. 2022
- [5.] M. Moussaoui, J.Miller, T.K. Howard, W. Marcum, "Outof-Pile Transient Blowdown Experiment", Proceedings of NURETH-20, August 2023





LUNA

Actual: 1:39 Displayed: 1:8



Pre-Test Fuel Performance Evaluation of THOR-C Experiments

Principal Investigator: Matt Mihelish Team Members/Collaborators: Colby Jensen, Pavel Medvedev

> he Advanced Reactor Experiments for Sodium Fast Reactor fuels (ARES) project is a joint project between U.S. Department of Energy (DOE) and the Japan Atomic Energy Agency (JAEA). The experimental program consists of several tests designed to evaluate transient performance of mixed-oxide (MOX) and metallic, sodium fast reactor (SFR) fuels. Detailed studies of transient fuel performance have been the subject of research in several historical research programs, but little testing has been done on SFR fuels since the shutdown of the Transient Reactor Test facility and the CABRI sodium loop in the early 1990's and early 2000's, respectively. The ARES test program includes three series of tests to be performed including a set of fresh fuel metallic fuel specimens called the Temperature Heatsink Overpower Response (THOR) Commissioning or C-series, the THOR-M-TOP/M-LOF experiments for irradiated metallic fuel specimens, and the THOR-MOXTOP experiments on irradiated MOX pins. DOE and JAEA are the primary principal investigators for the metallic fuel and MOX fuel scopes, respectively. All irradiated pins are from historical experiments in the

Experimental Breeder Reactor-II reactor. The C series is the primary topic of this document as the first set of experiments to be tested in the overall project.

The objective of this report is to provide a predictive fuel performance assessment of the THOR-C experiments using the BISON fuel performance code. The BISON temperature results will be compared to ABAQUS results of the THOR-C experiments. The calculations provided in this document represent coupled thermomechanical evaluation of the THOR-C experiments.

Project Description:

The BISON fuel performance code is the primary tool used for modeling the performance of the THOR-C experiments. Previously, ABAQUS was used to predict the thermal behavior of the experiments providing only temperature results. Those results are used for comparison to the BISON results to as model benchmarks for the thermal predictions. The BISON models and geometry used for the analysis of the THOR-C experiments were adapted from earlier versions to better match the final



Figure 1. BISON predictions of the thermomechanical performance for THOR-C-3-B versus THOR-C-4 experiments, which is a comparison between annular and solid fuel

2023 AFC ACCOMPLISHMENTS

Initial calculation results provide valuable insight to the performance of the experiments, the BISON approach to modeling THOR experiments, and a start to assessment of code capabilities for transient metallic fuel analysis.

test configurations. There are two models with similar input files and meshes for the solid and annular fuel. The models focus specifically on the fuel, cladding, and heat sink, while the surrounding components are excluded. Components not included in the analysis are the THOR capsule, outer containment, insulation, and outer can, which will have little impact on the thermal mechanical results of the fuel and cladding during the duration of the transient experiments.

Accomplishments:

The THOR-C experiments have been analyzed using the ABAQUS code and the BISON fuel performance code. Initial calculation results provide valuable insight to the performance of the experiments, the BISON approach to modeling THOR experiments, and a start to assessment of code capabilities for transient metallic fuel analysis. The results show some good agreement between the two codes for predicting specimen temperatures, however, some notable discrepancies exist especially during the high temperature phase of the transient that will be further investigated. Cumulative damage fraction calculations show failure of several experiments including THOR-C-2,

C-3A, C-3B, C-5. However, based on historical experiments peak cladding temperatures of > 1078°C have a high likelihood of failure due to eutectic penetration regardless of cladding creep damage. THOR-C-2, C-3B, C-4, and C-5 all cross this threshold. Comparison of annular and solid fuel results indicate increased fuel-cladding interaction though further analysis is required to draw clearer prediction of failure probabilities. None of the modeling to date includes the calculation and effects of fuel-cladding eutectic interaction, which will likely play a significant role in cladding mechanical behaviors and accelerate ultimate failure. A comparison between annular and solid fuel is shown in Figure 1. The two experiments have different cladding materials, and the annular fuel has approximately 7% higher input power. The annular fuel produces the largest fuel growth and cladding hoop strain. The cladding hoop stress increased significantly as compared to the solid fuel after 5 seconds into the transient. The cladding creep strain is more than a magnitude higher for the annular fuel.



FAST to EBR-II Comparison and HT9 Applicability

Principal Investigator: Boone Beausoleil Team Members/Collaborators: A. Swearingen, K. Paaren

This work is important as it establishes the divergence of FAST tests from conventional testing but validates their applicability by clarifying the critical life limiting phenomena of metallic fuels.

he Fission Accelerated Steady-state Test (FAST) tests are a key component of the Advanced Fuel Campaign (AFC) irradiation testing portfolio. The FAST methodology enables rapid integral testing of advanced fuel concepts partnered with control pins for simple yet effective comparisons between fuel-cladding systems. However, comparisons between performance necessitate parametric modeling assessments to understand the gaps within the general understanding of metallic fuel phenomena.

Project Description:

This project was executed to perform an assessment of the applicability of a FAST irradiation experiment to understanding HT9 cladding behavior when compared to the performance of HT9 cladding in Experimental Breeder Reactor (EBR)-II experiments. Due to the accelerated nature of a FAST experiment, the cladding is subjected to neutron fluences approximately 1/4 that of a conventional irradiation test. Additionally, the long duration of the conventional experiments normally lends themselves to meaningful thermal creep behavior but in a FAST test the stress imparted on the fuel is done so more rapidly while the duration is also reduced. These incongruencies between conventional testing in historical tests within EBR-II and the FAST tests require additional analysis.

This work therefor attempts to interpret the differences between the two irradiation conditions. First, as-run comparisons are made to show thermal and irradiation deformation of the cladding in both testing conditions. Second, EBR-II fuel fluxes were applied to the HT9 cladding to determine the divergence between the FAST and EBR-II irradiation performance.

Accomplishments:

The focus of this work are the control pins of the FAST experiment, which are all solid, sodium bonded, 75% smear density U-10Zr fuel in HT9 cladding. These pins are important as they allow a comparison to be made between historical metal fuel tests (primarily in EBR-II) and the novel designs within the FAST test matrix. The FAST pin FAST-008 was irradiated within the Advanced Test Reactor (ATR) to 3.9 %FIMA (fissions of initial metal atoms, or atomic %) and is currently undergoing post irradiation examination (PIE). Other pins, listed in Table 1, are continuing on-going irradiations. All of these pins have been modeled, along with the entirety of the AFC FAST tests, within the Multi Object-Oriented Simulation Environment (MOOSE) using the BISON fuel performance code. The configuration of the tests and model relationship is shown in Figure 1. The rodlets were

modeled using as-run neutronics data and projected powers for cycles yet to be irradiated with control pins (75% smear density, sodium bonded, solid pins in HT9). The modeling was done to make comparisons between thermal and irradiation effects on the cladding and how these two phenomena independently affect the performance of the FAST fuel system experiments. To this end, EBR-II irradiation data was taken from the Fuels Irradiation & Physics Database to match the conditions of the FAST experiments. The neutronics data was then taken and applied to the FAST experiment models so that the burnup behavior of the fuel would match the irradiation conditions of the EBR-II experiments. This comparison is shown in Figure 2. This exercise also revealed deficiencies in the Bison performance models. There is an issue with the fuel creep calculation where the solid geometry pins are diverging from a solution and not completing the simulations. This leads to the solid fuel pins needing to be simulated while excluding the fuel creep from the inelastic stress calculations. The annular pins can be simulated while including the fuel creep calculation. The cause of this issue with the fuel creep for solid pin geometries is still under investigation. In spite of these deficiencies, the models were still informative. Preliminary results show that the burnup is consistently underpredicted when compared to the Monte Carlo N-Particle as run simulations. The strain is overpredicted in the region where the fuel is located

Experiment	Burnup (%FIMA)	Cladding Fluence $\left(\frac{n}{cm^2}\right)$	PICT (°C)	Cladding DPA
FAST-008	3.9%	4.44×10^{20}	410	1.61
FAST-016	8.5%	7.98×10^{20}	470	2.89
FAST-031	9.54%	7.36×10^{20}	510	2.66
FAST-048*	14.6%	1.14×10^{21}	543	4.16
FAST-052*	13.2%	1.20×10^{21}	475	4.33
FAST-050*	18.9%	1.49×10^{21}	500	5.40
FAST-053*	17.8%	1.60×10^{21}	476	5.78
X425A-T423 (142B-0.15)	3.9%	1.77×10^{23}	411	16.43
X425A-T423 (146A-0.583)	8.03%	4.53×10^{23}	468	39.82
X425A-T423 (146B-0.55)	9.55%	5.51×10^{23}	512	47.96
X425A-T424 (144A-0.117)	3.83%	6.84×10^{22}	435	17.3
X425A-T424 (150A-0.717)	8.57%	4.55×10^{23}	478.6	42.2
X425B-T424 (149A-0.517)	9.48%	4.24×10^{23}	504	47.85
X425C-T424 (158A-0.783)	14.6%	1.78×10^{24}	526	73.52
X425B-T424 (153A-0.417)	13.78%	1.25×10^{24}	477	71.65
X425C-T424 (158A-0.517)	17%	2.17×10^{24}	489	90.49
X425C-T424 (158A-0.517)	17%	2.17×10^{24}	489	90.49

*Projected from previous ATR cycles.

Table 1. This table provides a brief summary of FAST and EBR-II pins with burnup, cladding fluence, and cladding dpa. For the EBR-II pins, the databases provide a summary fluence for the assembly but not specific fluence or dose for individual pins. EBR-II pins are listed by sub-assembly and pin number with the parenthesis capturing the cycle and location (Z/I) for the comparative local conditions

Figure 1. (Left) Simulated FAST Test Geometry (width x10 magnification). Note that each simulation was performed with individual rodlets and not in the double configuration. This was done to simplify computational time as there was negligible neutronic and thermal communication between rodlets. (Right) The BISON material model intercoupling flow chart for the fuel



Figure 2. A comparison plot between expected cladding strain in the FAST-008 test with the as-run ATR flux (predicted curves) and the scaled flux to match the equivalent EBR-II test (scaled curves). The temperature curves for both conditions are shown to demonstrate equivalent thermal conditions and the lack of impact that cladding deformation has on the thermal behaviors of the experiment

> and underpredicted in the plenum region. The plenum pressure is also consistently underpredicted when compared to the PIE results. These comparisons highlight gaps within the models and the discrepancy between irradiation

behavior in prototypic fast reactor tests on HT9 to that of the FAST tests. This helps provide a basis for using FAST experiments to design and test advanced nuclear fuels for use in advanced reactors.

CAPABILITY DEVELOPMENT

- 4.1 ATR Loop Installation
- 4.2 TREAT LOCA Testing
- 4.3 Refabrication and Instrumentation
- 4.4 Fast Neutron Irradiation

Expanding ATR Loop Testing Capability

Principal Investigator: Nate Oldham

Team Members/Collaborators: Kendell Horman, Vince Tonc, John Naughton, Madison Tippet, Kelly Ellis, Carlos Estrada-Perez, Brian Durtschi, Cody Race, Thomas Berti, R Dale Kepler, and Darrin Steffler

The I-Loop will expand fuel testing capacity, address LWR fuel irradiation testing capability gaps left by the HBWR closure and enhance testing capabilities including ramp testing. E xpand access and capability to the Advanced Test Reactor (ATR) Medium-I position with a flowing water loop, allowing testing of fuels and materials in prototypic Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) conditions.

Project Description:

The closure of the Halden Boiling Water Reactor (HBWR) and growing demand for extending light water reactor (LWR) fuel performance limits, has created urgency to find a testing solution to support near-term fuel testing needs. This need comes at a time when industry, via Electric Power Research Institute and the Department of Energy (DOE), via the Accident Tolerant Fuels Program, are pushing for performance and capability increases for new fuels. The I-Loop project aims to fill the gap created by HBWR's closure and provide a similar capability at ATR, by installing a flowing water loop in one Medium-I position and providing access to the other Medium-I positions for future experiments. To provide better access to the I-positions of ATR via the new reactor top head, a new Transfer Shield Plate - Mark II has been designed to allow expanded

access of the ATR for continued fuel and materials testing. To support the flowing water loop, the refurbishment of the 1A cubicle will be required including new pumps, piping, heat exchangers, and other equipment. Two separate I-Loops are planned to support a prototypic PWR and a BWR environment. The improvements will enable advanced fuel qualification to continue at the Idaho National Laboratory (INL) and support the industry and DOE desire for these capabilities in the United States, to support the continued safe use of existing LWR reactors and provide for future fuel testing for next generation reactors.

Accomplishments:

The I-Loop team has been busy in fiscal year (FY)2023 with the engineering design by completing the conceptual (30%) design on all tasks and some well into preliminary (60%) design. The objective is to establish a base design to the point that all major components are specified. This allowed the project to place the long lead procurement ahead of when the component is needed.

The I-Loop project is large in scope which requires many sub-teams working on discrete tasks. The main project areas are as follows: (1) I-Loop Tube, (2) I-Loop X-Core System, and (3) Transfer Shield Plate – Mark II.

1. The I-Loop Tube serves as the primary component within the ATR vessel (Figure 1). Its purpose is to contain the experiment in a separate environment at prototypic LWR fluid conditions. This includes the ability to perform two-phase flow similar to a BWR. In FY2023, the I-Loop Tube completed the preliminary design and moved to the final design phase. A finite element analysis was performed to identify the high stress areas. Those areas were modified to add additional thickness and support through iterative changes. The design now passes an American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III design criteria.

> The I-Loop Tube is a first-ofkind design with complex features that are challenging to fabricate. To mitigate this risk, extensive prototyping is needed. The team has been in contact with vendors/fabricators to determine optional features and



Figure 1. 3D model of I-Loop tube with quick-connect flanges



Figure 2. I-Loop test train next to an I-Loop tube



I-Loop Test Train

Hardware to Seal to new Top Head

Upper ILT (with offset)

welding techniques to ensure a successful fabrication. Additionally, a stainless-steel prototype (Figure 2) was fabricated and installed to INL's flowing autoclave loop for laboratory testing in FY2023 with great success. The stainless-steel prototype demonstrated that an I-Loop Test Train could be successfully inserted/removed I-Loop Tube and maintain mechanical integrity of the components prototype (Figure 3). Hydrodynamic testing of the stainlesssteel prototype also allowed the designers to gain valuable measurements to inform the overall loop design.

2. The I-Loop X-Core System is all the equipment outside of the reactor vessel, e.g., pumps, heat exchangers, valves, and piping (Figure 4). This system includes a series of sub-systems such as loop makeup, ion exchange/ filtrations, sampling, and loop chemistry. A major accomplishment in FY2023 has been the completion of the conceptual design of a purification system. The purification system is designed to clean-up a fission product release from testing specimens in off-normal conditions. This represents a major capability of the I-Loop system.

Figure 3. I-Loop test train installed in an I-Loop tube



Figure 4.I-Loop X-Core System layout

3. In FY 2021, the I-Loop project completed a modification to the ATR reactor vessel pressure boundary with a new Top Head Closure Plate. The main design benefit is the addition of eight new penetrations that can be utilized for irradiation testing. Access to these new penetrations requires modification to the shielding known as the Transfer Shield Plate. A final design is completed and a request for proposals has been sent to potential fabricators for the Transfer Shield Plate – Mark II (Figure 5).

2023 AFC ACCOMPLISHMENTS



Figure 5. Transfer shield plate – Mark II final design

TREAT LOCA Testing and Large Experiment Capability

Principal Investigators: Colby Jensen

Team Members/Collaborators: Todd Pavey, Cindy Fife, Klint Anderson, Nicolas Woolstenhulme, and Colby Jensen

Idaho National Laboratory establishes capability to conduct LOCA testing in TREAT recovering the gap lost by shutdown of the Halden Boiling Water Reactor in Norway.

he Halden Boiling Water Reactor (HBWR) in Norway was a key resource for assessing nuclear fuels and materials behavior to address performance issues and answer regulatory questions. When HBWR was shutdown in 2018 the capability to conduct Loss of Coolant Accidents (LOCA) in a test reactor was lost. The Advanced Fuels Campaign has developed the Transient Water Irradiation System in TREAT (TWIST) experiment capsule to restore the LOCA testing for the industry. To commission the TWIST capsule in the Transient Reactor Test Facility (TREAT), the reactor was modified for large experiment capability to allow insertion of the Big-Broad Use Specimen Transient Experiment Rig (Big-BUSTER) in the center of the reactor core. While TWIST commissioning will be the first use of Big-BUSTER, it is planned for use by all experiments for the foreseeable future, with backward compatibility with existing capsules, making a notable shift in experimental approaches in TREAT to loop-scale testing.

Project Description:

The TWIST capsule provides a static water environment surrounding the fuel rod with the ability to rapidly drain the water from the capsule into a flask via an electronic blowdown valve during the transient to simulate a LOCA on a fuel rod. A modified TWIST capsule has also been developed to support Reactivity Initiated Accident (RIA) testing of longer length fuel rods than currently possible in the Static Environment Rodlet Transient Test Apparatus (SERTTA) capsule. The TWIST capsule is equipped with state-of-the-art instrumentation to collect relevant data for posttest analysis. The TWIST capsule completed the first commissioning test at the end of FY23 with three additional experiments to follow in FY24. This is a critical step towards conducting LOCA experiments to investigate fuel fragmentation, relocation, and dispersal for high burnup light water reactor fuels.

To support TWIST capsule and future experiments a large experiment capability was established in TREAT after several years of planning and design. The large experiment capability included a new retro fit shield plug, new moderator assemblies to convert the core from four-inch by eightinch experiment opening to a ten-inch diameter experiment opening, replacing BUSTER with Big-BUSTER and a newly design and fabricated neutronic equivalent device (NED). The larger experiments allow for more room in the experiment capsules to enhance the experiments with new state-ofthe-art instrumentation and allow

for longer and multiple fuel pins. This large experiment capability is an important progression in the transient testing experiment program that is necessary for the forthcoming NASA hydrogen loop and NatriumTM sodium loop testing. The capability is designed with backward compatibility for existing (small) BUSTER test devices such as SERTTA and Temperature Heat Sink Overpower Response.

The TWIST capsule is also designed with hot cell integration in mind and design efforts are already underway to support testing previously irradiated fuels, with preliminary design for High-burnup Experiments in Reactivity Initiated Accidents (HERA)-High Burnup (HBu) and LOCA-HBu experiments. Pre-irradiated experiments are assembled in the Hot Fuel Examination Facility (HFEF). Additionally, design efforts continue towards developing remote handling capability in HFEF for the TWIST capsule to allow for assembly of pre-irradiated experiments.

Accomplishments:

The TWIST capsule final design was completed in February 2023. North Holmes Laboratory completed fabrication of the first TWIST capsule components meeting a level three milestone in May 2023. The TWIST capsule assembly completed all non-radioactive material assembly



Figure 1. First assembly of the TWIST test device for LOC-C-1 tested in Sept. 2023

2023 AFC ACCOMPLISHMENTS



Figure 2. Large experiment capability developed at TREAT in September 2023

2023 AFC ACCOMPLISHMENTS

and instrumentation installation at the Materials Science Laboratory. During this time, the fresh fuel test pin was fabricated and assembled by the Materials and Fuels Complex (MFC) fuel fabrication assembly team at the Advanced Fuel Fabrication (AFF) facility with a centerline thermocouple. The fuel fabrication team completed the capsule subassembly at AFF by welding the thermocouple into the fuel pin, fuel pin installation into the capsule, and closing the capsule completing a level two milestone ahead of schedule in August 2023 (Figure 1). The TWIST capsule was sent to TREAT operations team where they added water and leak checked the capsule prior to placing into the TREAT core. TREAT inserted the LOC-C-1 into the TREAT core in Big-BUSTER and completed the power calibration transient. The data from this experiment will be analyzed to validate predicated heating rates for the specimen in the TWIST device and ensure future experiment power targets are hit.

Large experiment capability was put into place by completing several plant modifications. The new retro fit shield plug was designed by the TREAT design team and fabricated at the MFC machine shop. The retro fit shield plug was replaced by TREAT operations personnel completing the first step towards the large experiment capability. The Moderator assemblies were designed by the TREAT design team and fabricated at the Advanced Test Reactor (ATR) machine shop and the MFC machine shop. To install the four moderator segments nine fuel elements from the center of the TREAT reactor core were removed

then replaced with the moderator assemblies. The moderator assemblies formed the ten-inch opening needed to support the new Big-BUSTER. Big-BUSTER and Big-BUSTER NED was designed by the NF&M design engineering team. The Big-BUSTER was fabricated at Nu-Tech Precision Metals in Canada. The Big-BUSTER NED was fabricated at the ATR machine shop. Both the **Big-BUSTER and Big-BUSTER NED** have been used in TREAT to support the first TWIST commissioning test, completing the large experiment capability that will enable loop-scale testing in TREAT (Figure 2).

Preparations for testing high burnup fuel from the Byron Nuclear Generating Station are also underway with completion of preliminary design of TWIST experiments, HERA-HBu and LOCA-HBu. This design will be the workhorse for the previously irradiated RIA and LOCA testing in the future.

Refabrication/Reinstrumentation Capability Development

Principal Investigator: Jason Schuthess

Team Members/Collaborators: Spencer Parker, Justin Yarrington, Clayton Turner, William Chuirazzi, Colby Jensen, Mark Cole

he Halden Reactor Project (HRP) was, until the recent closure, a key resource for assessing nuclear fuel and materials behaviors to address issues and answer regulatory questions supporting the light-water reactor community. HRP included significant experimental capabilities and knowledgeable staff developed over decades to perform this challenging work. Refabrication and reinstrumentation are considered enabling capabilities that allow access to fuel materials at any point in their respective lifecycles. They are also critical in supporting the deployment and qualification of accident tolerant fuel materials. As many candidate materials are already undergoing irradiation as lead test rods in commercial nuclear power plants, irradiation in the Advanced Test Reactor at Idaho National Laboratory (INL) alone cannot produce the quantities of materials necessary to support qualification. Refabrication and reinstrumentation allow for the use of the lead test rod materials that have been irradiated under prototypic commercial conditions. These conditions include localized phenomena like fretting, grid spacer effects, and

corrosion. Specific segments of these materials can then be selected for subsequent testing under steadystate, transient, or ramp conditions. Refabrication also allows instrumentation (e.g., thermocouples) to be added to these segments to measure parameters, such as temperature and fission gas release.

Project Description

The overall objective of this program is to develop the capabilities necessary to refabricate and reinstrument previously irradiated materials. Figure 1 illustrates the distinction between classic refabrication activities and advanced reinstrumentation. With closure of the HRP, these capabilities must be replaced and/or improved upon where feasible to support future nuclear fuel qualification needs. Specifically, some of the objectives and capabilities include:

- Development of dry drilling techniques in ceramic UO₂ fuel pellets
- Development of fuel rod end caps for centerline instrumentation
- Development of remote welding systems for end cap installation, pressurization, and leak checking
- Attachment of instrumentation to fuel rodlets.

-	Inspection of fuel rod (includes neutron radiography)
	First and method (includes neutron radiography)
	Fuel rod cut to length
	De-fuelling of fuel rod ends
	Oxide layer removed from cladding ends
•	End plugs welded to fuel rod (pressure transducer and
	thermocouple base plugs)
•	Fuel rod filled with liquid CO2 and frozen with liquid N2
	Drilling of centre hole (vacuum process)
+	Assembly of Mo-tube
•	Fuel rod dried at 300°C for 72 hours (vacuum)
	Second part of pressure transducer end plug welded to fuel rod
	Second part of thermocouple end plug welded to fuel rod
0.0	Measurement of fuel rod free-colume and gas flow properties
1.00	Surface TC attachment (welding)
	Fuel rod evacuated, filled with He and seal welded
-	He leak-test of fuel rod
-	Check-out / testing of fuel rod instrumentation
1.00	Final inspection of fuel rod



Baseline Refabrication

- Advanced Reinstrumentation

Accomplishments

The program accomplished many goals this year and this positive momentum will continue as INL continues its objective of developing state of the art refabrication and reinstrumentation capabilities.

Specific accomplishments include:

• Design and development of methodologies to perform integral junction cladding surface thermocouple welds using previously manufactured out-of-cell circumferential welding systems. Because of the complexity of visualizing, positioning, and welding thermal couple (TC) integral junctions to fuel cladding, an in-cell jig that interfaced with the existing welding systems needed developed. This jig would function as a TC wire positioner, holder, and heat sink, all of which Figure 1. Distinction between basic refabrication and advanced reinstrumentation activities



Figure 2. Out-of-cell welding system with first jig iteration

was fabricated and tested out-ofcell. Figure 2 shows the out-of-cell welding system with the first iteration of the TC holder jig inside the chuck. Initially, each weld was attempted with a single arc. After adding heat, the results produced the necessary burn back to "cut" the wire free from the jig, however the wire tended to ball up at the base of the heat sink. If the current was increased to get the proper washup, a substantial divot in the weld was produced. Further experimentation utilized running a lower current on two welds on each wire. The first weld would cut the wire free from the heat sink and the second weld would wash the wire end into the puddle. It was observed in previous circumferential welds with end caps on an oxidized part, that when proper gas coverage was provided, the oxide layer at and next to the weld was eliminated/reduced. Based on these results, it's anticipated that the first arc strike on setup

on each weld TC junction, when done on an oxidized sample, will work for removing the oxide layer at a TC junction point reducing porosity in the weld. A total of 5 different jigs were fabricated and experimented with. The final configuration, called Jig revision 5 (see Figure 3), added a second linear guide to square up the heat sinks. This helped improve contact surface of the heat sink to the cladding to ensure welds were more consistent. Three TC wire sets were loaded into the jig resulting in three successful welds with satisfactory results on wash up and color. Figure 4 shows the results of one of the three welds.

• Refined dry drilling techniques of UO₂ pellets utilizing the existing Hot Fuel Examination Facility (HFEF) mini-mill equipment. The basic refabrication steps were demonstrated in fiscal year (FY)21 inside the HFEF. These steps (see Figure 5) included sectioning, defueling a section of an irradiated fuel pin for a new spacer, wave springs, drilling, and end caps welding. During refabrication, a carbide endmill was used and able to drill through the fuel. A continuation of this research was performed to demonstrate more advanced refabrication on surrogate and preirradiated fuel pins. These refabrication steps included defueling, drilling a centerline hole in the fuel, supporting the hole with a molybdenum tube, and enabling the insertion of a thermocouple down the centerline of the fuel to monitor the temperature during further research. In FY23, there were three primary refabrication processes researched and experimented on.

Process #1 included centerline hole drilling followed by pressing in a support tube, defueling, then placing in a loose fit spacer. This process yielded decent results for the centerline hole and the initial tube pressing. It required a center drill with a drill bit to create a pilot hole, which adds a separate piece of equipment and step. However, the molybdenum tube did not hold up well during the defueling step. While the endmill cut through the hard cordierite, the tube split and cracked causing major top surface damage to the cordierite pellets. The tube required another operation with a drill bit to re-open the end of the tube so it could accept a thermocouple and the spacer, being a smaller outside diameter, slipped into the tube. This process also required a larger inside diameter to ensure the centerline hole lined up with the spacer.

Process #2 included defueling, pressing a spacer with an interference fit, centerline hole drilling through the spacer with a diamond core drill, then a drill bit through the surrogate fuel, and finally pressing in the support tube. The process resulted in a successful installation of a support tube inside the centerline hole. This was advantageous as it ensured all centerline holes through the spacer and the surrogate pellets were in line. However, the process to drill through the space on the mini-mill with no coolant and manual controls was very time consuming and difficult. It required several diamond core drills to be used and replaced while drilling due to excessive heat damaging the core drills.





Performing this process in-cell would more than likely prove to be infeasible.

Process #3 included defueling, pressing in a pre-drilled spacer with a slight interference fit, centerline hole drilling through the surrogate fuel, and finally pressing in the support tube. While the previous processes had somewhat acceptable results with varying degrees of success, this method utilized general steps of refabrication but with a different spacer design. Typical spacer designs are a loose fit to apply compression, from the wave springs, on fresh fuel stacks to hold them into place. This particular spacer was designed to have a slight interference fit into the cladding (.0000 - .0005 inches). The

Figure 3. Jig revision 5

Figure 4. Welds using Jig revision 5

With the closure of the Halden Reactor Project, INL is developing state of the art capabilities needed to refabricate and reinstrument previously irradiated materials.



Figure 5. Refabrication process on fuel rodlet



Figure 6. Refabrication test sample results

spacer also had a centerline hole drilled through approximately .006 to .010 inches larger than the drill bit. This accomplished three results. First, the spacer was used to stabilize the defueled face from collapsing into the defueled region during tool changeout and drilling; second, it reduced the removal of excess fuel around the drill bit during centerline drilling due to the tight hole; and third and most importantly, the centerline hole in the spacer acted as a pilot hole with a depth of the spacer thickness for the fuel centerline hole drill bit. This ensured the hole was drilled centered in the fuel and kept the drill bit from wandering as it passed through cracks in the fuel. Following centerline drilling the tubing was then pressed into the hole. This process was done to aid in the pressing of the spacer with the hot cell manipulators. This third process created the most accurate hole with only a minimal gap around the molybdenum tubing in the surrogate rodlets. Minimizing the gap around the molybdenum tubing was important as it demonstrated minimizing the amount of surrogate fuel lost. Figure 6 illustrates these 3 processes, which were each analyzed by use of x-ray technology.

• Completed a pre-conceptual design of a vertically oriented cryogenic drilling system that is compatible with HFEF existing infrastructure. Given the longterm success the HRP had with cryogenically drilling centerline holes in fuel, INL contracted with HRP to develop a pre-conceptual design of their cryogenic drilling module. This design was adapted to a vertical orientation to better fit within the physical constraints of HFEF. There are numerous reasons why INL made the decision to explore this cryogenic capability. First, although promising results had been achieved via dry drilling out-of-cell, there remained a gap and associated risk in successfully deploying dry drilling on actual irradiated materials in the hot cell. Second. based on an evaluation in the previous year, it suggested that a cryogenic drilling system would be plausible, although expensive due to the design, fabrication, and installation of a new main cell feedthrough. Third, based on the positive results utilizing an out-ofcell cryo-drilling module, it made sense to follow a similar pathway to an in-cell capability. Fourth, the HRP cryogenic equipment was designed to operate horizontally. Relative to space allocation, it was determined that with the generous vertical space available in HFEF,

the system must be designed to better fit. This would also allow a simpler approach to operability, such as getting the required utilities installed to the system.

 Completed a pre-conceptual design of a seal welding system capable of accommodating higher pressures, longer rodlets, and able to function with rodlets that already have surface thermocouples attached. In discussions with industry experts, established future fuel rodlet designs will have different design parameters (e.g., maximum pressure, rodlet length, rodlet diameter. etc.) than what was originally intended for the existing remote welding systems at INL. As discussed previously, while developing and testing integral junction TC wire welding on 3,000 psi pressurized rodlets, researchers noticed that weld puddles caused the internal pressures to blow out of the heated portion of the cladding. A potential solution to this problem would be to perform the TC welds prior to rodlet pressurization, which would require the existing welding equipment be modified allowing for these welds with already attached instrumentation. To adequately address the future rodlet geometries with instrumentation, a very large volume chamber would be required. A new remote welding system concept was developed that would allow the chamber to be attached to the end of a rodlet assembly. This concept utilizes a Swagelok fitting that would be part of the rodlet end cap. The Swagelok fitting would support the rodlet and wouldn't require a



support frame for the rodlet during the weld. This allows for open space around the rodlet for instrumentation and TC wires. Figure 7 illustrates this new Swagelok concept with the following features:

- Smaller volumes of inert gas to perform the welds
- Smaller volume pressure vessel with less potential energy
- Allows for instrumentation to be attached to the rodlet
- Allows for helium sniffing of circumferential welds for indication of leaks before seal weld is made
- Allows for longer rodlet assembly lengths

Figure 7. New welding system concept with Swagelok connection

Roadmaps and Collaborative Opportunities for Fast Neutron Irradiations

Principal Investigators: Nicolas Woolstenhulme Team Members/Collaborators: Mike Worrall, Calvin Downey

Neutronic assessments have shown that existing thermal spectrum test reactors can use spectral modification strategies to support meaningful collaborative irradiation opportunities for fast reactor fuels and materials.

nterest in fast spectrum reactors is at an all-time high for the present generation of nuclear technology developers. However, developing innovative fast reactor fuel designs remains encumbered without fast spectrum test reactors available in or to the U.S. Irradiation of fuel rodlets in cadmium-lined baskets in inner-core positions of the thermal spectrum Advanced Test Reactor (ATR) has long been used to reject thermal neutrons to help develop more prototypic radial power distributions. While useful, this design is not currently capable of housing rodlets long enough for some data needs, nor does it provide enough fast neutron flux for studying desired effects in materials. The Advanced Fuel Campaign (AFC) has been investigating other strategies for fast neutron irradiations in recent years, and other collaborative opportunities have emerged in this area. A workshop was held and neutronic studies were performed in fiscal year 2023 to explore cooperative scenarios more fully.

Project Description:

In May 2023 Idaho National Laboratory hosted the first "Fast Neutron Irradiation Testbed Workshop" to engage potential users from national labs and industry. A variety of fast reactor and fusion plant developers were represented along with the main four U.S. material test reactors (ATR, Transient Reactor Test Facility (TREAT), High Flux Isotope Reactor (HFIR), and Massachusetts Institute of Technology Reactor (MITR). The outcomes of this workshop were fruitful, yet somewhat nascent and hence difficult to describe in concrete terms. Developers of sodium fast reactors (SFR) see opportunities to use existing thermal spectrum reactors but have largely formulated their first-generation core design around historically proven fuel systems thus minimizing their needs for new irradiation data. All major SFR developers are currently proceeding with construction of demonstration plants and can foresee circumstances where interim surveillance can confirm that revived historic fuel technologies perform as expected. It is also foreseeable that future fuel innovations could undergo irradiations in demonstration plants, but here data obtained from material test reactors would be rather beneficial in streamlining Nuclear Regulatory Commission license amendments for lead test rods/assemblies. Compared to SFRs, developers of alternate fast reactor types (helium-cooled, lead-cooled, molten salt) do not have as much performance data for established fuel technologies. Thus, irradiation tests in existing material test reactors offer key opportunities for risk reduction. The same was found to be largely true, and perhaps

even more so, for developers of fusion reactors, especially those who have progressed their design to the point of giving earnest thought to material selection. In any case, the urgent need for fast neutron irradiation data and the angst in obtaining it without a fast spectrum test reactor were both apparent, but few are willing to state it so strongly since the reliable projects are not yet underway to create capabilities needed. These capability needs range from monumental projects like the Versatile Test Reactor or Fusion Prototypic Neutron Source, to smaller efforts such as refined spectral characterization configurations for existing test reactors.

Accomplishments:

AFC has been developing the Boosted Energy Advanced Spectrum Test (BEAST) as a next generation test rig for ATR which will use a flux trap position with a plate-type booster fuel ring to multiply incident thermal neutrons into fission fast neutrons. A multi-hole cadmiumlined basket sits within the booster ring to enable larger fuel rods to be tested in a more prototypic fast reactor spectrum. Various BEAST configurations were analyzed, and recommendations provided as detailed in report [1]. BEAST will offer the best possible approach to engineering scale fast spectrum irradiation using existing material test reactors. However, detailed engineering of BEAST will be a significant undertaking and, once in reactor, BEAST will continue to require ongoing expenditures to replace booster fuel rings and cadmium baskets as they become depleted. Thus, accelerating the realization of BEAST could benefit greatly from collaboration and potential cost sharing among potential users. AFC's support on BEAST focused on assessing conditions for fuel pins irradiations. The appreciable fission rates in the cluster of fuel pins centered in BEAST also helped boost fast neutron population, but sometimes irradiations are needed on structural materials only (such as mechanical property test specimens of cladding alloys). As part of a collaboration with the newly formed Innovative Nuclear Materials initiative, assessments were made to determine how best to provide fast neutron irradiations of materials only. Calculations showed the fast neutron population in BEAST was only slightly reduced if the center test hole were filled with material test specimens.

In 2027, ATR will begin operating in a new routine where a few High Temperature Steady State (HTSS) cycles are performed every year. These cycles will see a ~50% increase in power and resulting fast flux in the outboard A positions in the southwest lobe of the reactor. Not all cycles will be at the HTSS power level, so it



Figure 1. Illustration of the BEAST design and candidate flux trap locations



Figure 2. Location of Outboard-A positions in the SW lobe



Figure 3. Rendering of Cd-basket and cross section

Figure 4. Change in thermal neutron flux due to the presence of gadolinium-lined capsules in HFIR's central flux trap



is expected that fueled capsules will probably avoid the irregular power and temperature history they would experience in this part of the reactor. Materials test capsules, however, should be less sensitive to these changes and could benefit from the increased flux. The spectral environment was predicted for cadmium baskets in these positions and found to be respectable. This study also showed the viability of modestly increasing the uranium loading in ATR's driver fuel, but still well within the range shown viable in historic tests, to enable HTSS cycles to operate longer and reduce reactor downtime for refueling outages.

Other calculations were performed to evaluate HFIR's flux trap. HFIR's innate fast flux exceeds even that of BEAST, but the challenge lies with taming its extremely high fast flux since many fast reactor specimen materials can be sensitive to spurious effects from thermal neutron capture and transmutation. Small gadolinium-lined capsules were modeled in HFIR which showed a very good fast flux environment in the capsules, but also showed that one should expect a modest reduction in cycle length and a fairly significant decrease in thermal flux to HFIR's numerous isotope targets. Thus, use of thermal spectrum filtering in HFIR probably creates an untenable impact on other users. HFIR replaces its entire driver fuel load every cycle and cannot shuffle or arrange the core differently to compensate for these changes. The key conclusion is that HFIR offers the best fast flux environment available, but only for small specimens made from materials not overly sensitive to very high thermal flux.

References:

 N. Woolstenhulme, J. Brookman, C. Jesse, C. Downey, C. Murdock, "Scoping Study for Fast Flux Testing in the Advanced Test Reactor," April 2023, INL/RPT-23-72212.

Location	Thermal Flux (<0.625 eV) (n/cm2-s)	Fast Flux (>0.1 MeV) (n/cm2-s)	Fast-to-Thermal Ratio
ATR BEAST	1.39E+13	7.46E+14	53.7
ATR Outboard-A in Optimized HTSS Cycles	1.47E+13	5.55E+14	45.6
HFIR Central Flux Trap with Gd Capsule	2.82E+13	1.17E+15	41.6

2023 AFC ACCOMPLISHMENTS

APPENDIX

5.1 Puł	olications
---------	------------

- 5.2 FY-23 Level 2 Milestones
- 5.3 FY-23 Milestone Report List
- 5.4 AFC Nuclear Energy University Projects (NEUP) Grants
- 5.5 Acronyms
- 5.6 Divider Photo Captions

5.1 PUBLICATIONS

Author	Title	Publication
Adorno Lopes, D., Claisse, A., Metzger, K. & Lahoda, E.	Westinghouse Use of Advanced Material Modeling to Accelerate Fuel Development and Qualification	Proceedings of the TopFuel 2022 Light Water Reactor Fuel Performance Conference, Raleigh, NC, October 9–13, 2022, https:// www.ans.org/pubs/proceedings/ article-52177/
Boone, M., Wenzel, D. & Guler, C.	Westinghouse High Energy Fuel for the Future	Proceedings of the TopFuel 2022 Light Water Reactor Fuel Performance Conference, Raleigh, NC, October 9–13, 2022, https:// www.ans.org/pubs/proceedings/ article-52152/
Capps, N. Ridley, M. Yan, Y. Bell, S. & Kane, K. (2023).	BISON validation to in situ cladding burst test and high-burnup LOCA experiments	Annals of Nuclear Energy, Volume 191, pp. 109905, ISSN 0306- 4549, https://doi.org/10.1016/j. anucene.2023.109905
Carvajal, J., Stafford, S., Arndt, J., Sirianni, P., Tatli, E., Lahoda, E. & Heibel, M.	In-Rod Sensor System for Accelerated Fuel Qualification and Enhanced Power Distribution Applications	Proceedings of the TopFuel 2022 Light Water Reactor Fuel Performance Conference, Raleigh, NC, October 9–13, 2022
Chikhalikar, A., Roy, I., Abouelella, H., Umretiya, R., Hoffman, A., Larsen, M. & Rebak, R. (2022)	Effect of aluminum on the FeCr(Al) alloy oxidation resistance in steam environment at low temperature (400°C) and high temperature (1200°C)	Corrosion Science, doi: https://doi. org/10.1016/j.corsci.2022.110765
Czerniak, L., Lahoda, E., Metzger, K., Sivack, M., Gazza, J., Olson, J. & Gonderman, S.	Development of Silicon Carbide as a Nuclear Fuel Cladding	Proceedings of the TopFuel 2022 Light Water Reactor Fuel Performance Conference, Raleigh, NC, October 9–13, 2022
Dunbar, C., Jung, W., Demo, T., Corradini, M., Sridharan, K., Yeom, H., Armstrong, R., Kamerman, D., Maier, B., Hoffman, A. & Rebak, R.	Investigation of Cladding Thermal Behavior Under Simulated Reflood Conditions	Proceedings 20th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH), Washington DC, August 2023
Durham, B., Franceschini, F., Lam, H. & Wagener, C.	Levelized Cost of Electricity Evaluations for Westinghouse Advanced Fuel Management with High Enrichment and High Burnup Fuel to Optimize PWR Economics	Proceedings of the TopFuel 2022 Light Water Reactor Fuel Performance Conference, Raleigh, NC, October 9–13, 2022

Author	Title	Publication
Gagen, D., Lyons, J., Ricketts, B., Sivack, M., Lahoda, E., Metzger, K., Adorno Lopes, D., Claisse, A., White, J., McClellan, K., Stull, J. & Widgeon Paisner, S.	Steam Oxidation Behavior of Sintered High-Density Uranium Nitride Pellets Via Thermogravimetric Analysis	Proceedings of the TopFuel 2022 Light Water Reactor Fuel Performance Conference, Raleigh, NC, October 9–13, 2022
Garrison, B. Massey, C. Ren, W. Gussev, M. Graening, T. Sitterson, R. Capps, N. & Linton. K. (2023)	Mechanical Properties of Zircaloy Cladding Tubes and Contributions to M.E.T.A. Mechanical Property Database	ORNL/TM-2023/3046
Garrison, B., Massey, C., Gussev, M., Capps, N., Harp, J. & Linton, K.	Optimizing Nuclear Cladding Mechanical Property Output for Hot-cell Testing	Presented at The Minerals, Metals & Materials Society (TMS) 2023 Annual Meeting & Exhibition, San Diego, CA March 19-23, 2023
Garud, Y. & Rebak, R. (2023)	Effect of surface oxides on tritium entrance and permeation in FeCrAL alloys for nuclear fuel cladding: a review	Corrosion Reviews, Volume 41 (2), pp. 143-169. https://doi. org/10.1515/corrrev-2022-0033
Gong, B., Kardoulaki, E., Yang, K., Broussard, A., Zhao, D., White, J., McClellan, K., Lahoda, E. & Lian, J. (2022)	UN and U3Si2 composites densified by spark plasma sintering for accident-tolerant fuels	Ceramic International, Volume 48, pp. 10762, https://doi. org/10.1016/j.ceramint.2021.12.292
Hallman, L., Mitchell, D., Long, Y., Olson, L., Claisse, A., Karoutas, Z., Lahoda, E. & Wright, J.	Westinghouse Advanced Doped Pellet Technology (ADOPTTM) Fuel for PWR Applications	Proceedings of the TopFuel 2022 Light Water Reactor Fuel Performance Conference, Raleigh, NC, October 9–13, 2022
Hansen, R., Kamerman, D., Petersen, P. & Cappia, F. (2023)	Evaluation of the Ring Tension Test (RTT) for robust determination of material strengths	International Journal of Solids and Structures, Volume 282, pp.112471
He, M., Chen, M., Villarreal, E., Ban, H. & Rebak, R. (2022)	Experimental investigation of power transient flow boiling	Experimental Thermal and Fluid Science, Volume 144, pp. 110833, https://doi.org/10.1016/j. expthermflusci.2022.110833
Hoffman, A., Umretiya, R., Crawford, C., Spinelli, I., Huang, S., Buresh, S., Perlee, C., Mandal, T., Abouelella, H. & Rebak, R. (2023)	The relationship between grain size distribution and ductile to brittle transition temperature in FeCrAL alloys	Materials Letters, Volume 331, pp. 133427, https://doi.org/10.1016/j. matlet.2022.133427
Howard, T., Moussaoui, M., Miller, J., Weiss, A. & Marcum, W.	Overview of the Out-of-Pile Transient, Blowdown, Reflood, and CHF Experiment at OSU	Proceedings of NURETH-19, March 2022

Author	Title	Publication
Hu, C., Labuz, J. Koyanagi, T. & Le, J. (2023)	Mechanistic modeling of lifetime distribution of SiC/SiC composite claddings	Journal of the American Ceramic Society, Volume 106(5), pp. 3066-3077. https://doi.org/10.1111/ jace.18956
Kamerman, D. (2023)	The deformation and burst behavior. of Zircaloy-4 cladding tubes with hydride rim features subject to internal pressure loads	Engineering Failure Analysis, Volume 153, pp.107547 https://doi.org/10.1016/j. engfailanal.2023.107547
Karoutas, Z., Mitchell, D., Sung, Y., Wang, G., Byers, W. & Conner, M.	Use of Chromium Coating to Provide Protection of Zirconium Cladding at DNB Conditions	Proceedings of the TopFuel 2022 Light Water Reactor Fuel Performance Conference, Raleigh, NC, October 9–13, 2022
Karoutas, Z., Metzger, K., Pitruzzella, E., Vallencour, C., Atwood, A., Boone, M., Mueller, A., Gower, E., Lahoda, E., Jaworski, A., Parsi, A., Anness, M., Haas, C., Limbäck, M. & Wright, J.	Westinghouse EnCore Accident Tolerant Fuel and High Energy Program	Proceedings of the TopFuel 2022 Light Water Reactor Fuel Performance Conference, Raleigh, NC, October 9–13, 2022
Koyanagi, T., Karakoc, O., Hawkins, C., Lara-Curzio, E., Deck, C. & Katoh, Y. (2023)	Stress rupture of SiC/SiC composite_ tubes under high-temperature steam	International Journal of Applied Ceramic Technology, Volume 20(3), pp. 1658-1666. https://doi. org/10.1111/ijac.14283
Luna, A., Anderson, M., Corradini, M.	Deconvolving of Strain from Fiber Optic Sensor Measurements at Elevated Temperatures	Transactions of the ANS Winter Meeting, Phoenix AZ, November 2022
Miller, J., Howard, T. & Marcum, W.	Thermal Hydraulic Analysis of Annular Natural Circulation	Transactions of the ANS Winter Meeting, Phoenix AZ, November 2022
Moussaoui, M., Miller, J., Howard, T. & Marcum, W.	Out-of-Pile CHF LOCA Experiment	Transactions of the ANS Winter Meeting, Phoenix AZ, November 2022
Moussaoui, M., Miller, J., Howard, T. & Marcum, W.	Out-of-Pile Transient Blowdown Experiment	Proceedings of NURETH-20, August 2023
Olson, L., Pitruzzella, E., Walters, J., Roberts, E., Mitchell, D., Mueller, A., Hallman, L., Pan, G., Metzger, K., Maier, B., Lyons, J., Jaworski, A., Lahoda, E., Kobelak, J., Shockling, M., Capps, N. & Harp, J.	Accident Tolerant and High Burnup Hotcell PIE at ORNL	Proceedings of the TopFuel 2022 Light Water Reactor Fuel Performance Conference, Raleigh, NC, October 9–13, 2022
Parish, C., Werden, J., Gerczak, T., Harp, J., McKinney, C. & Capps, N. (2023)	STEM analysis of high burnup structure in LWR fuels	Microscopy and Microanalysis, Volume 29(1) pp. 1543-1545
Author	Title	Publication
---	--	--
Qu, H, Abouelella, H. Chikhalikar, Rajendran, A., Roy, I. Priedeman, J., Hoffman, A., Wharry, J., Rebak, R. (2023)	Effect of nickel on the oxidation. behavior of FeCrAl alloy in simulated PWR and BWR conditions	Corrosion Science, Volume 216, pp. 111093, doi: https://doi. org/10.1016/j.corsci.2023.111093
Qu, H., Higgins, M., Abouelella, H., Cappia, F., Burns, J., He, L., Massey, C., Harp, J., Field, K., Howard, R., Umretiya, R., Hoffman, A., Wharry, J. & Rebak, R. (2023)	FeCrAl fuel/clad chemical interaction in light water reactor environments	Journal of Nuclear Materials, Volume 587, pp. 154717, https://doi.org/10.1016/j. jnucmat.2023.154717
Ratnayake, R., Becker, R., Wang, G., Byers, W., Olson, L. & Hussey, D.	ATF Coating Response in KOH Adjusted PWR Water Under Extended Exposure	Proceedings of the TopFuel 2022 Light Water Reactor Fuel Performance Conference, Raleigh, NC, October 9–13, 2022
Ravi, S., Roy, I., Roychowdhury, S., Feng, B., Ghosh, S., Reynolds, C., Umretiya, R., Rebak, R. & Hoffman, A. (2023)	Elucidating precipitation in FeCrAL alloys through explainable AI: A case study	Computational Materials Science, Volume 230, pp. 112440, https://doi.org/10.1016/j. commatsci.2023.112440
Rebak, R. (2023)	Improved and innovative accident- tolerant nuclear fuel materials. considered for retrofitting light water reactors – a review.	Corros. Mater. Degrad., Volume 4, pp. 466-487 https://doi. org/10.3390/cmd4030024
Rebak, R. (2023)	Innovative accident tolerant fuel materials may help extending the life of light water reactors: A perspective	Corrosion, Volume 79 (3), pp. 328–332. https://doi. org/10.5006/4181
Rebak, R., Yin, L., Zhang, W. & Umretiya, R. (2023)	Effect of the redox potential on the general corrosion behavior of industrial nuclear alloys	Journal of Nuclear Materials, Volume 576, pp. 154257, https://doi.org/10.1016/j. jnucmat.2023.154257
Ridley, M., Massey, C., Bell, S. & Capps, N. (2023)	High temperature creep model. development using in-situ 3-D DIC. techniques during a simulated LOCA. transient	Annals of Nuclear Energy, Volume 193, pp. 110012, ISSN 0306- 4549, https://doi.org/10.1016/j. anucene.2023.110012
Roy, I., Abouelella, H., Rajendran, R., Chikhalikar, A., Larsen, M., Umretiya, R., Hoffman, A. & Rebak, R. (2023)	Effect of Al concentration on Fe-17Cr alloy during steam oxidation at 400 °C	Corrosion Science, Volume 217, pp. 111135, doi: https://doi. org/10.1016/j.corsci.2023.111135
Roy, I., Roychowdhury, S., Feng, B., Ravi, S., Ghosh, S., Umretiya, R., Rebak, R., Ruscitto, D., Gupta, V. & Hoffman, A. (2023)	Data-driven predictive modeling of FeCrAl oxidation	Materials Letters, Volume 17, pp. 100183, https://www.osti.gov/ biblio/1958676



Author	Title	Publication
Roy, I., Bojun Feng, Roychowdhury, S., Ravi, S., Umretiya, R., Christopher Reynolds, Ghosh, S., Rebak, R., Hoffman, A. (2023)	Understanding oxidation of Fe-Cr-Al alloys through explainable artificial intelligence	MRS Communications," Volume 13, pp. 82-88, https://doi. org/10.1557/s43579-022-00315-0
Schmitz, M., Kim, J.Y. & Jacobs, L.J. (2023)	Machine and deep learning for. coating thickness prediction using. Lamb waves	Wave Motion, Volume 120, pp. 103137, https://doi.org/10.1016/j. wavemoti.2023.103137
Woolstenhulme, N., Brookman, J., Jesse, C. Downey, C. & Murdock, C. (2023)	Scoping Study for Fast Flux Testing in the Advanced Test Reactor	INL/RPT-23-72212
Yang, K., Kardoulaki, E., Zhao, D., Broussard, A., Metzger, K., White, J., Sivack, M., Mcclellan, K., Lahoda, E. & Lian, J. (2021)	Uranium nitride (UN) pellets with controllable microstructure and phase – fabrication by spark plasma sintering and their thermal- mechanical and oxidation properties	Journal of Nuclear Materials, Volume 557, pp. 153272, ISSN 0022-3115, https://doi. org/10.1016/j.jnucmat.2021.153272
Yang, K., Kardoulaki, E., Zhao, D., Gong, B., Broussard, A., Metzger, K., White, J., Sivack, M., Mcclellan, K., Lahoda, E. & Lian, J. (2022)	Cr-incorporated uranium nitride. composite fuels with enhanced. mechanical performance and oxidation resistance	Journal of Nuclear Materials, Volume 559, pp. 153486, ISSN 0022-3115, https://doi. org/10.1016/j.jnucmat.2021.153486
Yeom, H., Dabney, T., Kamerman, D., Bales, M., Heuser, B. & Sridharan, K.	Experimental Investigation of Degradation Mechanism of ATF Coated Cladding under Transient Conditions	Proceedings of the 2022 ANS Winter Meeting and Nuclear Technology Expo, Phoenix, AZ
Yuan, G., Forna-Kreutzer, J. Xu, P., Gonderman, S., Deck, C., Olson, L., Lahoda, E., Ritchie, R. & Liu, D. (2023)	In situ high-temperature 3D imaging of the damage evolution in a SiC nuclear fuel cladding material	Materials & Design, Volume 227, pp. 111784, https://doi. org/10.1016/j.matdes.2023.111784
Zhong, W., Wang, H., McAuliffe, R.D., Yan, Y., Curlin, S., Graening, T. & Nelson, A. (2023)	Hydrogen effects on thermal diffusivity and electrical resistivity of zircaloy cladding	Journal of Nuclear Materials Volume 574, pp. 154213, https://doi.org/10.1016/j. jnucmat.2022.154213
Zhong, W., Wang, H., Godfrey, A., Ridley, M. Su, Y.F., Graening, T., Linton, K. & Nelson, A. (2023)	Summary of Laser Flash Analysis measurements on Cr-coated cladding	(No. ORNL/SPR-2023/2898). Oak Ridge National Lab, Oak Ridge, TN (U.S.)

5.2 FY-23 LEVEL 2 MILESTONES

Work Package Title	Site	Work Package Manager	Level 2 Milestone
Complete ARES Phase I project - irradiation and PIE – INL	INL	Smuin, Trevor	Complete assembly and irradiation of first previously irradiated pin in THOR
Integral Irradiation Testing - INL	INL	Hoggard, Gary	Issue ATF-2 Ramp Conceptual Design Report
HERA PreH Campaign - INL	INL	Kleimenhagen, Kip	First two experiments on pre-hydrided cladding ready for irradiation
High Performance ATF (SiC) Cladding Development - ORNL	ORNL	Koyanagi, Takaaki	SiC/SiC Development Strategy and 5-yr Execution Plan
Fabrication and Characterization of Coated Cladding - ORNL	ORNL	Capps, Nathan	Summary of LFA measurements on Cr-coated cladding tubes
Complete ARES Phase I project - irradiation and PIE – INL	INL	Smuin, Trevor	Complete THOR-MOXTOP test plan in BUSTER (nominally 4 transients)
Design and Install I-loop - INL	INL	Tonc, Vincent	Complete Transfer Shield Plate final design
Design and Install I-loop - INL	INL	Tonc, Vincent	Initiate procurement of four primary loop coolant pumps
TWIST Commissioning - INL	INL	Fife, Cindy	First commissioning test assembled
HERA Hotcell - INL	INL	Kamerman, David	PIE report on HERA PreH Series
Mechanical testing and modeling of Coated Zircaloy - LANL	LANL	Eftink, Ben	Complete high temperature ringpull tests on Cr coated Zircaloy Tube
Accelerated Ceramic Fuel Development - LANL	LANL	White, Josh	Prepare Handbook for ATF doped UO2
ATF/HBU Transient Testing - INL	INL	Pavey, Todd	Complete HBU LOCA experiment preliminary design

Work Package Title	Site	Work Package Manager	Level 2 Milestone
Complete ARES Phase I project - irradiation and PIE – INL	INL	Smuin, Trevor	Preliminary assessment and PIE report for THOR metallic fuel experiments
Design and Install I-loop - INL	INL	Tonc, Vincent	Complete conceptual design Cubicle and Cleanup System
ATF PIE INL - INL	INL	Stockwell, Jake	Coated Cladding Performance in ATF2 Comparison Report
Fabrication and Characterization of Coated Cladding - ORNL	ORNL	Capps, Nathan	Summary of the impact of the coating on high temperature creep and burst margin
High Burnup Fuel Performance - ORNL	ORNL	Capps, Nathan	Report summarizing tFGR and visualization data on high burnup
AR Campaign Management - INL	INL	Jensen, Colby	Draft Metallic Fuel Research and Development Plan
AF Irradiation Testing & PIE - INL	INL	Murdock, Chris	Draft manuscript on destructive PIE and assessment on FAST-1 rodlets
Na loop infrastructure at IDEF - INL	INL	Cole, Mark	Place order for a new glovebox
Na loop infrastructure at IDEF - INL	INL	Cole, Mark	Receive the sodium Test Loop from TerraPower
Matrix/BOR-60 shipment support plus line 35 type SOWs - LANL	LANL	Saleh, Tarik	Summary of HT9 and CWD9 cladding properties, design equations, and data gaps

5.3 FY-23 MILESTONE REPORT LIST

Work Package Title	Site	Work Package Manager	Milestone Title
ATF Fuel Safety - ORNL	ORNL	Linton, Kory	FY22 AFC HFIR Irradiations Status including RB Minifuel Experiment Readiness
Advanced cladding develop- ment – PNNL	PNNL	Maloy, Stuart	Submit Report on FFTF/MOTA Tempered Martensitic Steels available for Testing
AFC Coordination and Integra- tion - INL	INL	Mai, Edward	Complete Draft 2022 Accomplishments Report
HERA – ORNL	ORNL	Linton, Kory	Summary of Byron irradiated fuel ship- ment planning activities
Matrix/BOR-60 shipment support plus line 35 type SOWs - LANL	LANL	Saleh, Tarik	Progress Report on Elevated tempera- ture creep models
ATF/HBU Shipment Support - INL	INL	Mai, Edward	Prepare and issue ATF national trans- portation and PIE Plan
Integral Irradiation Testing - INL	INL	Hoggard, Gary	Issue ATF-2 Ramp Conceptual Design Report
High Performance ATF (SiC) Cladding Development – ORNL	ORNL	Koyanagi, Takaaki	SiC/SiC Development Strategy and 5-yr Execution Plan
Development of cross cutting advanced ceramic fuel concepts – LANL	LANL	Kardoulaki, Eri	Assessment of advanced ceramic composites for cross-cutting reactor applications
Fabrication and Characteriza- tion of Coated Cladding - ORNL	ORNL	Capps, Nathan	Summary of LFA measurements on Cr-coated cladding tubes
GE INL Test Pin Fabrication and PIE – INL	INL	Stockwell, Jake	Report of LEU+ Fuel Pellet Demonstra- tion
Accelerated Ceramic Fuel Development – LANL	LANL	White, Josh	Progress report on the measurement of thermophysical datasets for doped UO2 ATF
Complete ARES Phase I project - irradiation and PIE - INL	INL	Smuin, Trevor	Complete THOR-MOXTOP test plan in BUSTER (nominally 4 transients)
Complete ARES Phase I project - irradiation and PIE - INL	INL	Smuin, Trevor	Complete a summary report of the pre-transient predictions of metallic experiments
HT-9 & CWD9 cladding data- base & model evaluation/devel- opment - PNNL	PNNL	Maloy, Stuart	Provide input on Summary of HT9 and CWD9 cladding properties, design equa- tions, and data gaps to LANL

Work Package Title	Site	Work Package Manager	Milestone Title
AFC Coordination and Integra- tion - INL	INL	Mai, Edward	Update AFC Execution Plan
Accelerated Ceramic Fuel Development - LANL	LANL	White, Josh	Assessment and initial fabrication of novel ATF dopants
ATF Fuel Safety - ORNL	ORNL	Linton, Kory	Community contribution of Zr-4 data to cladding mechanical properties database
HERA Hotcell - INL	INL	Kamerman, David	PIE report on HERA PreH Series
Framatome INL PIE - INL	INL	Stockwell, Jake	Final PIE Report on Framatome 30 GWd Rods
Framatome INL Irradiation Testing - INL	INL	Hoggard, Gary	Issue ATF-2 Irradiation Testing Report
Development of cross cutting advanced ceramic fuel concepts - LANL	LANL	Kardoulaki, Eri	Development of advanced ceramic nuclear fuel composites
Assessment of FAST application to U-10Zr/HT9 system - INL	INL	Beausoleil, Geoffrey	Draft Manuscript detailing the results and conclusions
Accelerated Ceramic Fuel Development - LANL	LANL	White, Josh	Prepare Handbook for ATF doped UO2
Accelerated Ceramic Fuel Development - LANL	LANL	White, Josh	Progress report on coordination efforts towards advanced non-destructive pulsed neutron PIE of HBU fuel
Complete ARES Phase I project - irradiation and PIE - INL	INL	Smuin, Trevor	Preliminary assessment and PIE report for THOR metallic fuel experiments
ATF Fuel Performance Assess- ment - BNL	BNL	Todosow, Mike	Assessment of Performance of Advanced LWR Fuel Concepts in Normal and Accident Conditions -Summary Report on FY22 Activities
ATF PIE INL - INL	INL	Stockwell, Jake	Coated Cladding Performance in ATF2 Comparison Report
Fabrication and Characteriza- tion of Coated Cladding - ORNL	ORNL	Capps, Nathan	Summary of the impact of the coating on high temperature creep and burst margin
High Burnup Fuel Performance - ORNL	ORNL	Capps, Nathan	Report summarizing tFGR and visualiza- tion data on high burnup
AR Campaign Management - INL	INL	Jensen, Colby	Draft Metallic Fuel Research and Devel- opment Plan

Work Package Title	Site	Work Package Manager	Milestone Title
AF Irradiation Testing & PIE - INL	INL	Murdock, Chris	Draft manuscript on destructive PIE and assessment on FAST-1 rodlets
Matrix/BOR-60 shipment support plus line 35 type SOWs - LANL	LANL	Saleh, Tarik	Summary of HT9 and CWD9 cladding properties, design equations, and data gaps
Westinghouse INL PIE - INL	INL	Stockwell, Jake	Final PIE Report on Westinghouse ATF-2B Fuel Pins
AF Irradiation Testing & PIE - INL	INL	Murdock, Chris	MFF Pins PIE Status Report
SiC Modelling Performance - INL	INL	Kamerman, David	Performance Assessment of SiC-SiC Cladding Concepts
TREAT Reinstrumentation Capability (non capital) - INL	INL	Cole, Mark	Issue a report on the progress of Rein- strumentation activities
Advanced Fab Development for Metallic Fuel - INL	INL	Fielding, Randy	EBR-II fuel fabrication experience and mass balance report
Develop and qualify metallic fuel database and SFR assess- ment- ANL	ANL	Mo, Kun	Guidance for BISON metallic fuel model development
Develop and qualify metallic fuel database and SFR assess- ment- ANL	ANL	Mo, Kun	Guidance for TREAT experiments

2023 AFC ACCOMPLISHMENTS



Lead University	Title	Principal Investigator
University of California, Berkeley	Understanding of degradation of SiC/SiC mate- rials in nuclear systems and development of mitigation strategies	Peter Hosemann
University of Minnesota, Twin Cities	Probabilistic Failure Criterion of SiC/SiC Composites Under Multi-Axial Loading	Jialiang Le
University of Wisconsin- Madison	Advanced Coating and Surface Modification Technologies for SiC-SiC Composite for Hydro- thermal Corrosion Protection in LWR	Kumar Sridharan
University of Michigan	Mechanistic Understanding of Radiolytically Assisted Hydrothermal Corrosion of SiC in LWR Coolant Environments	Peng Wang
University of Florida	Multiaxial Failure Envelopes and Uncertainty Quantification of Nuclear-Grade SiCf/SiC Woven Ceramic Matrix Tubular Composites	Ghatu Subhash
University of Notre Dame	Radiolytic Dissolution Rate of Silicon Carbide	David Bartels
University of South Carolina	Development of Multi-Axial Failure Criteria for Nuclear Grade SiCf-SiCm Composites	Xinyu Huang
University of California, Berkeley	Bridging the length scales on mechanical prop- erty evaluation	Peter Hosemann
Purdue University	Microstructure-Based Benchmarking for Nano/ Microscale Tension and Ductility Testing of Irradiated Steels	Janelle Wharry
University of Utah	Benchmarking Microscale Ductility Measure- ments	Owen Kingstedt
University of Nebraska, Lincoln	Bridging microscale to macroscale mechanical property measurements and predication of performance limitation for FeCrAl alloys under extreme reactor applications	Jian Wang

Lead University	Title	Principal Investigator
University of South Carolina	Remote laser based nondestructive evaluation for post irradiation examination of ATF cladding	Lingyu Yu
North Carolina State University	Novel miniature creep tester for virgin and neutron irradiated clad alloys with benchmarked multiscale modeling and simulations	Korukonda Murty
University of Tennessee at Knoxville	Radiation-Induced Swelling in Advanced Nuclear Fuel	Maik Lang
University of Minnesota, Twin Cities	High throughput assessment of creep behavior of advanced nuclear reactor structural alloys by nano/microindentation	Nathan Mara
University of Pittsburgh	Thermal Conductivity Measurement of Irradi- ated Metallic Fuel Using TREAT	Heng Ban
The Ohio State University	Neutron Radiation Effect on Diffusion between Zr (and Zircaloy) and Cr for Accurate Lifetime Prediction of ATF	Wolfgang Windl

Lead University	Title	Principal Investigator
Georgia Institute of Technology	Linear and nonlinear guided ultrasonic waves to characterize cladding of accident tolerant fuel (ATF)	Laurence Jacobs
Rensselaer Polytechnic Institute	Chemical Interaction and Compatibility of Uranium Nitride with Liquid Pb and Alumina- forming Austenitic Alloys	Jie Lian
University of Wisconsin- Madison	Investigation of Degradation Mechanisms of Cr-coated Zirconium Alloy Cladding in Reactivity Initiated Accidents (RIA)	Hwasung Yeom
University of Wisconsin- Madison	Maintaining and building upon the Halden legacy of In-situ diagnostics	Michael Corradini
University of California, Berkeley	Femtosecond Laser Ablation Machining & Exami- nation - Center for Active Materials Processing (FLAME-CAMP)	Peter Hosemann

Lead University	Title	Principal Investigator
University of Wisconsin- Madison	Post-DNB Thermo-mechanical Behavior of Near-term ATF Designs in Simulated Transient Conditions	Hwasung Yeom
University of Tennessee at Knoxville	Safety Implications of High Burnup Fuel for a 2-Year PWR Fuel Cycle	Nicholas Brown
University of Tennessee at Knoxville	Modeling high-burnup LWR fuel behavior under normal operating and transient conditions	Giovanni Pastore
Oregon State University	Characterizing Fuel Response and Quantifying Coolable Geometry of High-Burnup Fuel	Wade Marcum
Texas A&M University	Multiscale Modeling and Experiments for Inves- tigating High Burnup LWR Fuel Rod Behavior Under Normal and Transient Conditions	Karim Ahmed
University of Tennessee at Knoxville	Fuel-to-Coolant Thermomechanical Behaviors Under Transient Conditions	Nicholas Brown
University of Florida	High-fidelity modeling of fuel-to-coolant thermo- mechanical transport behaviors under transient conditions	Justin Watson
Massachusetts Institute of Technology	Experimental investigation and development of models and correlations for cladding-to-coolant heat transfer phenomena in transient conditions in support of TREAT and the LWR fleet.	Matteo Bucci
University of Pittsburgh	Fragmentation and Thermal Energy Transport of Cr-doped Fuels under Transient Conditions	Heng Ban
Pennsylvania State University	Estimation of low temperature cladding failures during an RIA transient	Arthur Motta

Nuclear Energy University Cooperative Agreements

Lead University	Title	Principal Investigator
Massachusetts Institute of Technology	Understanding of ATF Cladding Performance under Radiation using MITR	David Carpenter
University of Texas at San Antonio	International Collaboration to Advance the Tech- nical Readiness of High Uranium Density Fuels and Composites for Small Modular Reactors	Elizabeth Sooby
University of Wisconsin, Madison	Development of Advanced Control Rod Assembly for Improved Accident Tolerance and High Burnup Fuel Cycle	Kumar Sridharan
Purdue University	Physics-Guided Smart Scaling Methodology for Accelerated Fuel Testing	Hany Abdel-Khalik
Massachusetts Institute of Technology	ATF Solutions to Light Water-Cooled SMRs	Koroush Shirvan

2023 AFC ACCOMPLISHMENTS

Lead University	Title	Principal Investigator
Oregon State University	Getting AnCers: Metallothermic Molten Salt Synthesis and Reaction Thermodynamics of Actinide Ceramic Fuels	Alexander Chemy
University of Florida	Understanding Constituent Redistribution, Thermal Transport, and Fission Gas Behavior in U-Zr Annular Fuel Without a Sodium Bond	Michael Tonks
Brigham Young University	Improving Reliability of Novel TRISO Fuel Forms for Advanced Reactors via Multiscale, High- Throughput Characterization and Modeling	Troy Munro
University of Florida	Physics-Informed Artificial Intelligence for Non-Destructive Evaluation of Ceramic Composite Cladding by Creating Digital Fingerprints	Joel Harley
University of Wisconsin - Madison	Thermal-Hydraulics Assessment of SiC Compared to Other ATF Cladding Materials and Performance to Mitigate CRUD	Mark Anderson
University of Michigan	Grand Challenge to Accelerated Deployment of Advanced Reactors – A Predictive Pathway for Rapid Qualification of Core Structural Materials	Gary Was

5.5 ACRONYMS

AFA Alumina Forming Austenitic Alloys
AFC Advanced Fuels Campaign
AFFAdvanced Fuel Fabrication
AFQAdvanced Fuel Qualification
AI Artificial Intelligence
AIN Aluminum Nitride
AL Air Liquide
ANLArgonne National Laboratory
APT Atom Probe Tomography
ARESAdvanced Reactor Experiments for Sodium Fast Reactor Fuels
ASMScale Modeling
ATFAccident Tolerant Fuel
ATRAdvanced Test Reactor
ATTAxial Tension Test
BCC Body Centered Cubic
BDBA Beyond Design Basis Accident
BEASTBoosted Energy Advanced Spectrum Test
BFBright Field
Big-BUSTER Big-Broad Use Specimen Transient Experiment Rig
BNLBrookhaven National Laboratory
BR2Belgium Reactor – 2
BRR
BSEBackscattered Electron
BU
BWR Boiling Water Reactor
CCCTFCore Conduction Cooldown Test Furnace
CEA

CHF Critical Heat Flux
CIC Core Internal Change-out
CMCCeramic Matrix Composite
CNNConvolution Neural Network
CNWGCivil Nuclear Energy Working Group
CoCCertificate of Compliance
CRAFTCollaborative Research on Advanced Fuel Technologies
CS Cold Spray
CVD Chemical Vapor Deposition
CVI Chemical Vapor Infiltration
DEH Decay Energy Heatup
DIC Digital Image Correlation
DOEDepartment of Energy
EATF Enhanced Accident Tolerant Fuel
EBR Experimental Breeder Reactor
EBSD Electron Backscatter Diffraction
EC Constellation Energy Corporation
EDF Électricité de France
EDMElectrical Discharge Machining
EDS Energy Dispersive Spectroscopy
EDXEnergy Dispersion X-Ray
EGFM Expert Group on Fuel Materials
EGIFE Expert Working Group on Innovative Fuel Elements
EGSM Expert Group on Structural Materials
EIL Engineering Innovation Laboratory
EPRIElectric Power Research Institute
FAST Fission Accelerated Steady-state Testing

FCC
FCCI Fuel Cladding Chemical Interaction
FCMIFuel-Cladding Mechanical Interaction
FCRDFuel Cycle Research and Development
FCWM Fuel Cycle Waste Management
FEA Finite Element Analysis
FFRDFuel Fragmentation, Relocation, and Dispersal
FFTF
FGBFission Gas Bubbles
FGR Fission Gas Release
FIB Focused Ion Beam
FIDESFramework for Irradiation Experiments
FLAME-CAMP Femtosecond Laser Ablation Machining & Examination - Center for Active Materials Processing
FMP Five Metal Precipitates
FOS Fiber-optic Sensor
FPTTEG Fuel Performance and Testing Technical Experts Group
FTIR Fourier Transform Infrared
FWHM Full Width at Half Max
FYFiscal Year
GA General Atomics
GA-EMSGeneral Atomics – Electromagnetic Systems
GASRGas Assay Sample & Recharge
GBGrain Boundaries
GE General Electric
GMGrid Method
GNF Global Nuclear Fuels
HAADF High Angle Annular Dark Field
HBFFHigh Burnup Fuel Fragmentation
HBuHigh Burnup

HBHEHigh Burnup Higher Enrichment
HBWR Halden Boiling Water Reactor
HEA High Entropy Alloy
HERAHigh Burnup Experiments in Reactivity initiated Accidents
HFEFHot Fuel Examination Facility
HFIRHigh Flux Isotope Reactor
HGRHeat Generation Rates
HiPIMS Hybrid High Power Impulse Magnetron Sputtering
HP-MR Heat-Pipe Microreactor
HPWL High Pressure Water Loop
HRPHalden Reactor Project
HTSS High Temperature Steady State
HTWLHigh Temperature Water Loop
IAC Industry Advisory Committee
IAEAAInternational Atomic Energy Agency
ICP-MSInductively Coupled Plasma Mass Spectrometry
ICWUPSIn-Cell Weld Under Pressure System
IMCLIrradiated Materials Characterization Laboratory
INLIdaho National Laboratory
IWTUIntegrated Waste Treatment Unit
JAEAJapanese Atomic Energy Agency
JEEPJoint Experimental Programmes
KIT
KTHRoyal Institute of Technology
LAGB Low Angle Grain Boundaries
LA-ICP-MS Laser Ablation- Inductively Coupled Plasma - Mass Spec
LANCR Lattice Neutronic Characteristics Evaluation & Research
LANL Los Alamos National Laboratory
LAROMance Los Alamos Reduced Order model for Advanced Non-linear Constitutive Equations

Cr-barrier

1 mm



Interaction

LBLOCALarge Break Loss of Coolant Accident
LEDLight Emitting Diode
LEULow Enriched Uranium
LEXALong-term Exposure Autoclave
LFA Laser Flash Analysis
LFRLead-cooled Fast Reactor
LOCA Loss of Coolant Accident
LOFLoss of Flow
LS-EVP-FFT Large Strain Elasto-viscoplastic Fast Fourier Transform
LTALead Test Assembly
LTRLead Test Rod
LUA Lead Use Assembly
LVDT Linear Variable Differential Transformer
LWRLight Water Reactor
LWT Legal Weight Truck
M&S Modeling and Simulation
MCNP Monte Carlo N-Particle
MCS Monte Carlo Simulations
MD Molecular Dynamics
META Methodology, Evaluation, Testing, and Analysis
METLMechanisms Engineering Test Loop
MFCMaterials and Fuels Complex
MITMassachusetts Institute of Technology
MITR Massachusetts Institute of Technology Reactor
MOOSEMulti Object-Oriented Simulation Environment
MOTAMaterials Open Test Assembly
MOXMixed Oxide
MOXTOP Mixed Oxide Transient OverPower
MSCPMechanisms-based Single Crystal Plasticity

MSMMaster Slave Manipulator
MTR Material Test Rod
NA1North Anna 1
NA2North Anna 2
NAC Nuclear Assurance Corporation
NCS Nitrogen Cold Spray
NCSU North Carolina State University
nCT Neutron Computed Tomography
NDENondestructive Evaluation
NE Nuclear Energy
NEA Nuclear Energy Agency
NEDNeutronic Equivalent Device
NE Nuclear Energy
NEA Nuclear Energy Agency
NEAMSNuclear Energy Advanced Modeling and Simulation
NEFT Northeast Flux Trap
NEINuclear Energy Institute
NEUP Nuclear Energy University Project
NGS Nuclear Generating Station
nRADNeutron Radiography
NRCNuclear Regulatory Commission
NRL Nuclear Research Lab
NSC Nuclear Science Committee
NSRRNuclear Safety Research Reactor
ODOuter Diameter
ODSOxide Dispersion Strengthened
OECDOrganisation for Economic Co-operation and Development
ORNL Oak Ridge National Laboratory
OSU Oregon State University

PbLiquid Lead
PCMIPellet Cladding Mechanical Interaction
PI Principal Investigator
PICT Peak Inner Cladding Temperature
PIDProportional-Integral-Derivative
PIEPost-irradiation Examination
PNNL Pacific Northwest Nuclear Laboratory
POSTECHPohang Institute of Science & Technology
PSIPaul-Scherrer Institute
PVD Physical Vapor Deposition
PWR Pressurized Water Reactor
QA Quality Assurance
QCQuality Control
R&DResearch & Development
RD&DResearch, Development, and Demonstration
RAIRequest for Additional Information
RBRemovable Beryllium
RHTRing Hoop Test
RIA Reactivity-initiated Accident
RILResearch Information Letter
RITCRod in Tube Canister
ROI Region of Interest
RPIRensselaer Polytechnic Institute
RTTRing Tension Test
SARSafety Analysis Report
SATSSevere Accident Test Station
SCIPStudsvik Cladding Integrity Project
SDSmear Density
SEHStored Energy Heatup

SEMScanning Electron Microscopy
SERTTAStatic Environment Rodlet Transient Test Apparatus
SFFSodium Free Fuel
SFRSodium Fast Reactor
SI Structural Integrity
SiCSilicon Carbide
SMRSmall Modular Reactor
SNC Southern Nuclear Company
SNMSpecial Nuclear Materials
SP State Point
STEM Scanning Transmission Electron Microscopy
STEP Systematic Technology Evaluation Program
STS Secure Transport Services
TAVATime-average, Volume-average
TCThermal couple
TCMThermal Conductivity Microscope
TEMTransmission Electron Microscopy
tFGR Transient Fission Gas Release
THOR Temperature Heat Sink Overpower Response
THOR-CTHOR Commissioning
THOR-M THOR Metallic
TREATTransient Reactor Test Facility
TRISO Tristructural Isotropic
TWISTTransient Water Irradiation System in TREAT
UUranium
UBr University of Bristol
UCBUniversity of California Berkeley
UCOUranium Oxycarbide
UHTUltra-high Temperature



UIUC	University of Illinois -Urbana Champaign
UN	Uranium Nitride
USC	University of South Carolina
UVa	University of Virginia
UWM	University of Wisconsin-Madison
VFX	Vertical Experiment Facilities
VTR	Versatile Test Reactor
WEC	Westinghouse Electric Company
WGFS	
WPFM	Working Party for Fuels and Materials
ХСТ	X-ray Computed Tomography
XPS	X-ray Photoelectron Spectroscopy
XRD	X-Ray Diffraction
XSI	X-ray Spectrum Imaging

5.6 DIVIDER PHOTO CAPTIONS



Page 2

Transverse metallography of the FAST-008 pin across the center of the pin. (B. Beausoleil/L. Capriotti).



Page 93

Instrumented Assembled THOR-MOXTOP-1. (T. Smuin)



Page 4 Uprighting LOC-C-1 in the stand (in TREAT). (T. Pavey)



Page 107

Ring tension testing of ATF cladding with the furnace and load frame in the HFEF hot cell. (R. Hansen)



Page 83

Joseph Cummings and Travis Callison removing LOC-C-1 from the stand in preparation to placing into the Big-BUSTER. (T. Pavey)



Page 131

TEM characterization results of fuel-Cr diffusion barrier interacting interface. (Y. Wang)



Page 189

Transient Heat sink Overpower Response (THOR)-MOXTOP - 1 Instrumentation Assembly at the Measurement Science Lab (MSL). (T. Smuin)



Page 236

Cross-sectional Scanning Electron Microscopy (SEM) image of an Advanced Test Reactor (ATR) irradiated HT-9 clad/U-10Zr fuel with Cr diffusion barrier (top). The high-resolution SEM image (bottom) reveals the interaction between the fuel and the Cr diffusion barrier, but no interaction between the fuel and the clad after irradiation testing. ((Y. Wang)



Page 218

Transient Heat sink Overpower Response (THOR)-mixed oxide transient overpower (MOXTOP) - 1 pin loading at the Hot Fuel Examination Facility (HFEF) (T. Smuin)



Page 241

Characterization of SiC composite cladding after TREAT safety test by High Resolution Transmission Electron Microscopy via remote controls. Scientists: Tiankai Yao and Fei Xu (P.Xu)



Page 225

Removing BRR cask from transport trailer in HFEF. (E. Woolstenhulme)

