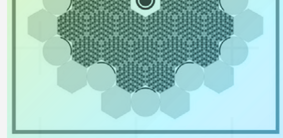
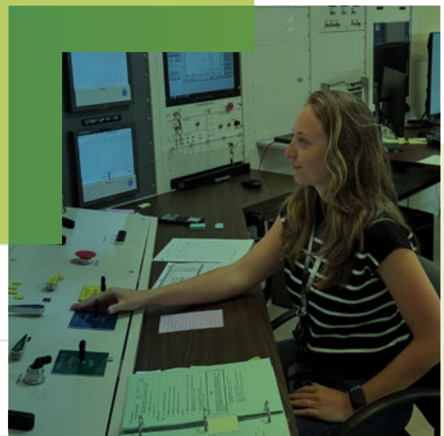
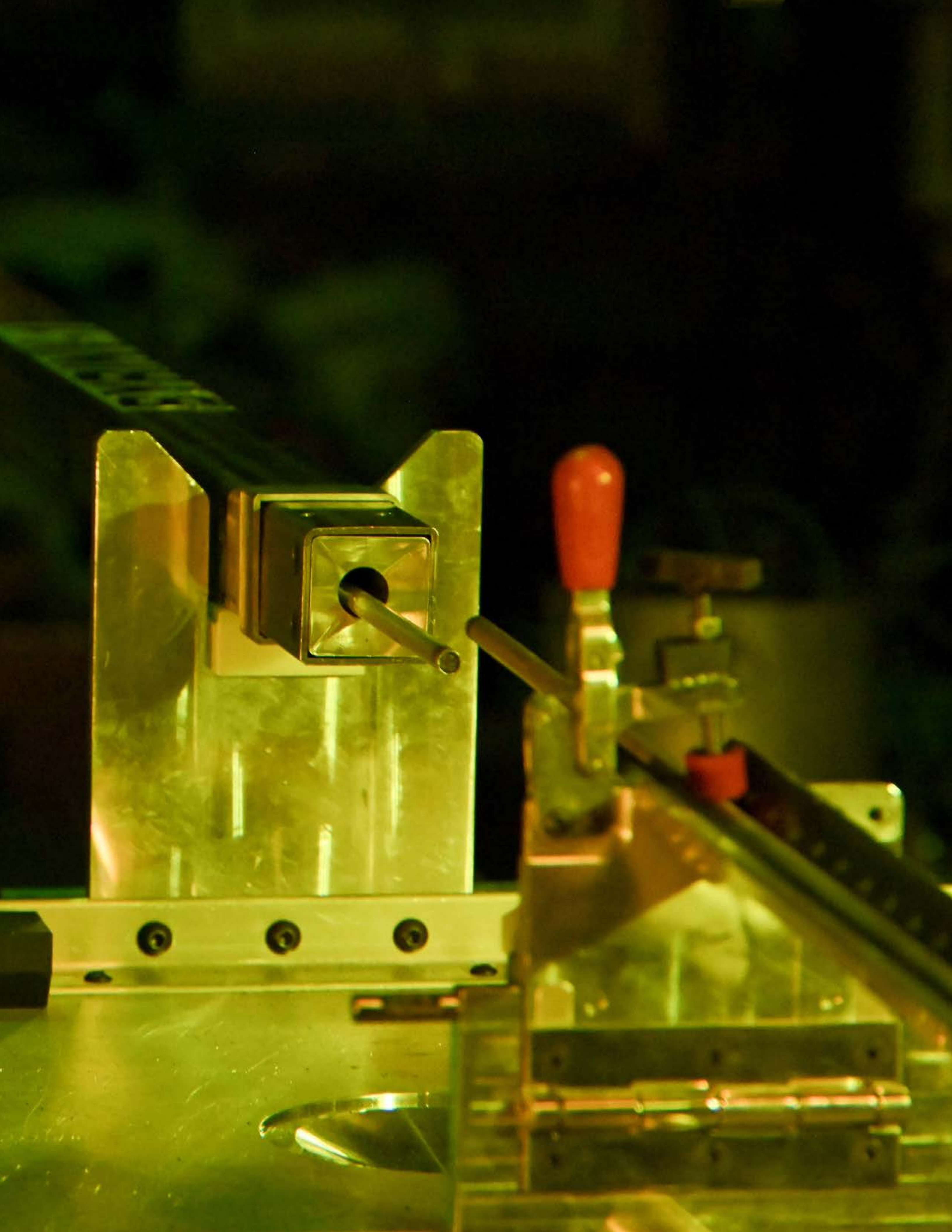




# ADVANCED FUELS CAMPAIGN
## 2024 Accomplishments

AFC Advanced Fuels Campaign



Fuel Cycle Technologies

# Advanced Fuels Campaign
# 2024 Accomplishments

INL/RPT-24-81719

November 22, 2024

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11-22-2024

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11-22-2024

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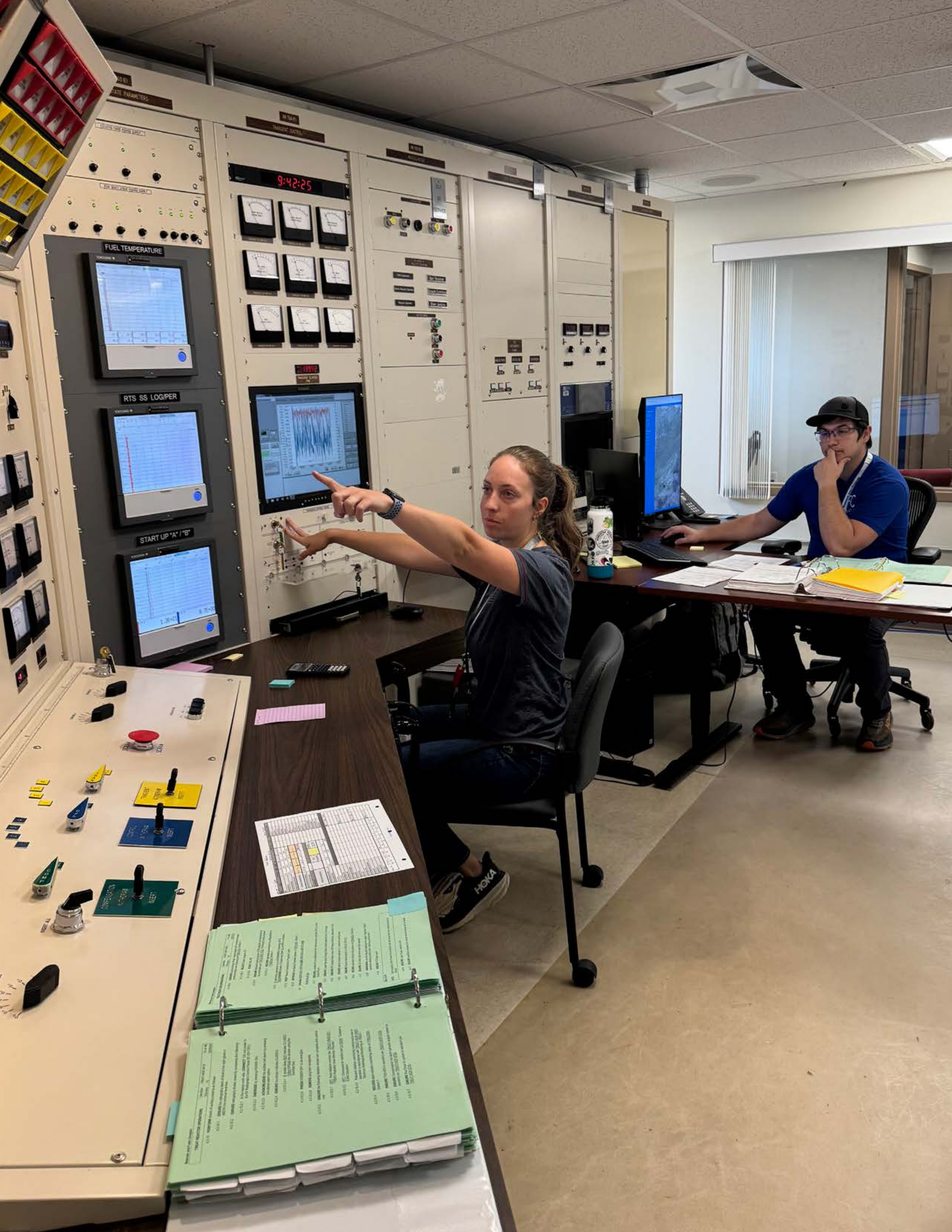

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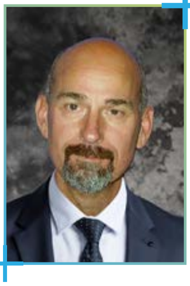

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# 1.2 FROM THE DIRECTOR

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Broad growth in energy demand across the globe, with a clear preference for 'clean' energy, has fueled focus on recent opportunities for nuclear energy expansion. A definitive signal of intent was sent in December 2023 at the COP28 meeting where 20 countries signed a declaration to triple nuclear energy production worldwide by 2050. This would mean adding an astronomical ~800 GW of nuclear power in just over 25 years. This serves as both a joyous affirmation and a sobering challenge to the relatively small community of nuclear technology experts already engaged in research, technology deployment, and regulatory activities. Rising to this moment will require foundational shifts in investment and behavior across the nuclear ecosystem.

It is very clear that succuss will require stakeholders to collectively take conscious action to realize this outcome. This includes governments, industrial end-users, investors, utilities, EPCs, and OEMs. Both domestically and internationally. In particular, governments will need to provide motivation for collective development and deployment of first-of-a-kind technologies. In parallel, they must drive establishment of new methodologies that accelerate the development and acceptance of innovative technologies. All while industry builds robust supply chains that open the aperture for rapid scaling of manufacturing for existing and new technologies.

All of this must happen simultaneously on two technology platforms, each with distinct stakeholders communities. The existing marketplace must respond quickly to satisfy increases in demand with near term energy production growth (though power uprates and new builds). This will require an emphasis on economic stabilization of the current fleet. In parallel, advanced technologies must be rapidly demonstrated and matured to complement existing technology and open new markets in the coming decades (including process heat, hydrogen, desalination, marine propulsion, etc.) and to enhance fuel utilization (increased efficiency, fissile recycle).

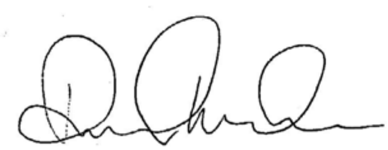
The nuclear fuel research and development community must play a leading role in this transformation to drive efficiencies in fissile utilization and the integrated nuclear fuel cycle. Already, demand for uranium ore/enriched uranium is on the rise as utilization growth collides with supply chains disrupted by geopolitical forces, responsible management of used fuel (including storage, recycle, and ultimate disposition) becomes a social priority, and the foundations for modern proliferation resistant technology and policy are being reformulated to match the evolving landscape. Fortunately, AFC is well-positioned to act on this opportunity.

Over the past decade, DOE-NE nuclear fuel development programs have led the way in enabling the technology that will fuel this revolution. The ATF program has demonstrated how to leverage stakeholder partnerships between research, industry, and regulators to achieve significant and practical outcomes. Including a transformation in attitude toward innovation that will be essential in the coming years. DOE-NE has also, through long term perseverance and commitment, pioneered the development of advanced fuels that will power the demonstration facilities expected to launch the next generation of reactors (e.g. TRISO and metal fuels).

However, this is not the conclusion to this difficult journey. In fact, the pace of innovation needs to drastically accelerate in order to meet these lofty goals. For the last 20 years DOE-NE has invested in the implementation of new tools that will accelerate bridging of science to engineering in nuclear fuels. Our next challenge is, as we complete the critical missions of today, to systematically apply these tools to realize rapid innovation in the immediate future. To rethink the way we integrate idea generation with application driven research. To seamlessly blend modeling and simulation with experimental results and project them both to utilization scenarios.

I'm very excited about where AFC is headed. The re-formulation of our mission into a two-branch program, ATF to provide near term support to the existing industry and Next Generation Fuels (NGF) to drive mature technologies to deployment while simultaneously pursuing new innovations, provides an ideal structure to respond systematically to this generational opportunity.

Absolutely a future to look forward to!



Dan Wachs

## TWIST Device and Rodlet Refabrication

*Principal Investigator: Klint Anderson (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Colby Jensen, Cindy Fife, Jason Schulthess, Charles Folsom, Robert Armstrong, Austin Fleming, Ashley Lambson, Geran Call, Chase Case, Connor Michelich, Mark Cole, Spencer Parker, Jordan Argyle, Jerry Kahn, Justin Yarrington, Clayton Turner, Andrew Chipman, Changhu Xing, Ryan Sandbeck, Sterling Morrill, Nicolas Woolstenhulme, David Kamerman, Daniel Wachs (All INL)*

***The deployed TWIST device and rodlet refabrication expertise developed by INL provides the U.S. with the capability to perform critical safety testing on HBu LWR fuels, necessary for advancing fuel technologies and extending the allowable fuel burnup limit of the current LWR fleet.***

The unexpected 2018 closure of the Halden Boiling Water Reactor (HBWR) in Norway eliminated the only in-reactor Loss of Coolant Accident (LOCA) testing capability available in the western world. This testing capability, which allows research on Fuel Fragmentation Relocation and Dispersal of high burnup (HBu) fuel during a LOCA, is critical to support ongoing efforts to extend the allowable fuel burnup limit in the United States fleet of light water reactors (LWR) beyond 62 GWd/t. To eliminate this and other capability gaps resulting from the HBWR closure, namely the ability to refabricate and instrument previously irradiated fuel rod segments for use in research and development, a high-performance team across various INL facilities developed a method and system for irradiating refabricated HBu fuel in the Transient Reactor Test Facility (TREAT).

### Project Description

Born out of necessity to develop a LOCA capability, the Transient Water Irradiation System in TREAT (TWIST) is an enhanced capability device designed to serve as the HBu test vehicle for all capsule type accident testing of LWR fuels in TREAT, including Reactivity-Initiated Accident (RIA) and Accident Tolerant Fuel (ATF) safety research testing. From original concept to commissioning, the TWIST device is five years in the making and has required significant investment to achieve, including the reconfiguration of the TREAT reactor core to provide a large experiment capability. By itself, the core reconfiguration is a remarkable undertaking and achievement.

The TWIST device consists of a static water filled capsule which provides a controlled environment around the fuel specimen. As shown in Figure 1, the device can be configured for LOCA testing by attaching a blowdown valve and expansion tank below the capsule, or for RIA testing by plugging the bottom port on the capsule and attaching a flux filter to the capsule. The design of the TWIST device, coupled with the unique capability of TREAT, allows highly representative testing using nuclear heating, with achievable rod powers and temperatures prototypic of LWR accident scenarios. The TWIST capsule accommodates rodlets with fueled lengths ranging from 25 to 50 cm and features a state-of-the-art instrumentation package to collect relevant time-dependent data for post-test analysis. Instrumentation is specifically designed and tailored to detect

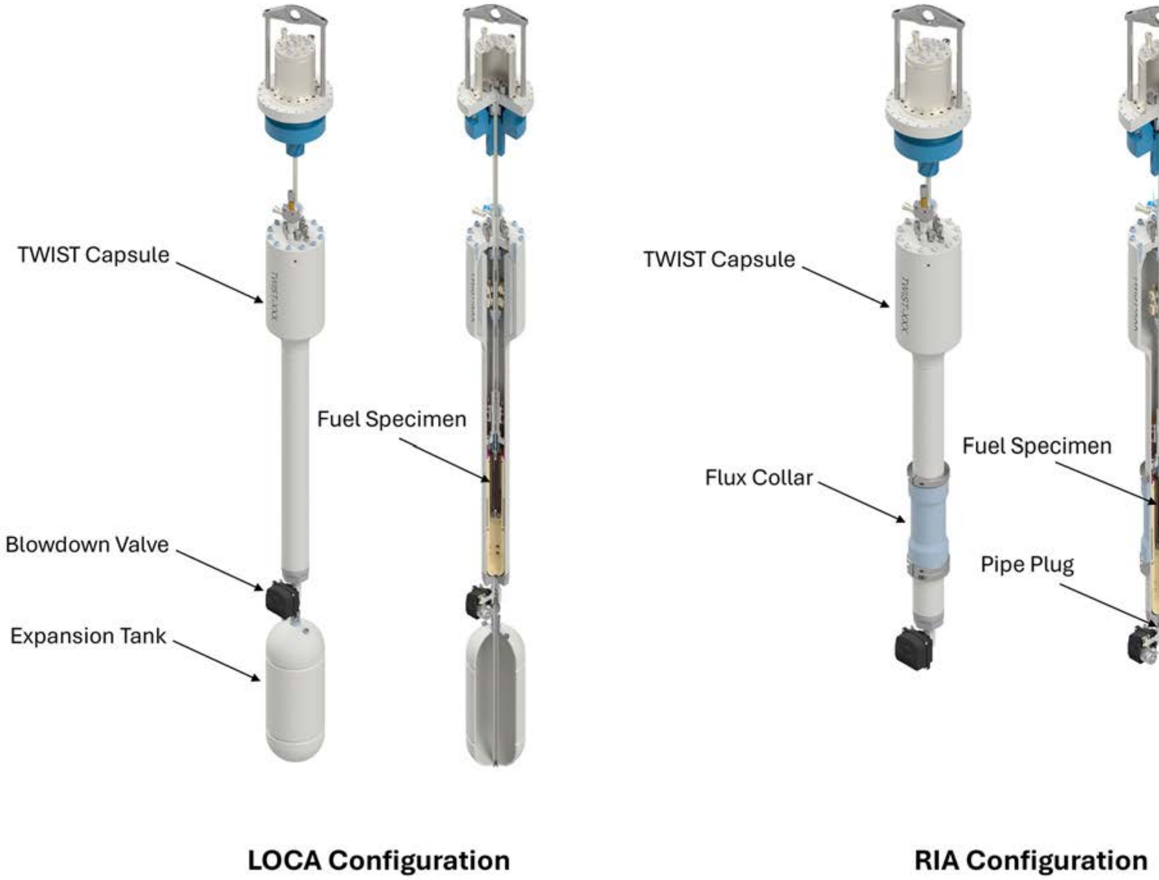



*Figure 1. TWIST device configurations for LOCA and RIA testing.*

phenomena associated with LWR accident scenarios including an electroimpedance sensor to detect phase change events in the water and cladding radial distension, 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 rodlet fuel fission gas release or cladding elongation of refabricated rodlets from HBu fuel rod segments.

### Accomplishments

While designed for remote assembly inside the Hot Fuel Examination Facility (HFEF), the TWIST device was commissioned with fresh fuel specimens prior to experiments with HBu fuel. A commissioning plan was developed which includes a series of tests to establish, demonstrate, and qualify the complete experimental system for LOCA and RIA testing in TREAT. A total of ten irradiations on the first three TWIST devices have been completed to characterize fuel

Figure 2. Experiment team with TWIST device and Institut de Radioprotection et de Sûreté Nucléaire collaborators.

power coupling with reactor power, thermal-hydraulic conditions, and instrumentation performance. The commissioning experiments have provided the experiment team with results, documented in more detail elsewhere in this report, which have been integral in characterizing the device and aligning supporting analyses. Fabricating, assembling, installing, and irradiating these TWIST devices (see Figure 2) have provided the experiment team, located at multiple facilities across the INL, with knowledge and expertise required to ensure successful completion of more complicated HBu tests. A functionality test of the blowdown valve, completed after the second transient in TREAT, provided valuable information on the effects of neutron dose on the valve's electronics. This knowledge would have been difficult to obtain elsewhere. Modifications were quickly made by the experiment team to remove the determined sensitive components from the reactor core, developing a gamma and neutron radiation resistant valve actuator which will be used on all future LOCA tests. The following transient was completed shortly after with successful valve actuation and blowdown in the device.

The design of the TWIST device was updated to simplify the specimen loading process of HBu fuel segments while protecting sensitive instrumentation required to be in close proximity with the test specimen. This design update, a collaboration between experiment design, instrumentation experts at the Engineering Innovation Laboratory (EIL) at INL, and operations and engineering at HFEF, leveraged knowledge and expertise gained through the remote assembly of previous experiment campaigns. Remote handling equipment was designed and fabricated to facilitate the TWIST device, and the first two phases of remote handling qualification were successfully completed, proving the updated design can be assembled remotely in HFEF (see Figure 3).

Figure 3. Remote handling qualification of the TWIST design for HBu fuel specimens.

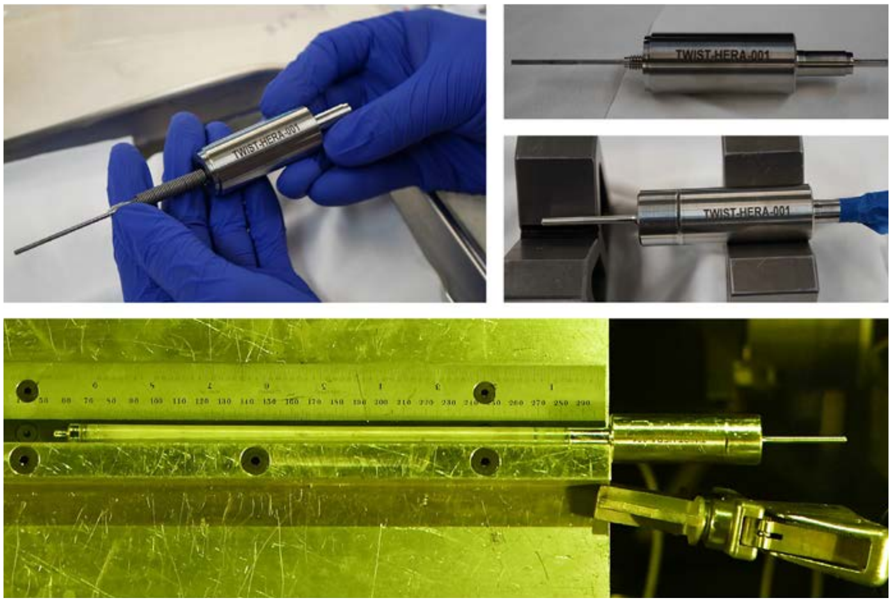



*Figure 4. Instrumented end caps and complete refabricated HBu rodlet.*

Finally, HBu fuel shipped to INL from the Byron Nuclear Generating Station was selected and successfully refabricated into a fuel specimen for use in the TWIST device. This refabrication effort, documented in more detail elsewhere in this report, leveraged previously developed capabilities and expertise at HFEF to segment, defuel, and weld a sectioned HBu rod to end caps specifically designed to incorporate within TWIST. The rodlet and its end caps, as shown in Figure 4, contain components which integrate into and are calibrated with an electromagnetic transducer, forming a rodlet fuel fission gas release sensor. This challenging refabrication effort and its associated learning opportunities provided the experiment team with even more expertise which will be leveraged to increase refabrication process capabilities and further eliminate gaps caused by the HBWR closure.

Looking ahead, the TWIST device is well poised to capitalize on the impactful accomplishments of 2024. The first refabricated rodlet will be irradiated as part of the High-burnup Experiments in Reactivity Initiated Accidents test program in TWIST, which will evaluate pellet-cladding mechanical interaction of HBu fuel. Several more HBu rodlets will be refabricated with planned RIA and LOCA transients executed to support the ongoing fuel burnup extension efforts. The team is also implementing design improvements, including international collaboration to add steam capability to the device, to further the capability of the TWIST device to support the current and growing needs of the LWR community.

# Assessment of Readiness to Perform Digital Image Correlation Strain Mapping Experiments on the In-cell Severe Accident Test Station

*Principal Investigator: Peter Doyle (Oak Ridge National Laboratory [ORNL])*
*Team Members/Collaborators: Mackenzie Ridely, Samuel Bell, Jason Harp, Nathan Capps, Brandon Johnson, Adam Willoughby, Matt White (All ORNL)*

Milestone report M3FT-24OR020204041 discusses preparations to upgrade the data collection capabilities on the Severe Accident Test Station (SATS) by addition of the digital image correlation (DIC) technique for in-situ strain mapping. A 12 lamp furnace, capable of inducing high heating rates >100°C/s, was modified to install a viewport, prepare it for the hot cell, and reduce the number of lamps so as to match the power limits in the hot cell. Additional hot-cell specific equipment to enable DIC measurements was then built and tested. A specialized rig was developed to illuminate the samples and collect images in-situ. Specialized tools were also built for in-cell 3D DIC calibration and speckle painting of fuel rod samples in-cell.

***Upgrading the capabilities of the Severe Accident Test Station to increase heating rate and provide capability to conduct in-situ strain measurements will permit higher quality accident testing data and enable more robust modeling of accidents in support of industry licensing objectives.***

## Project Description

Evaluation of the response of nuclear fuel and cladding to loss-of-coolant accidents (LOCAs) is critical to the licensing of novel cladding concepts as well as licensing conventional $UO_2$ fuels to high burnup. At ORNL, the SATS is used for LOCA testing. Conventional LOCA testing has involved conducting the test and then using calipers or other tools to assess total strain. In recent years, the out-of-cell version of SATS has also been equipped with the ability to collect strain maps in-situ using DIC on unfueled unirradiated rods. The governing objective of this work was to prepare and characterize upgrades to the in-cell version of the SATS to enable DIC on irradiated nuclear fuel and cladding.

In support of the governing objective, this work purposed to 1) modify a 12-lamp furnace, 2) construct a camera-deployment system optimized for hot cell application, 3) develop a system with which to paint fuel rods with the necessary speckle pattern in cell, and 4) test the modified furnace to assess heat rate functionality. The 12-lamp furnace modification involved creation of a viewport through which to image the sample for DIC,

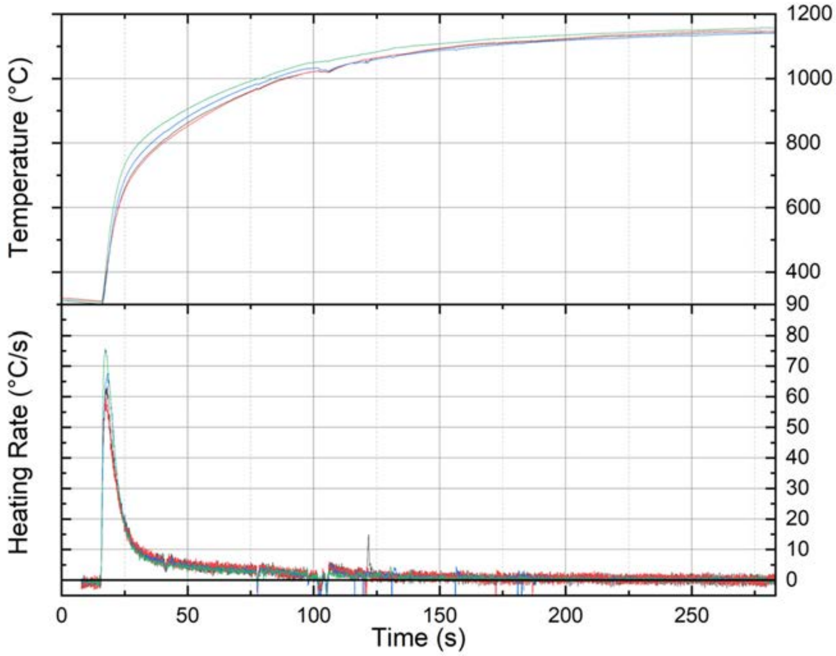



*Figure 1. Temperature (°C) and instantaneous heating rate (°C/s) as a function of test time, starting at the end of the 300 °C hold period and 100% power heating test using the modified SATS 2 DIC furnace.*

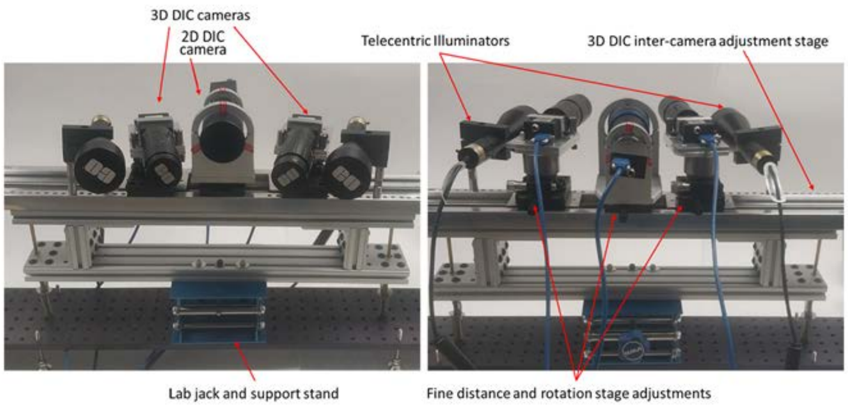



*Figure 2. DIC data collection and sample illumination system.*

rework the electrical and water coolant lines for hot-cell deployment, and adapt the lamp configuration to the available hot-cell power. Camera-deployment system and fuel rod painting system development involved adapting the currently existing DIC system to facilitate positioning of cameras at precise working distances and stereo-angles, replacement of cameras, which fail within two weeks in the hot cell, and conservation of hot cell space. Testing of the modified furnace required the use of the out-of-cell SATS at controlled target ramp rates of 5, 10, and 15°C/s and a maximum power test assessing temperature uniformity across the sample and achievable ramp rates.

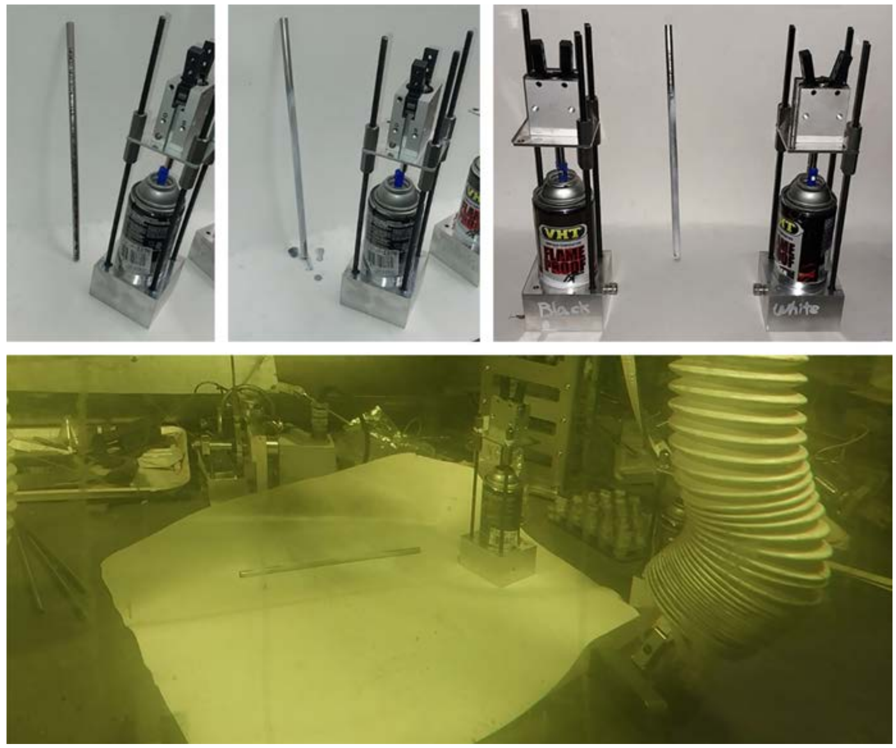



*Figure 3. Demonstration of DIC rod painting with aerosol spray painter system for in-cell application.*

**Accomplishments**

Existing technology for using DIC in out-of-cell environments has been translated to be ready for the hot cell environment. Additionally, a 12-lamp furnace has been modified to prepare it for DIC testing in the hot cell under existing power capacity with thought to maximum thermal uniformity during testing and two existing 4-lamp furnaces for SATS 1 have been examined at high power. Standup testing of the modified furnace was done with a surrogate fuel rod to assess the new maximum heating rates and to demonstrate the use of DIC on the modified furnace. Tests demonstrated excellent uniformity regardless of heating condition.

The furnace successfully performed up to 1000 °C with a ramp rate of 5 °C/s. At 10 °C/s, the ramp rate was uniform up to around 850 °C, at which point it began to decrease with increasing temperature. At 15 °C/s, the ramp rate was uniform up to around 600°C before degrading with increasing temperature. The maximum possible ramp rate was assessed with a test where the furnace was provided maximum power throughout the test duration, Figure 1. Under 600 °C, ramp rates were controllable to 33 °C/s. However, as the temperature increased above 600 °C, the ramp rate continuously decreased. Maximum power testing of the standard the SATS 1 furnace for non-DIC testing found result higher controllable heating rates of 40 °C/s up to 700 °C with further degradation of the heating rate above 700 °C. Finally, a maximum power DIC test at 13 MPa was conducted to burst on the SATS 1 DIC furnace. While ramp rates with this furnace were well in excess of 50 °C/s, the thermal asymmetry of the rod during ramp reached up to 120 °C by the burst. Additionally, capabilities for in-cell DIC testing at ORNL have been developed (Figures 2 and 3).

Work since the milestone completion has progressed and is ongoing in two areas. First, the controller system for the out-of-cell SATS 1 unit has been upgraded to take advantage of the full power range of the SATS 1 LOCA furnace and the power capability of the SATS 1 controller system has been increased. Second, in-cell furnace selection is underway. To accommodate this, the modified furnace used in this work was upgraded from 6 to 10 lamps and re-tested to evaluate improvement in the heating rate at the enhanced power capacity. At the higher current and with additional lamps, the 10-lamp furnace achieved higher linear heating rates to higher temperatures. Additional options for customized furnace designs are also being explored. Finally, a plan has been developed to upgrade power infrastructure within the hot cell. On completion such an effort will permit deployment of either a DIC or high ramp rate furnace in the cell.

# New Approach for Leadout Testing in ATR (IMPACT)

*Principal Investigator: Calvin Downey (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Chris Murdock, Nate Oldham, Jason Brookman, Matthew Mihelish, Kyle Gagnon, Kort Bowman, Austin Fleming (All INL)*

***The IMPACT experiment will be the first in a series of instrumented capsule experiments in ATR to irradiate various advanced nuclear fuel specimens and materials with embedded thermal conductivity probes. This design will accelerate irradiation experiment data collection by utilizing novel in-situ instrumentation, eliminating the need for a considerable number of drop-in capsules with discrete burnup or displacements per atom (DPA) targets.***

The Irradiated Material Properties Accelerated Characterization Test (IMPACT) experiment is a new form of instrumented capsule experiment which allows for instrumentation and in-situ data collection during irradiation. These new experiment designs utilize the recent installation of the Advanced Test Reactor (ATR) Top Head Closure Plate–Mark II (THCP MkII). Within the new THCP MkII, vessel penetrations out of ATR have been increased to allow for leadouts from both I positions and Small B positions, effectively doubling the experiment penetration capacity through the top of ATR. The IMPACT experiment will be the first experiment in ATR to utilize these new capabilities.

## Project Description

The IMPACT experiment will be the first in an anticipated series of experiments to irradiate advanced nuclear metallic fuel specimens with novel embedded thermal conductivity probes in ATR. This new capability will provide direct access to parameters more commonly inferred through post-irradiation measurements and modeling and minimize specimen throughput (by reducing physical testing quantities) by extracting more specimen information during irradiation testing. Additionally, the evolution of metallic fuel thermal conductivity during this early phase burnup (and over the course of burnup) has never been successfully measured in-situ in pile. Post-irradiation examination measurements made by Bauer gave a measurement uncertainty of ±20% for nine discrete local burnup values (ranging between 0.5–1.9 at% burnup). The embedded conductivity probe capability allows for continuous measurements rather than discrete points, and previous laboratory testing has demonstrated the technique is capable of thermal property measurement with uncertainty better than 10% on a range of sample materials. For the first IMPACT experiment the thermal conductivity of uranium zirconium (U10Zr) will be measured for multiple fuel geometries (ex. slotted cross-section design) for a fuel burnup of 7.5% and the change in thermal conductivity over this burnup range will be characterized.

## Accomplishments

Final design reviews, design verification, and design closeout were completed for the IMPACT-01 experiment. The IMPACT-01 design features three double encapsulated rodlets and a stainless-steel reference capsule in the in-core test train. Three fuel designs are planned for irradiation as the first iteration of the IMPACT design: annular, slotted, and sodium bonded shown in Figure 1. New leadouts have been designed to reach out of the new ATR THCP MKII I-penetration from ATR Small-B positions, shown in Figure 2. Novel thermal conductivity probes are sealed into each

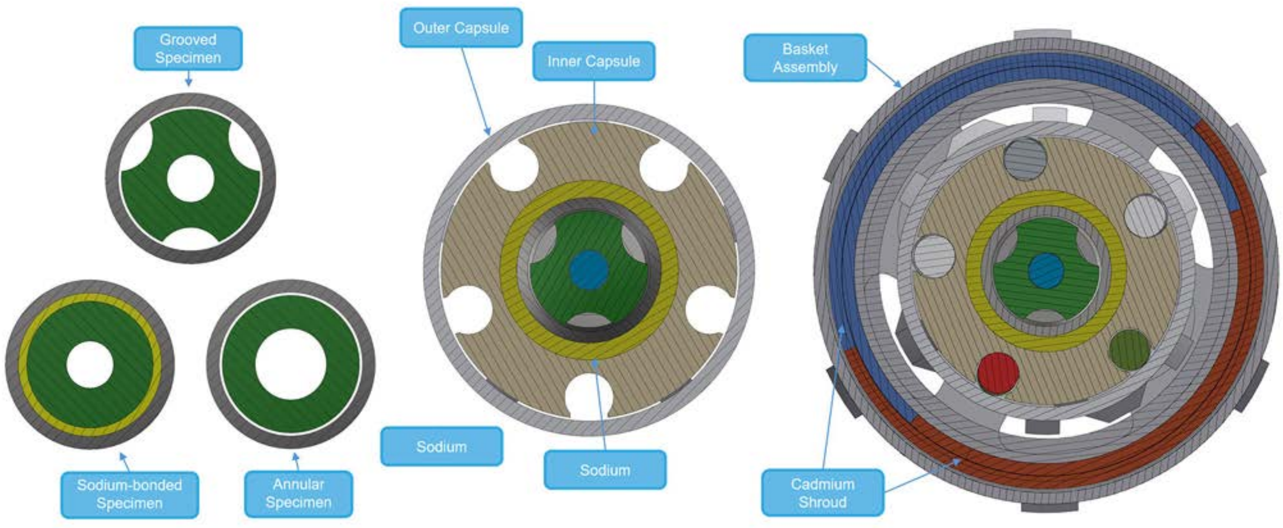



*Figure 1. Cross-section views of a) fuel rodlets, b) inner and outer capsules, and c) installation of experiment in a cadmium shrouded basket.*

fuel rodlet and are transitioned to soft cable to reach the top of the reactor head within the leadout tube and are assembled into micro electrical connectors at the top of the experiment within the ATR shield cylinder. Existing Accident Tolerant Fuels cabling at the top of the reactor head connect to the sensor leads to provide power and receive signal into a control cabinet in the 1A basement cubicle. Control systems will provide power to the heated sensors and record in-situ changes in temperature within the fuel. Prior to experiment final design control cabinets were fabricated and out-of-pile testing was completed for the thermal conductivity sensor.

The experiment fabrication and assembly is currently underway for the IMPACT experiment. Autogenous welding and high temperature brazing procedures are currently being developed at the Test Train Assembly Facility (TTAF) at the Test Reactor Area and at the Advanced Fuels Fabrication Facility at the Materials and Fuels Complex for the IMPACT assembly. While previous brazing efforts have been performed for design validation, new brazing capabilities at TTAF will be used for the IMPACT instrumentation brazing. All in core components including rodlet, inner capsule, outer capsule and fuel rodlets completed fabrication as shown in Figure 3. Design is completed for experiment lifting tooling, operator support tooling, out of pile pressure testing systems, and the ATR lateral support arms for the IMPACT experiment. Fabrication of these items is slated to begin by the beginning of fiscal year (FY) 2025 and completion of all fabricated items is planned for March FY 2025.

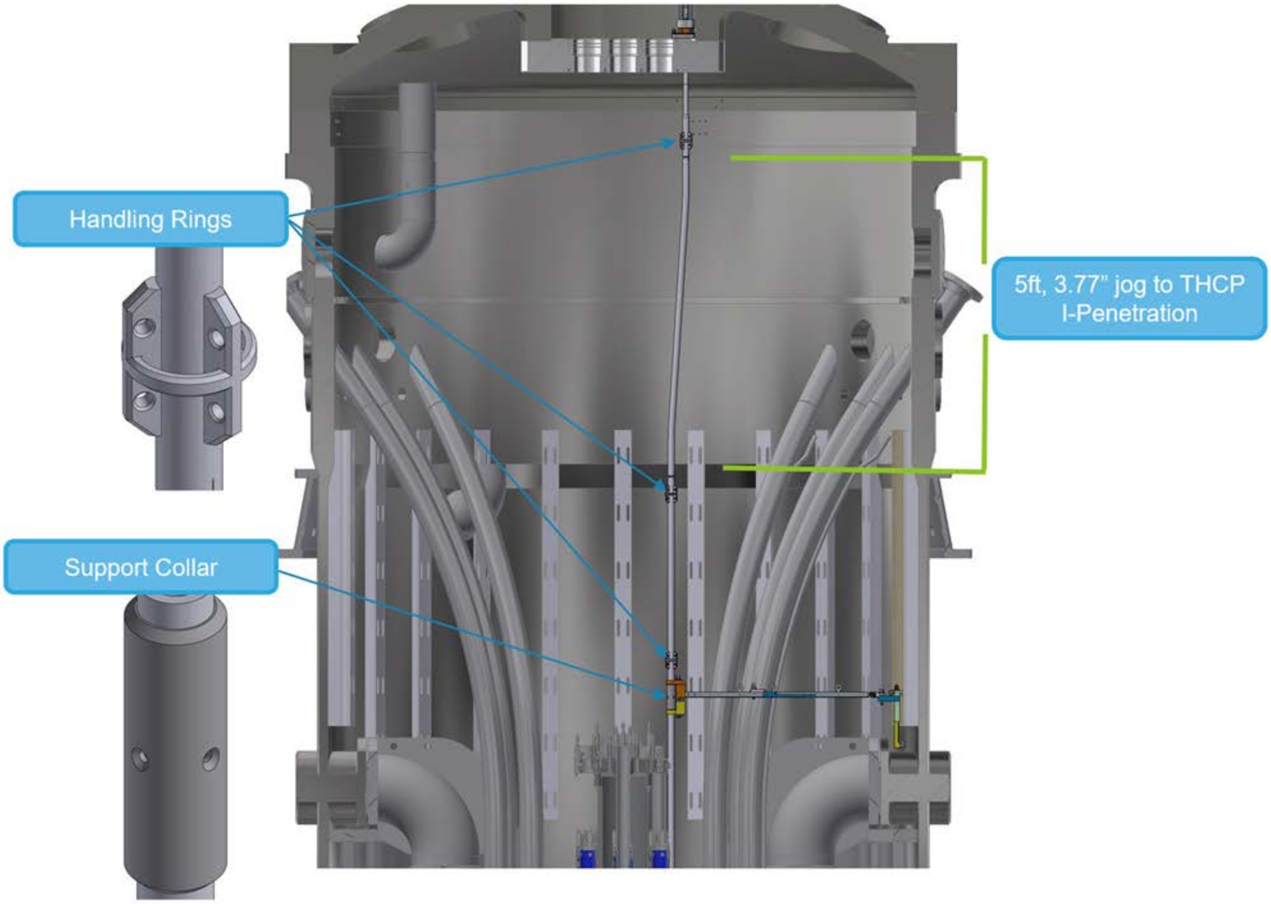
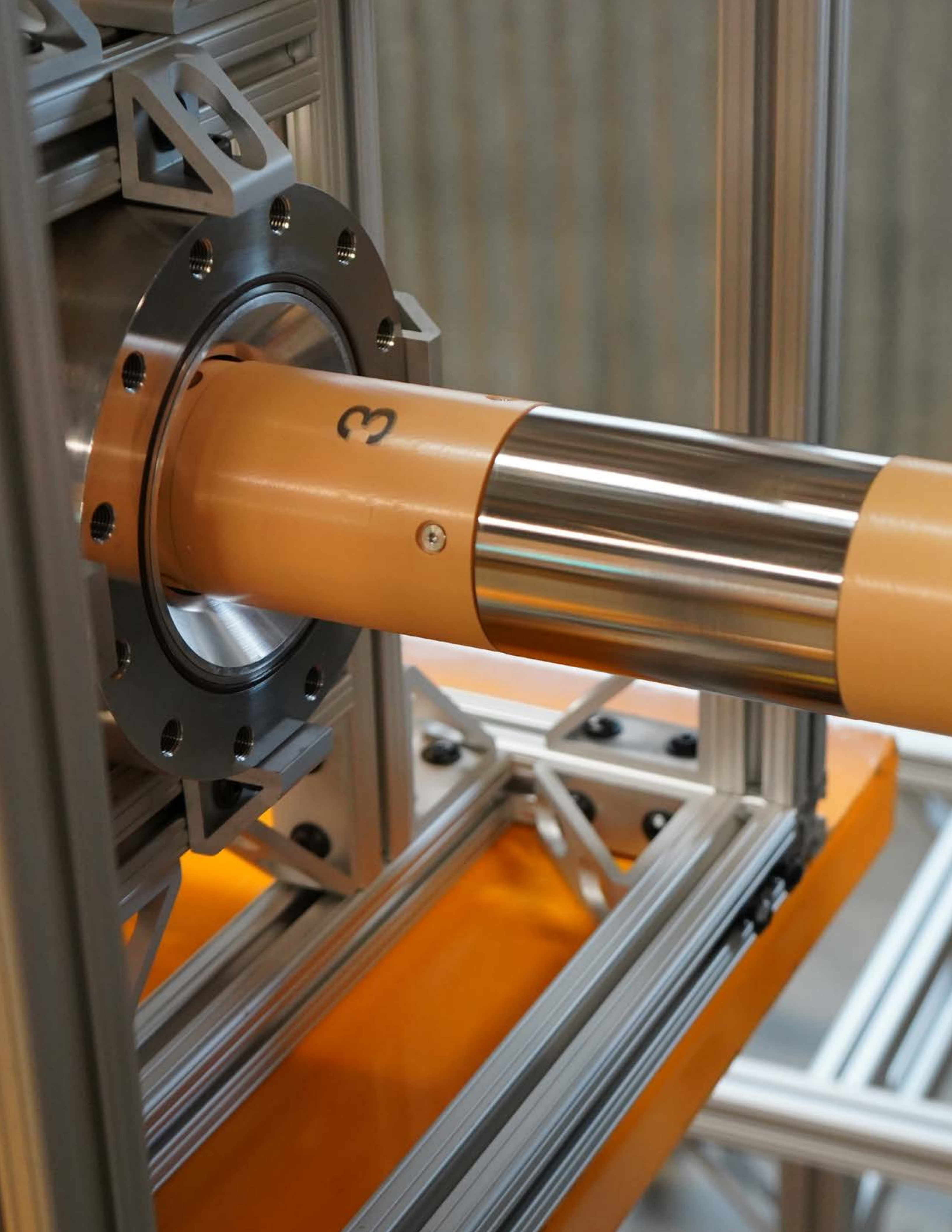



*Figure 2. Break-out view of the IMPACT experiment leadout section installed in ATR. THCP-MkII seal hardware seals the experiment through the top of the reactor, handling rings are used for installation, and the support collar interfaces with the lateral support arm.*


*Figure 3. Completed IMPACT capsule hardware for the sodium bonded fuel capsule. Note that the fuel and sodium guard pictured is fabricated from SS316 for prototyping and assembly mockup.*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## 1.4 FY-24 MEETING INTEGRATION AND INTERNATIONAL COLLABORATIONS

### Campaign Integration Activities and International Collaborations

*Phyllis King (Idaho National Laboratory [INL])*

In fiscal year (FY) 2024, the Advanced Fuels Campaign (AFC) proactively engaged in a comprehensive suite of integration activities, including a series of strategic domestic and international meetings, aimed at fostering collaborative relationships and knowledge exchange in nuclear fuel technologies. These forums have been instrumental in aligning AFC's research goals, disseminating recent advancements, and laying the groundwork for future partnerships in the global nuclear energy landscape. Key activities throughout the year included strategic and technical meetings with the AFC team, collaborative research discussions on advanced fuel technologies, planning meetings for upcoming fiscal years, workshops on metal fuel technology, and various international collaborations that provided forums for exchanging insights on the latest developments in nuclear fuels.

Campaign integration highlights included the AFC Annual Meeting in Texas, the meeting of AFC leads at Oak Ridge National Laboratory (ORNL), the Electric Power Research Institute (EPRI) Collaborative Research on Advanced Fuel Technologies (CRAFT) meeting focusing on accident tolerant fuels (ATF) and high burnup (HBU), and the virtual planning sessions for FY 2025. The Metal Fuel Workshop attracted diverse stakeholders, while the post-irradiation examination (PIE) and transient testing strategies for Byron fuel rods were reviewed. Discussions at EPRI's power uprates workshop outlined future strategic directions for light water reactors (LWR), and budget planning discussions were held online to refine AFC work packages.

International collaborations were a cornerstone of AFC's activities in FY 2024, with meetings fostering deeper, more impactful collaborations, participation, engagements, and bilateral meetings. These interactions ranged from sharing technical expertise on ATF cladding to enhancing international partnerships in nuclear fuel and materials research.

The AFC team also presented at numerous conferences and workshops, including TopFuel 2024, sharing insights into HBU PIE data, progress on ATF rods examinations, the effects of hydrogen on cladding performance, and advancements in nuclear fuel designs and irradiation testing technology. These contributions to international forums underscore AFC's commitment to advancing the state-of-the-art in nuclear fuel management and fostering global collaboration in nuclear research and development.

*Figure 1. Attendees from the AFC Annual Meeting held at Texas A&M.*

**Campaign Integration**

AFC Campaign Integration refers to the processes and activities undertaken by AFC to ensure that its research and development (R&D) efforts in nuclear fuel technologies are well-coordinated, both internally and with external partners. This involves aligning research goals, sharing advancements, and fostering collaborative relationships to drive innovation in the nuclear energy sector. Campaign integration is a key facet of the AFC's strategy to advance nuclear fuel technologies and ensure that research efforts are coherent, collaborative, and directed towards meeting the current and future needs of the nuclear energy industry.

The Annual Meeting for AFC for FY 2023 took place at Texas A&M University (TAMU) in College Station, Texas, from November 7th to 9th, 2023. Attendees included representatives from the Department of Energy Headquarters (DOE-HQ), the Department of Energy Idaho (DOE-ID), various national laboratories, industry partners, and academic institutions, who presented their achievements from the campaign over the FY (See Figure 1). Additionally, participants had the opportunity to tour TAMU laboratories, which included the Fuel Cycle & Materials Laboratory, the Accelerator Laboratory, and the Thermal-Hydraulic Research Laboratory.

The AFC leads convened for a meeting from February 27th to 29th at ORNL. The gathering saw participation from over 30 AFC Leads, including those from the tristructural isotropic (TRISO) program, who engaged in technical discussions on potential program outcomes that could influence the strategic plan update for the years 2025 through 2029. They also provided summaries of critical programs that could impact the campaign and shared a mid-year technical update on the activities planned for 2024. Attendees were offered tours of several facilities including the hot cell, the Severe Accident Testing Station, advanced characterization labs, TRISO fuel fabrication facilities, and the High Flux Isotope Reactor (HFIR). In addition to these, a visit to the Ultra Safe Nuclear Corporation, which operates a pilot-scale commercial plant for TRISO fuel production, was also conducted.

The second quarter meeting for 2024 of the EPRI CRAFT for LWRs took place in April. This event served as an extension of the yearly workshops jointly held by EPRI, DOE, and Idaho National Laboratory (INL), focusing on ATF and HBU discussions. The CRAFT initiative facilitated this meeting, which examined significant updates on advanced fuel technology programs from various countries and organizations, including comprehensive technical discussions on ATF concepts and the effects of increased burnup on LWR fuel.

The first day of the meeting was dedicated to global stakeholders sharing updates on ATF programs. The second day shifted focus to HBU issues with the Fuel Fragmentation, Relocation, and Dispersal (FFRD) Technical Experts Group (TEG) meeting, addressing related technical subjects. On the third day, the Time-at-Temperature (t@T) TEG convened to discuss advancements in materials testing plan development.

The CRAFT meeting serves as a platform to highlight ongoing major development programs and research activities in the field of advanced fuel technologies across the globe. These include various multilateral projects, some of which are managed or sponsored by national governments, commercial vendors, and international bodies. The meeting provided attendees with the opportunity to gain insights into these initiatives and explore potential collaborative endeavors. The TEG sessions also facilitated discussions on current action items and technical matters related to FFRD, as well as t@T considerations.

The AFC leads attended a virtual planning meeting in April, with the goal of preparing the planning packages for FY 2025. Additional discussions were had to discuss the cross-cutting scope of AFC's work with that of partner programs, including the Nuclear Energy Advanced Modeling and Simulation (NEAMS) Program, Advanced Reactor Technologies Program, Advanced Sensors and Instrumentation Program, and the Advanced Reactor Demonstration Program.


*Figure 2. Participants from the Second Metal Fuel Workshop held in Idaho Falls, ID.*

AFC hosted the second Metal Fuel Workshop in May, bringing together a diverse group of stakeholders encompassing industry representatives, various DOE programs, national laboratories, actively engaged university partners, and international associates (See Figure 2). This workshop saw contributions from DOE-affiliated labs, DOE-ID, DOE-HQ, companies specializing in reactor design and fuel production, the Nuclear Regulatory Commission (NRC), and DOE programs such as NEAMS, Fuel Cycle R&D under the Fuel Recycling Program, and Material Recovery and Waste Form Development. The agenda was arranged over three days, with the first two days devoted to presentations and discussions involving both AFC and its stakeholders and collaborators. The final day was reserved for guided tours and specialized presentations with an emphasis on the intricacies of fuel fabrication.

Key conclusions expand upon the foundational takeaways from 2023:

- Metallic fuel technology has a broad set of stakeholders interested in collaboration to enable deployment and long-term success. While Sodium Fast Reactor (SFR) applications dominate, stakeholders also include interests in LWR applications. The second annual workshop was well-received and encouraged to continue annually.

- The AFC program has developed a first revision for an R&D plan and is soliciting review and input from stakeholders to ensure federal investments are as impactful and far-reaching as possible. The plan builds off planning stemming from 2020 with the formation of the Leading innovation in Fuel Technologies (LIFT) proposal and addresses conclusions from last year's workshop. Priority work is towards:
  - Establishing a reference design basis through NRC-reviewed reports and the Experimental Breeder Reactor-II Fuels Irradiation & Physics Database remains a crosscutting priority for all stakeholders to lay foundation for first deployments and next innovations. These activities are progressing.
  - Data gaps identification and addressal using precious and limited legacy metallic fuel materials preserved at INL. The DOE program has started work in most identified data gap areas. A current capability/data gap is related to furnace testing for medium-slow transient testing.
  - The AFC program leadership will share the R&D plan and follow up with individual stakeholder meetings about feedback and interests.
  - Next-generation fuel design in the AFC program still plans to focus on fabrication technology development and improved high temperature strength cladding alloys. An interactive survey was conducted with workshop participants to capture general input on fuel design options to kick off a detailed engineering evaluation to be conducted by the AFC program.
  - Reactor designers are aggressively pursuing fuel qualification efforts for their startup fuel designs. Several designers are also interested in next generation fuel designs including goals of higher burnups and fuel recycle designs.
  - Broad interest (DOE to industry) in transuranic (TRU) fuels is resurging in the U.S. due to economic and waste management reasons. International interests in Japan and Korea continue strong in this area.
  - International interest in metallic fuels is also increasing. Representatives from the Japan Atomic Energy Agency (JAEA) and the Korean Atomic Energy Research Institute (KAERI) presented strong R&D programs emphasizing the importance of metallic fuels in their respective national fast reactor strategies. Japan will make a national decision in 2026 to select either oxide or metallic fuel after it completes an ongoing evaluation.

Representatives from Westinghouse Electric Company, ORNL, and INL convened to review the progress and future strategies for PIE and transient testing of Byron fuel rods at INL and ORNL. Discussions also covered upcoming commercial deliveries, specifically the Byron Third Cycle. Additionally, tours of facilities related to the R&D efforts involving AFC were conducted.

AFC participated in the EPRI workshop on power uprates. Industry efforts are currently divided into two phases of power uprates. The first phase, known as Margin Uncertainty Reduction (MUR), results in a modest power boost of under 2% in total megawatt electric (MWe). The second phase's goal is to achieve a more significant increase in plant output, ranging from 5-15%, which will require major plant upgrades with associated costs running into the hundreds of millions. These uprate phases do not depend on changes to the Specified Acceptable Fuel Design Limits (SAFDL); instead, the purchase of additional fuel is adequate to even out power distribution and maintain current SAFDLs. Following the attainment of this uprate level, the strategy involves a reduction in fuel batch size, which will be facilitated by incorporating fuel at temperature (t@T) considerations and implementing SAFDL adjustments.

AFC stands at a decisive point, shaping the long-term strategic direction for LWRs. Crafting a detailed plan to support this ambitious objective is essential. We already have many of the necessary components for success; however, as the complete strategy unfolds, the necessity for strategic coordination and the vigorous engagement of AFC becomes increasingly evident to ensure the realization of our goals.

The budget planning meeting for the AFC leads took place online from June 25 to June 27. During this virtual meeting, staff members reexamined and revised the initial Campaign work package and milestones that were first put forward in April. The updated information from this session was presented at DOE Office of Nuclear Energy (NE)-4 Planning Package Meeting, which was held in Washington, D.C., in August.

#### International Collaborations

The collaborative research meeting between the French Institut de radioprotection et de sûreté nucléaire (IRSN) and INL took place from October 9 to October 13, 2023. This annual event was hosted by INL and saw the participation of eight IRSN delegates (see Figure 3) and over 20 INL researchers engaging in multiple sessions.

The agenda featured five technical sessions categorized by themes, which included Reactivity-Initiated Accidents (RIA), Loss-of-Coolant Accidents (LOCA), Reactor Physics, Severe Accidents, and Instrumentation. In addition, attendees were taken on tours of the Advanced Test Reactor (ATR), the Transient Reactor Test (TREAT) Facility, the Hot Fuel Examination Facility (HFEF), the Irradiated Materials Characterization Laboratory (IMCL), and INL facilities dedicated to measurement science and thermal hydraulics. At the start of the meeting, it was unanimously agreed upon that a primary and immediate goal of the IRSN-INL collaboration should be to foster a deeper and more impactful direct collaboration.

Josh White represented the U.S. as the technical authority on ceramic fuel properties during meetings in Paris, France, on October 19-20, 2023, as a part of the Expert Group on Innovative Fuel Elements (EGIFE). This group operates under the Organisation for Economic Co-operation and Development (OECD)/Nuclear Energy Agency (NEA) Nuclear Science Committee's (NSC) Working Group on Scientific Issues of Advanced Fuel Cycles. The EGIFE's meeting focused on reviewing and incorporating feedback from external reviewers on the preliminary report on mixed oxide fuel (MOX) and metallic fuel properties, feedback that has been collected since March FY 2023. The team's efforts were directed at refining the Properties Report and contemplating the extension of their research to include more physical properties and types of nuclear fuel.

The EGIFE brings together international experts to enhance the understanding of properties and performance of state-of-the-art nuclear fuels. The U.S. nuclear fuels programs benefit significantly from participating in this group, gaining access to comprehensive insights about the properties and performance of advanced fuels, insights that surpass the scope of data traditionally produced by U.S. research avenues. Engaging with the EGIFE serves as a key strategic tool to augment the efficacy of U.S. investments in fuel development for Advanced Fuel Cycles. This is particularly crucial because the current U.S. funding for advanced reactor fuels is relatively limited in comparison to past funding allocations.


*Figure 3. Visitors from IRSN touring TREAT (October 2023).*

Sven Vogel visited Tokyo, Japan, to participate in the Neutron Wavelength (NEUWAVE) conference, a leading event focused on energy and wavelength-specific neutron imaging—a field in which techniques are honed at the Los Alamos Neutron Science Center (LANSCE) for the evaluation and characterization of both fresh and irradiated nuclear fuels, along with other nuclear materials pertinent to the nuclear energy mission, such as molten salts and cladding. The conference included a tour of the Japan Proton Accelerator Research Complex (J-PARC) spallation source, which provided insight into cutting-edge neutron imaging beamlines that could inform the development of new beamlines or the enhancement of existing ones at LANSCE. Additionally, Vogel took part in a training course on the Rietveld-type analysis code (RITS) data analysis software, which is instrumental for extracting data from Bragg-edges to map UO2 lattice parameters in two-dimensional experiments.

The trip also facilitated collaboration opportunities through invitations from the JAEA and academic researchers and included a workshop dedicated to neutron-based microstructure characterization. Each of these one-day events was aimed at fostering information exchange that could guide research on uranium-based alloys or cladding materials, applying microstructural characterization to materials relevant to the nuclear energy mission. The NEUWAVE conference, being a premier forum for wavelength and energy resolved neutron imaging, provided a platform to learn about the latest advancements in techniques like grating interferometry, Bragg-edge analysis, and neutron absorption resonance imaging. These techniques are pertinent to LANSCE's work in characterizing fresh and irradiated nuclear fuels.

Luca Capriotti traveled to Karlsruhe, Germany, to visit the Joint Research Centre (JRC)-Karlsruhe hot laboratories, aiming to reconnect with researchers at this prominent European R&D hub. In conjunction with this visit, he participated in the 16th Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation (IEMPT). The engagement at JRC-Karlsruhe, which is part of the European Commission, was tailored to enhance international collaboration and rekindle partnerships focusing on post-irradiation examinations and expertise in nuclear fuel; it also opened avenues for potential new collaborations between the European Union (EU) and the U.S., such as prospective The International Nuclear Energy Research Initiative (INERI) projects.

The NEA-IEMPT workshop served as a platform for sharing information and provided opportunities to stay informed about the latest technologies involved in closing the nuclear fuel cycle, developing transmutation fuels, and managing the inventory of minor actinides. This meeting played a crucial role in broadening access to international data related to fast reactor irradiations and post-irradiation examination on the topics discussed, as well as to the computational codes employed to simulate these experiments.

Several staff members from the AFC attended the NEA Framework for Irradiation Experiments (FIDES) technical advisory meeting and its governing board meeting. Colby Jensen, INL's Technical Lead for Transient Testing and the principal investigator (PI) for a new proposal on LOCA testing, dubbed "LOC," as well as the lead for the ongoing High Burnup Experiment on Reactivity Accident (HERA) project's international modeling exercise, presented the outcomes of the HERA modeling exercise. Additionally, he introduced the new LOC proposal for the upcoming FIDES funding cycle. Jensen engaged in numerous technical exchanges on these and other irradiation testing projects.

Dan Wachs, the National Technical Director (NTD) for AFC and a co-chair of the FIDES Technical Advisory Committee (TAC), was also in attendance. During this time, FIDES administrators called for proposals for future projects. INL submitted a follow-on proposal, which Jensen presented, allowing for direct engagement with NEA FIDES stakeholders, who were able to ask questions and discuss the proposal in detail.

David Kamerman, the INL coordinator for FIDES and the DOE project representative, attended the meeting too. He presented a proposal for an upcoming INL and DOE project under the FIDES framework.

Chris Murdock, another meeting participant, serves as the INL Experiment Manager (EM) for the FIDES Accelerated Testing of Materials in Capsules (AToMiC) experiment. He oversees the AFC Program's Advanced Fuels ATR Irradiations experiment projects and is the Experiment Manager for the new FIDES Joint Experimental Programme (JEEP) on the AToMiC experiment series. AFC's objectives were advanced through the presentation of the AToMiC experiment proposal and the provision of comprehensive planning for collaboration and integration regarding the AToMiC proposal. This strategic approach was designed to attain approval and secure funding from FIDES for irradiation testing within the Advanced Test Reactor (ATR).

Alex Swearingen participated in the Materials Modelling and Simulation for Nuclear Fuel Workshop (MMSNF 2023) held in Ontario, Canada. There, a presentation was given on comparing Fission Accelerated Steady-state Test (FAST) experiments with BISON computational simulations. Alex engaged with fellow modelling experts to exchange methods and best practices related to nuclear fuel modelling. A primary objective was to cultivate a global community of researchers with diverse expertise in fuel performance modeling and metallic fuel design. The development of this network aims to draw on external insights to enhance the modeling of FAST experiments and optimize the application of simulation outcomes for advancing AFC and advanced low enriched uranium (aLEU) designs.

Cindy Adkins took part in the Thermodynamics of Advanced Fuels - International Database (TAF-ID) Programme Review Group in Hamilton, Canada. The event served as a platform for engaging with the global community's advancements in the thermodynamic database for nuclear fuels and materials. Insights gained from this meeting will be integrated into INL's ongoing research activities. The review group provided a forum to learn about new techniques and approaches that enhance the accuracy of examinations for DOE programs. INL's recent research was presented, and dialogues with international partners were conducted, covering topics like recent test results, cooperative fuel development initiatives, advancements in characterization, and thermodynamic modeling methods. Discussions also delved into the quantification of microstructures in advanced nuclear fuel forms, measuring thermodynamic properties post-irradiation, and interpreting these findings. Data from these discussions will be contributed to the TAF-ID working database for collective reference and use.

Nicolas Woolstenhulme, Klint Anderson, and Nate Oldham attended the Design and Engineering of Vehicles for In-Core Experiments in Materials Test Reactors (DEVICE-MTR) Working Group Meeting in Mito, Japan, a consortium of skilled irradiation test designers and engineers from top materials test reactors globally. At this gathering, they shared insights, discussed design solutions, and addressed common challenges in the realm of in-core experiment vehicles, crucial for advancing state-of-the-art irradiation techniques under the DOE's fuels and materials test programs.

Nicolas Woolstenhulme, as the irradiation testing lead for AFC, shared his expertise on spectral tailoring design strategies, contributing to the development of unique conditions for nuclear material experiments. Klint Anderson, the lead design engineer for the Transient Water Irradiation System in the TREAT (TWIST) experiment vehicle, highlighted the recent inauguration and commissioning tests of the TWIST capsule. He also engaged with peers to enhance hot cell fabrication and assembly techniques in preparation for upcoming tests. Nate Oldham, the Technical Lead for I-Loop equipment designs at ATR, exchanged ideas on irradiation test rig designs with fellow engineers and researchers.

Their participation played a pivotal role in bolstering networking and collaborative efforts, setting the stage for future research proposals and the development of irradiation capabilities within frameworks like FIDES, ensuring DOE's test programs remain at the forefront of advanced fuel and material development.

Dr. Aaron Colldeweih embarked on a research trip to Villigen, Switzerland, where he utilized the high-resolution Neutron Microscope (NM) at the esteemed Paul Scherrer Institute (PSI) to perform advanced neutron radiography measurements. These experiments aimed to characterize the performance of chromium-coated ATF cladding; a prominent ATF concept being developed under the U.S. AFC program. By examining the cladding under simulated LOCA conditions previously conducted at INL and ORNL, Dr. Colldeweih was able to collect unparalleled post-LOCA test data on pre-strained ATF cladding, thereby contributing to its ongoing development.

This venture not only facilitated the exchange of technical expertise, which will be instrumental in enhancing the high-resolution radiography capabilities at INL's Neutron Radiography (NRA) reactor, but also reinforced the collaborative efforts between INL and ORNL in ATF cladding advancement. Through this work, INL continues to underscore its position as a global leader in nuclear fuel and materials research and development.

Jason Schulthess and Nathan Capps traveled to Nykoping, Sweden to participate in a Studsvik Cladding Integrity Program (SCIP) cross-discipline collaboration meeting, which concentrated on the outcomes of phase 4 and planning for phase 5 of the project. The discussions were anchored in the ATF and Fuel Safety Research Program, leveraging INL's TREAT Facility to conduct transient experiments critical for the qualification of new reactor fuels and amendments to existing fuel licenses.

The meeting was a strategic platform to enhance the scope of research conducted using the TREAT facility by transferring knowledge from earlier phases of the cladding integrity project to INL. It fostered technical dialogues between INL representatives and SCIP team, facilitating the sharing of insights supporting ongoing experiments and modeling work on fuel and cladding interactions, as well as fuel safety research. Key discussion points included the behavior of coated claddings under transient conditions, outcomes from out-of-pile loss of coolant tests at Studsvik, upcoming in-pile tests at INL, and various aspects such as ceramography of irradiated fuel, cladding creep, and ring tensile testing.

Jason Schulthess, leading INL's PIE team for fuel safety research, capitalized on the opportunity for technical exchanges with international counterparts at Studsvik conducting parallel studies. Meanwhile, Nathan Capps attended to assimilate results and propose enhancements to test conditions, aiming to integrate findings into the broader AFC and the NEAMS programs. The insights gathered through the SCIP program are set to inform and refine future testing within the AFC framework.

The AFC team at INL welcomed representatives from the French Alternative Energies and Atomic Energy Commission (CEA) as part of their collaborative efforts under the Bilateral Cooperation Agreement with the U.S. DOE. The participants are pictured in Figure 4. The meeting focused on the progress and initiatives of Working Group 3, which specializes in Advanced Fuels and Materials. This partnership has also laid the groundwork for the recently proposed expansion of nuclear energy cooperation, known as the 'Macron Roadmap,' at the administrative level between the two nations.

Discussions during the meeting encompassed various aspects of the DOE program activities, including advancements in fuel performance codes and the insights from experimental research such as fuel safety testing, characterization of material properties, and PIE.

Technical experts from both CEA-Cadarache/CEA-Saclay and INL, along with colleagues from Los Alamos National Laboratory (LANL), convene annually to review the progress of their joint ventures and to strategize future collaborative activities. These meetings are essential for maintaining synergy and advancing shared objectives in nuclear fuel and materials research.

Nathan Capps participated in and delivered a presentation at the Workshop on Multi-Physics Core and Plant Simulation in Vienna, Austria, which emphasized fuel behavior in LWR-based Small Modular Reactors (SMRs). Hosted by the International Atomic Energy Agency (IAEA), the workshop facilitated discussions on the performance of SMR fuel and pinpointed opportunities for benchmarking. The meeting played a critical role in substantiating the need for further fuel testing as well as the advancement of related modeling efforts.


*Figure 4. CEA representatives tour the TREAT Facility as part of the CEA Meeting (December).*

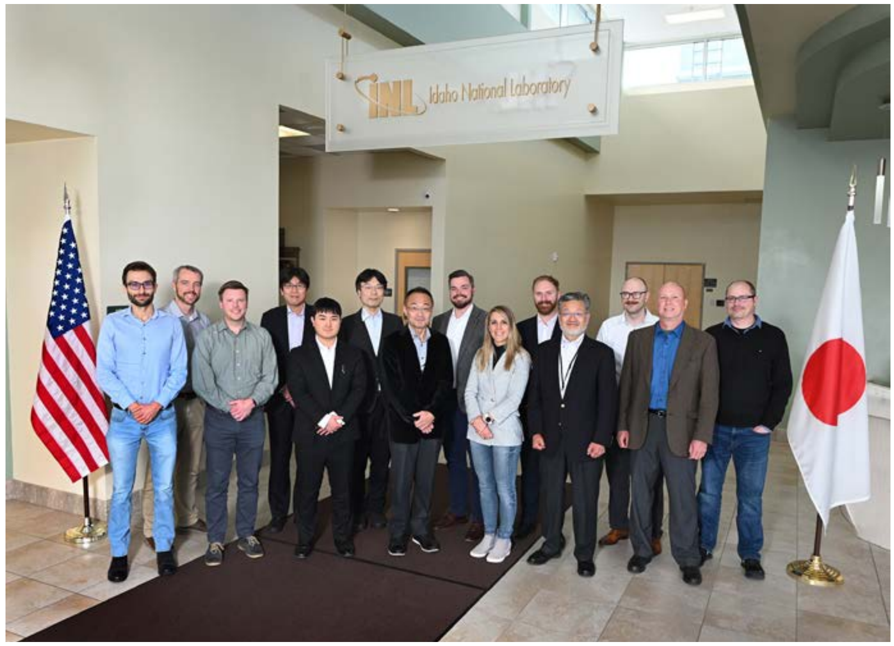



*Figure 5. Participants from the CNWG Meeting held at INL (March).*

The Advanced Fuels Technical Experts Meeting between the JAEA and INL took place at INL from March 11-13, 2024, under the auspices of the Civilian Nuclear Working Group (CNWG). The group is pictured in Figures 5 and 6. This annual meeting was part of the ongoing collaboration outlined in the JAEA-DOE bilateral agreement. The JAEA team received updates on the TREAT Facility experiment's irradiation progress, particularly the Clad Mixed Oxide Transient Over Power (MOXTOP) experiment, which has been a cornerstone of the collaborative research and development agreement (CRADA) with Japan for the past three years.

This engagement served to fortify the DOE and INL's bilateral partnerships in nuclear energy research and helped INL to meet its commitments under the CRADA for conducting TREAT Facility experiments. JAEA representatives were briefed on the outcomes of recent experiments and deliberated on the planning of future endeavors. The successful execution of these experiments represents the culmination of the CRADA (No. 20-CR-01) activities with Japan over the previous three years.

The meeting included comprehensive overviews and the latest updates on several key tasks, including:

- **Task 1:** Conducting experiments to determine the basic properties of oxide fuels.
- **Task 2:** Developing and evaluating PIE data for advanced oxide fuels.
- **Task 3:** Modeling and simulation efforts to describe characteristics of irradiated fuels.
- **Task 4:** Planning for the irradiation tests of a Minor Actinide (MA)-MOX fuel pin.
- **Task 5:** Core materials research, focusing on the materials used within the reactor core.

In Aix-en-Provence, France, personnel from AFC took part in a workshop with the IRSN to outline and deliberate on the proposed irradiation projects slated for the TREAT Facility. The AFC team provided insights on the capacities and limitations of conducting LOCA tests in TREAT. They also spearheaded conceptual discussions alongside IRSN experts to refine the scope of this emerging program.

A significant development from this workshop was the drafting of an international collaborative program proposal for LOCA testing with the FIDES group. The establishment of such a program is instrumental in maintaining the high standards of nuclear safety research that are characteristic of both American and European approaches, particularly in the development and implementation of nuclear fuels for LWRs.

Participants also delineated technical objectives for future cooperation, finalized arrangements for an IRSN researcher's upcoming tenure INL, and laid the groundwork for a new understanding between IRSN and DOE, succeeding the recently expired memorandum of understanding (MOU) between INL and IRSN.

In Amsterdam, Netherlands, AFC staff participated in the NEA FIDES technical advisory and governing board meetings. At these meetings, Colby Jensen, the INL Technical Lead for Transient Testing, who is also the P) for a new proposal on LOCA testing named "LOC," and the current lead for HERA project's international modeling exercise, shared insights on these initiatives. The discussions centered around the prospective FIDES funding for the new LOCA proposal, and other attending staff engaged in various technical exchanges regarding the projects and other irradiation testing endeavors.

The meeting's objectives were twofold: firstly, to assert international leadership and secure funding for the novel FIDES LOC proposal, which advocates for LOCA testing in the TREAT Facility —the sole in-pile testing venue outside of Russia—aiming to support U.S. industry ambitions to raise burnup limits. Secondly, to bolster and expand collaborative efforts on fuel safety testing with IRSN, a premier R&D organization, which expressed interest in financing future projects in TREAT.

The face-to-face presentations provided an opportunity for stakeholders to delve deeply into the topics, fostering full engagement, ensuring transparent communication, and reinforcing essential partnerships within the group.

In Tokyo, Japan, staff from AFC attended a CNWG session with representatives from JAEA. The focus of this meeting was on advancing the qualification process for multiple ATF cladding candidates, aimed at their potential deployment in existing light water reactors. Additionally, the team visited Toshiba Energy Systems & Solutions Corporation in Yokohama, a Japanese frontrunner in the development of ceramic cladding, to explore possible international collaborations on cladding test research.

Such international cooperation and agreements are vital to the progress of advanced nuclear fuel cladding technology. Through a bilateral agreement with Japan, AFC is committed to fostering international partnerships. This meeting was instrumental in deepening the collaborative ties with Japan, ultimately contributing to the expedited advancement of safer cladding materials for LWRs, particularly under accident conditions.

In Paris, the Working Party for Fuel and Materials (WPFM), operating under the auspices of the NSC, is committed to forging an international consensus on the acceleration of nuclear fuel innovation. Dan Wachs, the chair of the WPFM at the NEA, led a pivotal meeting at NEA headquarters, where he provided a detailed update on the group's endeavors to the NSC.

Representatives from AFC took an active role in the Expert Group of Fuel Materials (EGFM), focusing on the appraisal and advancement of nuclear fuels for Generation IV reactors. They put forward methodologies for the qualification of these advanced fuels and collaborated with EGFM colleagues in pinpointing and addressing knowledge gaps in the field.

This involvement with the NEA's working parties is crucial for achieving the United States' ambitions for the development and implementation of nuclear technologies. As chair, Dan Wachs' direction and the collaborative efforts of AFC are vital contributions to the global dialogue on innovating nuclear fuels, an initiative that is strategically overseen by the NSC at NEA.

AFC personnel participated in a productive meeting with the SCIP team in Stockholm, Sweden, focusing on collaborative efforts in fuel safety research. They engaged in a detailed exchange of information to support ongoing experiments and modeling efforts on the interaction and integrity of fuels and cladding. Topics of discussion included the behavior of coated cladding under transient conditions, particularly how different phenomena influence burst behavior, findings from out-of-pile LOCA tests conducted at Studsvik, upcoming in-pile LOCA tests at INL, and various aspects of fuel-cladding interaction. Additionally, the meeting covered ceramography of irradiated fuel, cladding creep behavior, and ring tensile testing.

This cross-disciplinary meeting, sponsored by the SCIP group, concentrated on the progress from phase 4 of their project and strategized for phase 5. The discussions were intrinsically linked to the ATF and Fuel Safety Research Program, which leverages the TREAT Facility for conducting crucial fueled transient experiments. These experiments are integral to the qualification of new reactor fuels and the amendment of licenses for existing fuels. The collaborative nature of this meeting was instrumental in broadening the scope of research made possible by TREAT and in facilitating the transfer of valuable insights from earlier stages of the cladding integrity project to the AFC team.

Between July 1-3 IRSN and AFC convened at INL to discuss new TREAT Facility experiment designs and fabrication (see Figure 6). This meeting built on the collaborative relationship established in 2017, under a MOU, where IRSN conducts complementary experiments at France's CABRI reactor, akin to INL's TREAT's mission. Recognized for their expertise in nuclear safety, IRSN's visit to INL was key for sharing insights into TREAT's experiment processes. Attendees toured relevant INL facilities and exchanged details about the Transient Water Irradiation System in TREAT (TWIST) test device, with IRSN aiming to create their own TWIST by 2026, enhancing the partnership's commitment to nuclear research.

Colby Jensen presided over the annual meeting of the SFR Advanced Fuels (AF) Project Management Board (PMB), part of the Generation IV Forum (GIF), from August 20-23, 2024. DOE supports the GIF initiative as a key organizing member, facilitating technical contributions from various U.S. national laboratories. Hosting duties for the SFR AF PMB meeting rotate among the member countries each year. This international gathering brought together PMB representatives from multiple nations to engage in a collaborative dialogue, sharing the progress and updates from their respective countries on the advancement and implementation of advanced reactor technologies and associated fuel systems.

*Figure 6. Participants from the CNWG Meeting held at INL (March).*

### Working Group for Fuel Safety (Paris)

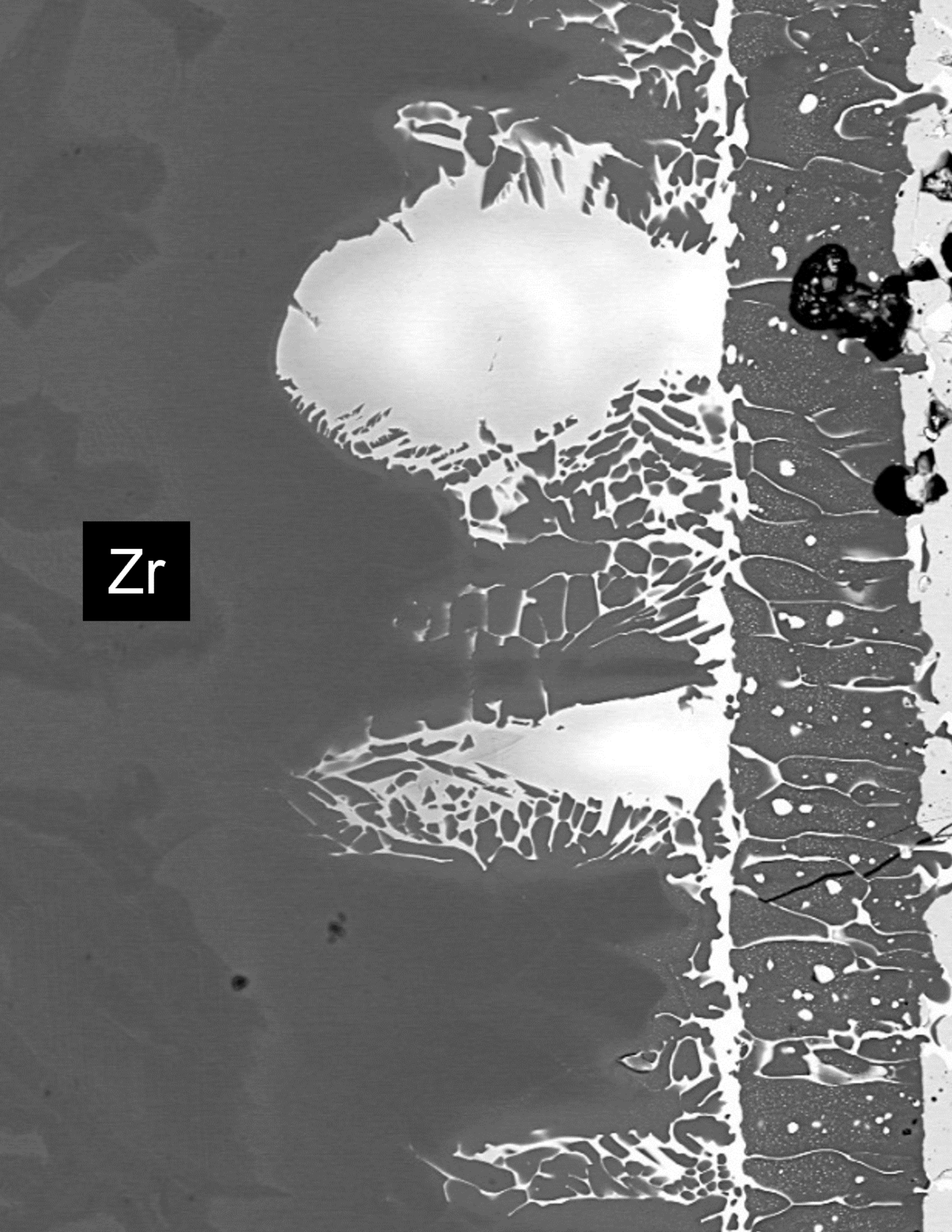
AFC staff actively participated in the TopFuel 2024 conference in Grenoble, France, contributing through presentations and a poster session. Some of the presentations include:

- Nathan Capps presented a paper on validating the BISON fuel performance code against HBU PIE data from ORNL, showcasing the collaborative efforts with the NEAMS program and the simulation of commercially irradiated materials.
- David Kamerman shared progress on post-irradiation examinations of ATF rods from Byron generating station, emphasizing the importance of such data in advancing nuclear fuel R&D at INL.

- Jason Harp engaged in discussions on fuel performance, particularly in the context of high burnup and accident-tolerant fuels. His collaborations with industry giants like Westinghouse, General Electric, and Framatome were highlighted, especially in transient fission gas release and PIE of light water reactor fuel.
- Mackenzie Ridley's presentation focused on the effects of hydrogen on cladding performance during simulated LOCA conditions, underscoring the necessity of considering hydrided cladding in nuclear fuel performance codes and reactor cladding failure criteria.
- Nicolas Woolstenhulme delivered two papers on advanced nuclear fuel designs and developments in irradiation testing technology, crucial for advancing DOE-NE's mission in nuclear fuels R&D and fostering international collaborations.
- Robert Hansen showcased a poster on mechanical testing and PIE capabilities at INL for ATF cladding technologies, based on his recent work submitted to the International Journal of Solids and Structures.

TopFuel is an international annual meeting orchestrated by the European Nuclear Society (ENS), the American Nuclear Society (ANS), the Atomic Energy Society of Japan, the Chinese Nuclear Society, and the Korean Nuclear Society.

This premier gathering convenes world-renowned experts in the nuclear sector for an exchange of the latest developments in nuclear fuel management technology. Through a series of discussions and presentations, participants gain insights into state-of-the-art research findings. These insights are then applied by professionals in their daily work to enhance the design, production, operation, and disposal of contemporary and future high-performance nuclear fuels.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## Celebrating Excellence in AFC

*Phyllis King (Idaho National Laboratory [INL])*

Takaaki Koyanagi received the esteemed 2024 Frontiers of Materials Award from The Minerals, Metals & Materials Society (TMS) (see Figure 1). This accolade is competitively bestowed upon an outstanding early career professional who demonstrates the ability to coordinate a significant Frontiers of Materials Event, focusing on a cutting-edge or emerging technical subject, at the TMS Annual Meeting & Exhibition. Takaaki successfully arranged a special symposium as a Frontiers of Materials Event, highlighting Innovative Ceramic Processes for Nuclear Applications.

https://www.tms.org/portal/portal/Professional_Development/Honors___Awards/Frontiers_of_Materials_Award.aspx


*Figure 1. Takaaki Koyanagi received the TMS 2024 Frontiers of Materials Award.*

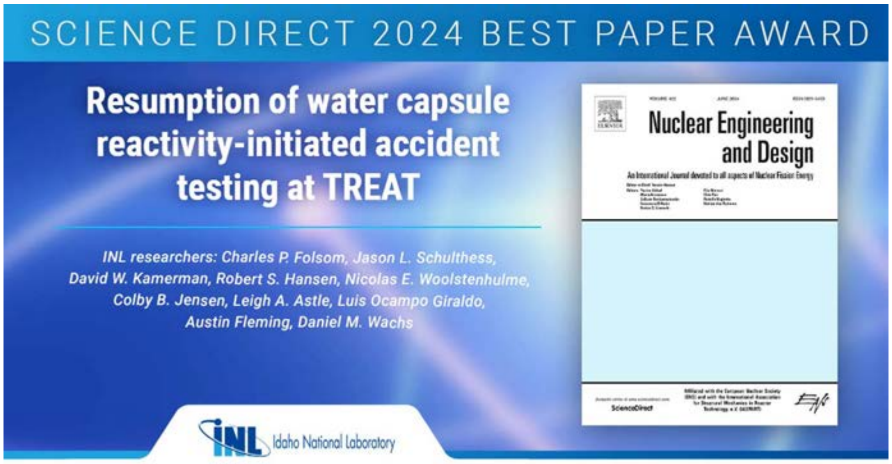



*Figure 2. Science Direct 2024 Best Paper Award from the Nuclear Engineering Design Journal.*

A Transient Reactor Test (TREAT) Facility experiment conducted by Idaho National Laboratory (INL) earned the prestigious 2024 Best Paper Award from the Nuclear Engineering Design Journal, published by Science Direct (See Figure 2). The findings from this research are instrumental in enhancing computer models and validating advanced, more secure reactor designs and fuel systems.

*Charles P. Folsom, Jason L. Schulthess, David W. Kamerman, Robert S. Hansen, Nicolas E. Woolstenhulme, Colby B. Jensen, Leigh A. Astle, Luis Ocampo Giraldo, Austin Fleming, Daniel M. Wachs, Resumption of water capsule reactivity-initiated accident testing at TREAT, Nuclear Engineering and Design, Volume 413, 2023, 112509, ISSN 0029-5493, https://doi.org/10.1016/j.nucengdes.2023.112509.*

https://www.sciencedirect.com/science/article/pii/S0029549323003588

**Abstract:** A series of integral reactivity-initiated accident commissioning experiments were completed in a new static water capsule in the Transient Reactor Test Facility (i.e., TREAT), marking the first such tests in the United States in more than 40 years. The test campaign included a verification test followed by five tests in the Static Environment Rodlet Transient Test Apparatus capsule. The capsule's initial conditions varied from room temperature and pressure up to 200 °C and 2.5 MPa, with energy depositions varying between ~ 500–1,100 J/g$UO_2$. The series of tests allowed for a number of instrumentation qualifications and demonstrations, including cladding thermometry, rodlet plenum pressure, cladding elongation, and an electro-impedance boiling detector. Post-transient examinations such as gamma emission spectroscopy, profilometry, and microscopy were performed to document the end state of the fuel rods. The results

*Figure 3. (From left) Erin Searcy, Acting Deputy Laboratory Director for Science and Technology, TWIST Capsule Team, John Wagner, Laboratory Director.*

from the experiments show that the fuel rodlets behaved very similar to historical tests under similar energy depositions. This paper documents the design of the capsule and highlights some results from the commissioning tests and post-transient examination.

During the 28th Annual Laboratory Director Awards Ceremony in May at the Idaho National Laboratory, the TWIST Capsule Team was commended for transforming a challenge into a breakthrough, establishing a novel worldwide proficiency in nuclear testing.

They were honored with an Outstanding Impact Award. Their contributions have the potential to influence both current and forthcoming commercial nuclear power fleets. The team, comprising Colby Jensen, Klint Anderson, Cindy Fife, Austin Fleming, Charles Folsom, Matthew Ramirez, and Todd Pavey (who is not depicted), is featured in Figure 3 above.

## Second Annual Metal Fuel Research Workshop

*Principal Investigator: Colby Jensen (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Phyllis King (INL)*

The Department of Energy (DOE) Advanced Fuels Campaign (AFC) Metal Fuel program under Next Generation Fuels serves broad U.S. interests in development of clean energy technology through optimized nuclear fuel cycle solutions. Metal fuel for fast reactors is a high priority fuel technology that provides several unique advantages and is a recognized mature technology for near-term deployment. The AFC program has initiated annual Metal Fuel workshops in 2023 to bring stakeholders together to bolster community relationships and ensure DOE research and development efforts will achieve needed through review of technical work and plans.

### Project Description

The AFC program organized and hosted the second annual metallic fuels workshop inviting stakeholders from industry, the Nuclear Regulatory Commission (NRC), DOE programs and national laboratories, active university partners, and international collaborators. The workshop included participation (see Figure 1) from five DOE laboratories, DOE-Idaho, DOE-Headquarters, seven reactor design/fuel companies, NRC, DOE programs including Nuclear Energy Advanced Modeling and Simulation, Fast Reactor Program, and Material Recovery and Waste Forms Development. The meeting agenda (attached) included two days of presentations and discussion split between the AFC program and other stakeholders and collaborators and a third day of tours and highlight presentations with a thematic focus on fuel fabrication.

### Accomplishments

Key conclusions expand upon the foundational takeaways from 2023:

- Metallic fuel technology has a broad set of stakeholders interested in collaboration to enable deployment and long-term success. While sodium fast reactor applications dominate, stakeholders also include interests in light water reactor applications. The second annual workshop was well-received and encouraged to continue annually.
- The AFC program has developed a first revision for a research and development (R&D) plan and is soliciting review and input from stakeholders to ensure federal investments are as impactful and far-reaching as possible. The plan builds off planning stemming from 2020 with the formation of the LIFT proposal and addresses conclusions from last year's workshop. Priority work is towards:
  - Establishing a reference design basis through NRC-reviewed reports and the Experimental Breeder Reactor-II Fuels Irradiation & Physics Database remains

*Figure 1: Photo of the 2nd Annual Metal Fuel Research Workshop participants.*

***The AFC program hosted the 2nd Annual Metal Fuel Research Workshop with enthusiastic participation from more than 60 participants coming from 15 industrial and research institutions from around the world.***

  a crosscutting priority for all stakeholders to lay foundation for first deployments and next innovations. These activities are progressing.
  - Data gaps identification and addressal using precious and limited legacy metallic fuel materials preserved at INL. The DOE program has started work in most identified data gap areas. A current capability/data gap is related to furnace testing for medium-slow transient testing.
  - The AFC program leadership will share the R&D plan and follow up with individual stakeholder meetings about feedback and interests.
- Next-generation fuel design in the AFC program still plans to focus on fabrication technology development and improved high temperature strength cladding alloys. An interactive survey was conducted with workshop participants to capture general input on fuel design options to kick off a detailed engineering evaluation to be conducted by the AFC program.
- Reactor designers are aggressively pursuing fuel qualification efforts for their startup fuel designs. Several designers are also interested in next generation fuel designs including goals of higher burnups and fuel recycle designs.
- Broad interest (DOE to industry) in transuranic fuels is resurging in the US due to economic and waste management reasons. International interests in Japan and Korea continue strong in this area.
- International interest in metallic fuels is also increasing. Representatives from the Japan Atomic Energy Agency and the Korean Atomic Energy Research Institute presented strong R&D programs emphasizing the importance of metallic fuels in their respective national fast reactor strategies. Japan will make a national decision in 2026 to select either oxide or metallic fuel after it completes an ongoing evaluation.

# Participation in Nuclear Energy Agency's Framework for Irradiation Experiments

*Principal Investigator: David Kamerman (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Colby Jensen, Boone Beausoleil (All INL)*

The Nuclear Energy Agency's Framework for Irradiation Experiments (FIDES) is a response to the shutdown of the Halden reactor in 2018 and subsequent cessation of irradiation experiments under the Halden Reactor Project. The international framework provides a platform for research institutions, regulatory bodies, nuclear operators and vendors to have access to a variety of research reactors for jointly funded irradiation experiments.

## Project Description

The current makeup of the FIDES community includes participation of many western European countries, the United States, as well as the Republic of Korea and Japan. The group meets twice a year to hear updates on the irradiation tests and discuss the strategic future of the framework. In FY24 the group met in Mito Japan in October 2023 (Figure 1), and in Amsterdam Netherlands (Figure 2) in April of 2024.

The Advanced Fuels Campaign (AFC) has a leading role in the FIDES project through its sponsorship of 3 of the Joint Experimental Projects (JEEPs) in the FIDES portfolio. Together these JEEPs represent close to 50% of the resources in FIDES. The national technical director of the AFC also serves as the Co-chair of the FIDES technical advisory group and AFC members have been instrumental in developing many of the guidance documents for FIDES including the FIDES strategic plan.

AFC sponsored JEEPs include the High burnup Experiments in Reactivity-initiated Accidents (HERA) project, the Loss of Coolant High burnup (LOC-HBu) project, and the Accelerated Testing of Materials in Capsules (AToMiC) project. The projects span a variety of topics from safety testing of light water reactor fuels to accelerated testing of advanced materials for new generation 4 reactor concepts.

***The Advanced Fuels Campaign shows international leadership in the area of Irradiation Testing of Nuclear Fuel Materials through its participation in the Nuclear Energy Agency's Framework for Irradiation Experiments which is seen as the successor project to the Halden Reactor Project.***

*Figure 1. Group photo of FIDES meeting in Mito Japan, October 2023.*

**Accomplishments**

HERA was the first AFC project and part of the inaugural FIDES portfolio. HERA is dedicated to the understanding of modern light water reactor (LWR) fuel performance at high burnup under reactivity-initiated accidents. The objectives of HERA are to:

1. Quantify the impact of pulse width on fuel performance, offering new insight into the applicability of existing data
2. Generate new data on high burnup fuel under pulse conditions prototypic of LWRs
3. Quantify the additional margin provided by modern cladding alloys to pellet cladding mechanical interaction failure limits
4. Offer improved data for modelers using specially designed tests that eliminate key uncertainties in high-burnup fuel tests

HERA involves a total of 12 transient pulse type irradiations, 4 in the Nuclear Safety Research Reactor in Japan and 8 in the TREAT reactor at INL. Four of the transients irradiations have been completed with the remaining 8 to occur in the next 2 years. During a ranking of projects for consideration in the second phase FIDES, HERA received the highest ranking of all projects being considered.

*Figure 2. Group photo of FIDES meeting in Amsterdam Netherlands, April 2024.*

LOC-HBu is the second AFC project. LOC-HBu is dedicated to the understanding of LWR fuel performance at high burnup under loss of coolant accident (LOCA) conditions. The objectives of LOC-HBu are to:

1. Evaluate the integral impacts of stored energy heatup and decay energy heatup conditions (depending on the HBu core design) on fuel fragmentation, relocation and dispersal and transient fission gas release (tFGR) during a LOCA
2. Measure key behavioral phenomena in situ (e.g., cladding deformation and burst dynamics, tFGR, fuel relocation and dispersal, and cladding balloon surface temperature) to reduce the uncertainty in phenomena interdependencies (e.g., temperature profile and plenum pressure/volume), while also allowing for first-of-a-kind model development and validation
3. Perform LOCA tests on material samples with detailed relevant microstructural characterization pre- and post-testing
4. Provide expanded LOCA datasets for model development and code validation regarding relevant HBu fuels near important burnup thresholds

LOC-HBu involves 4 transient irradiations with shaped transients to mimic LOCA conditions including critical post blowdown phase of the transient when heating rates are driven more by stored energy in the fuel. During a ranking of projects for consideration in the second phase of FIDES, LOC-HBu received the second highest number.

The ATOMIC proposal seems to irradiate fuel and cladding materials for advanced (Generation IV, GenIV) reactors within drop-in capsules in the Advanced Test Reactor. The objectives of ATOMIC are to investigate the microstructural evolution of fuel and cladding materials at relevant temperatures and irradiation conditions. The ATOMIC project will utilize the Fission Accelerated Steady-state Testing method by using geometrically scaled down tests to achieve accelerated irradiation rates for advanced mixed oxide fuels, metallic fuels, and SiC/SiCf cladded $UO_2$ for high temperature reactor applications. Additionally, the ATOMIC test will utilize an instrumented test called the Irradiated Material Properties Accelerated Characterization Test to validate the ability to use series of thermal couples to test the thermal conductivity of metallic fuels in-pile. Lastly, ATOMIC will utilize a series of capsules to test TRi-structural-ISOtropic fuels to investigate uncertainty in behavior at low and high temperatures. This was initially presented to the Technical Advisory Group and Governing Board in November, 2023 and was accepted as the only proposal in FIDES investigating advanced reactor fuels.

# Inaugural Annual LWR Fuel Research Workshop

*Principal Investigator: Dan Wachs (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Phyllis King (INL)*

The Department of Energy (DOE) Advanced Fuels Campaign (AFC) Accident Tolerant Fuel (ATF) program serves broad U.S. interests in development of clean energy technology through collaborative development of innovative fuel technology for light water reactors (LWRs). AFC has been performing research in collaboration with a full spectrum of stakeholders (research, industry, and regulatory), both domestically and internationally, to develop ATF technology for over a decade. It is expected that realization of this technology will deliver enhanced safety features and improve economic performance of both pressurized water reactors and LWRs. The AFC program initiated an annual LWR Fuel Workshop in 2024, complementing the ongoing annual industry-led Collaborative Research on Advanced Fuel Technologies (CRAFT) workshops, to bring stakeholders together to bolster community relationships and disseminate state-of-the-art research results.

## Project Description

The AFC program organized and hosted the inaugural LWR fuels workshop inviting stakeholders from industry, the Nuclear Regulatory Commission (NRC), DOE programs and national laboratories, active university partners, and international collaborators. The workshop attracted more than 100 participants (see Figure 1) from five DOE laboratories, DOE-Idaho, DOE-Headquarters, three industry fuel suppliers, several utilities, several reactor/fuel development companies, NRC, as well as DOE programs including Nuclear Energy Advanced Modeling and Simulation, Light Water Reactors Sustainability Program, and Material Recovery and Waste Forms Development. The meeting agenda included two days of cross-cutting LWR research presentations and discussion split between the AFC program and other stakeholders and collaborators, two days focused on topics of specific interest to the CRAFT community (e.g., fuel fragmentation, relocation, and dispersal (FFRD) and time at temperature (t@T)), and a fifth day of tours of INL loss of coolant accident (LOCA) testing related facilities at the Materials and Fuels Complex (MFC).

***The AFC program hosted the inaugural LWR Fuel Research Workshop with enthusiastic participation from more than 100 participants coming from wide array of stakeholder communities (including research, industrial and regulatory institutions) from around the world.***

### Accomplishments

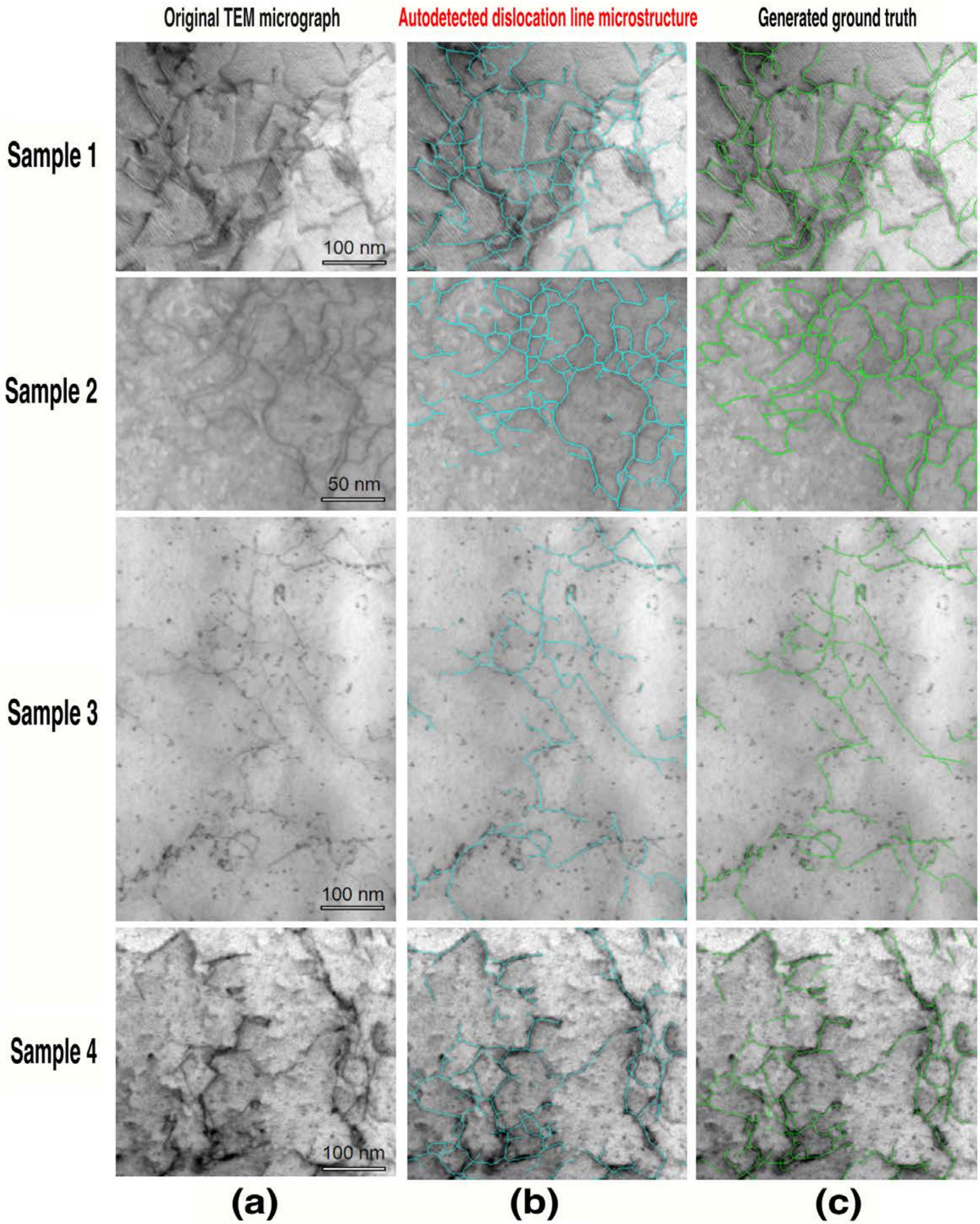
Key conclusions expand upon the foundational takeaways from 2023:

- **Innovation in LWR fuel technology is being achieved through joint development of ATF technologies.**
  - Some ATF technologies, described as near-term ATF, are approaching deployment phase including coated claddings, doped fuels, high burnup (HBU), and low enriched uranium. Progress in this area was the focus of the meeting.
  - In complement, longer-term ATF technologies will be transitioned into the innovative technology development branch of the Next Generation Fuels program to be launched in FY 2025.
- **Innovative LWR fuel technology has a broad set of stakeholders interested in collaboration to enable deployment and long-term success.** Stakeholders universally recognize the need to enable innovations that will improve the stability of the existing nuclear industry that delivers significant amounts of clean, affordable power to the public. This research is also critical to enabling the next generation of LWRs, both gigawatt scale and small modular reactor plants.

Figure 1. LWR Fuels Research Workshop: Photo of the inaugural LWR Fuel Research Workshop participants.

- The AFC program, in collaboration, with fuel suppliers and utilities has performed significant post irradiation exams on commercially irradiated ATF rods including 1st and 2nd cycle Westinghouse rods irradiated at Byron. Examinations include both non-destructive and destructive analysis. Preliminary results for ATF (both commercial lead test rods and specimens irradiated in the Advanced Test Reactor's ATF-2 experiment) were presented at the meeting.
- Mechanical testing methods for characterizing irradiated claddings were presented by four research organizations (INL, Oak Ridge National Laboratory [ORNL], Los Alamos National Laboratory, and Studsvik). These methods are being assessed under a round robin exercise that is intended to cross-validate them for 'universal' use. Although the limitations and constraints associated with available materials and hot cell facilities prevent development of appropriate standard, joint efforts to improve are anticipated.

- Ongoing research on reactivity-initiated accident behavior of ATF and HBU fuel was presented. Significant new HBU data is anticipated to be available next fall as part of the Nuclear Energy Agency Framework for Irradiation Experiments Project, High Burnup Experiments in Reactivity-initiated Accidents, being executed by INL in collaboration with the Japan Atomic Energy Agency and the Institute for Radiation Protection and Nuclear Safety.
- The AFC program has developed a stakeholder consensus research and development plan for FFRD behavior in HBU fuel. This plan was extensively reviewed and endorsed by the stakeholder community via CRAFT. Updates to ongoing work conducted at ORNL in the Severe Accident Testing Station and INL at the Transient Reactor Test (TREAT) facility were summarized as dual engagement with the CRAFT FFRD Technical Experts Group (TEG). This included:
    - Several tests at ORNL that focused on the effects of spacer grids on ballooning and release
    - Transient Water Irradiation Systems in TREAT LOCA test rig commissioning tests at TREAT and HBU rod refabrication and instrumentation to support these tests
- The CRAFT t@T TEG conducted a meeting as part of the workshop. The meeting assembled experts from laboratory, industry, and regulatory community to conduct an initial review of the t@T research plan. Activities included:
    - A detailed technical review of the relevant phenomena
    - Review of relevant materials available at hot cell facilities
    - Review of potential methodologies to assess response of zirconium alloys to short term thermal exposure
    - Review of thermal hydraulic uncertainties in currently used analysis tools



Original TEM micrograph
Autodetected dislocation line microstructure
Generated ground truth
Sample 1
Sample 2
Sample 3
Sample 4
100 nm
50 nm
100 nm
100 nm
(a)
(b)
(c)

# ADVANCED LWR FUELS/ HIGH BURNUP

## Development of High Temperature Thermomechanical Testing

*Principal Investigator: Joshua White (Los Alamos National Laboratory [LANL])*
*Team Members/Collaborators: Christopher Butler, Austin Nichols, Deana Tsang (All LANL)*

***A new high temperature load frame has been designed and installed that will enable validation of creep and fracture models of high-performance ceramic fuels of near-term accident tolerant fuels and next generation fuels.***

The Advanced Fuels Campaign (AFC) as well as the Nuclear Energy Advanced Modeling and Simulation (NEAMS) program are interested in understanding the high temperature thermomechanical performance of nuclear fuels to improve the mechanistic understanding of light water reactor (LWR) fuels. A specific focus of this effort is to examine the creep behavior of Cr-doped uranium dioxide ($UO_2$) fuels under operational temperatures and to characterize the mechanisms of creep acting with a varying Cr dopant concentration. In fiscal year 2024, a load frame was adapted to incorporate a high temperature furnace with controlled atmospheres to apply compressive loading and determine the creep behavior of ceramic fuels.

### Project Description

The determination of the mechanical properties of $UO_2$ fuels has been a challenge since the beginning of the use of nuclear power. Since the Fukushima Daiichi accident in 2011, there has been a renewed effort in the Department of Energy (DOE) and the greater scientific community to develop fuels that can reduce risk in accident scenarios. To facilitate this, the DOE-NE AFC is working to characterize several Accident Tolerant Fuels (ATF) approaches that can withstand accidents defined by the Nuclear Regulatory Commission (NRC). One such approach that is the subject of this work is the characterization of fuel with additive concepts that are softer and can undergo creep at higher temperatures under higher strains. At the pellet cladding interface this would reduce strain on the cladding from the fuel and in a loss of coolant accident scenario would reduce the likelihood of failure of the cladding. However, there is a lack of information regarding $UO_2$ with additives and while it is theorized that the fuel becomes softer at higher temperatures and strains it has not been experimentally verified. The grain size of actinide fuels provides a useful method to control fission gas release. As Kr and Xe have a low diffusion coefficient in $UO_2$ the increase of grain size increases the distance of travel required and will increase the fuels ability to retain fission gasses. Larger grain sizes, however, will result in a lower creep rate. The addition of dopants to actinide oxide fuels has been experimentally shown to augment the self-diffusion process at higher temperatures. This work was initiated to provide non-proprietary data and provide valuable information to benchmark near term ATF models within the NRC, the NEAMS program, and the nuclear research community.

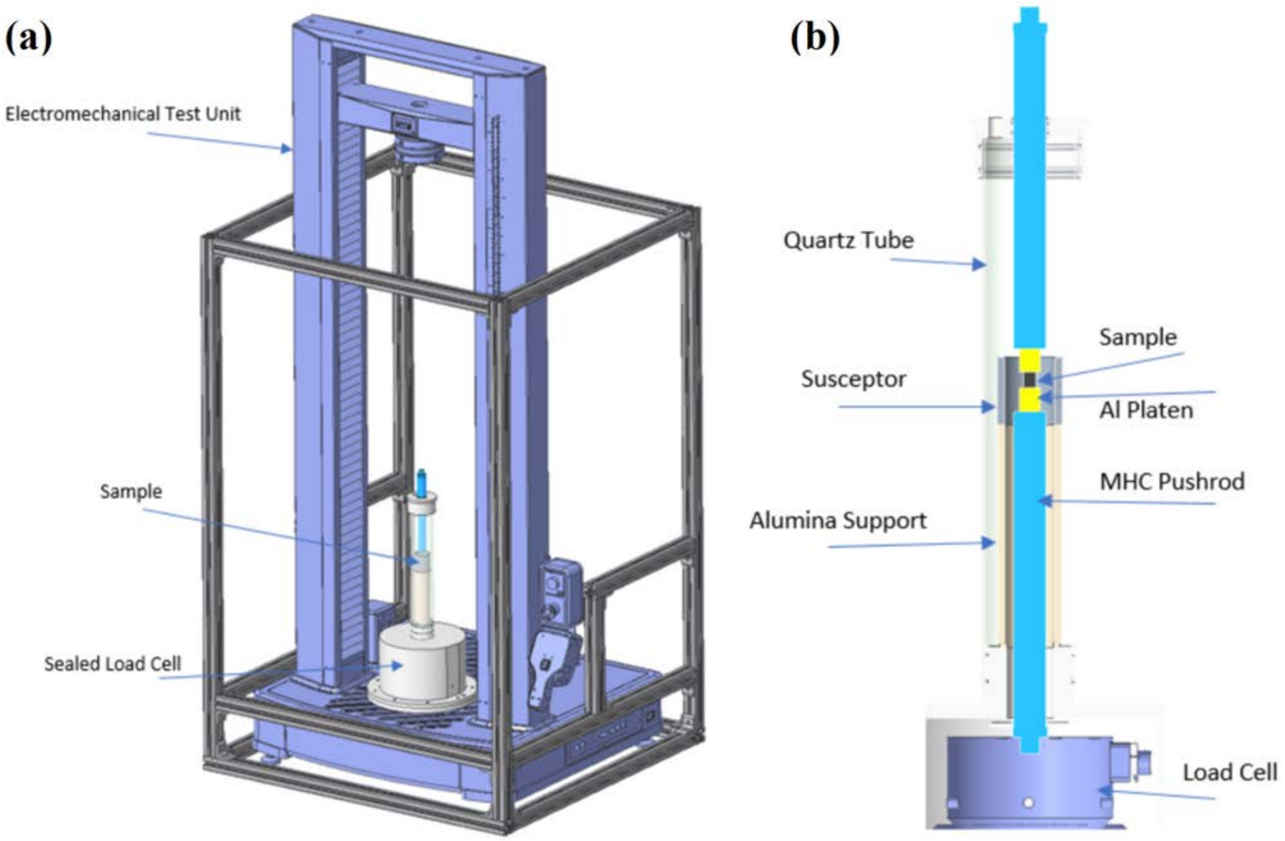



*Figure 1. (a) Conceptual design of the RF furnace with atmosphere in relation to the load frame with (b) zoomed in region of the furnace and sample environment.*

**Accomplishments**

LWR fuel is typically exposed to compressive loads and high temperature service ranging from 72 MPa [1] to 1600°C [2] temperatures under steady state operations. As such, the system was design to accommodate such conditions as well as be flexible to evaluate accident conditions. This system shall be composed of a Mechanical Test Systems Criterion C43.104 Electromechanical test system which was chosen to achieve typical failure loads of $UO_2$ from the literature. A loadcell provided with the system and a digital displacement gauge shall capture force and displacement to accurately measure the stress and strain within the system. The MTS Criterion C43.104 has minimum cross head speed of 0.005 mm/min to a maximum speed of 2000 mm/min with a position resolution of 50 nm. This should allow the test to operate with strain rates of 10-7 min-1 to 1.33·10-6 /min which are values measured in pile on fuel during reactor operation [3]. The device has a maximum rated force capacity of 10 kN. The furnace design in Figure 1 shows the integrated parts of the radio frequency furnace in the load frame with accompanying parts to maintain atmosphere within the ceramic tube.

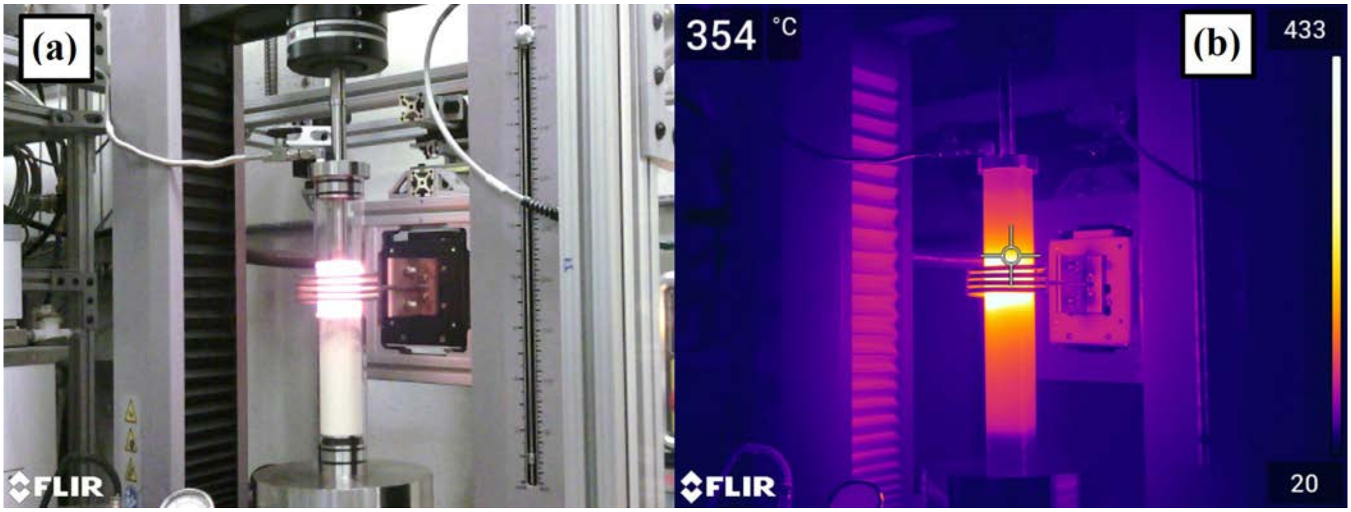



*Figure 2. (a) Visible light photo during operation and (b) thermal camera photo of load frame during furnace operation. Note the thermal camera is measuring the quartz surface temperature not sample/susceptor temperature.*

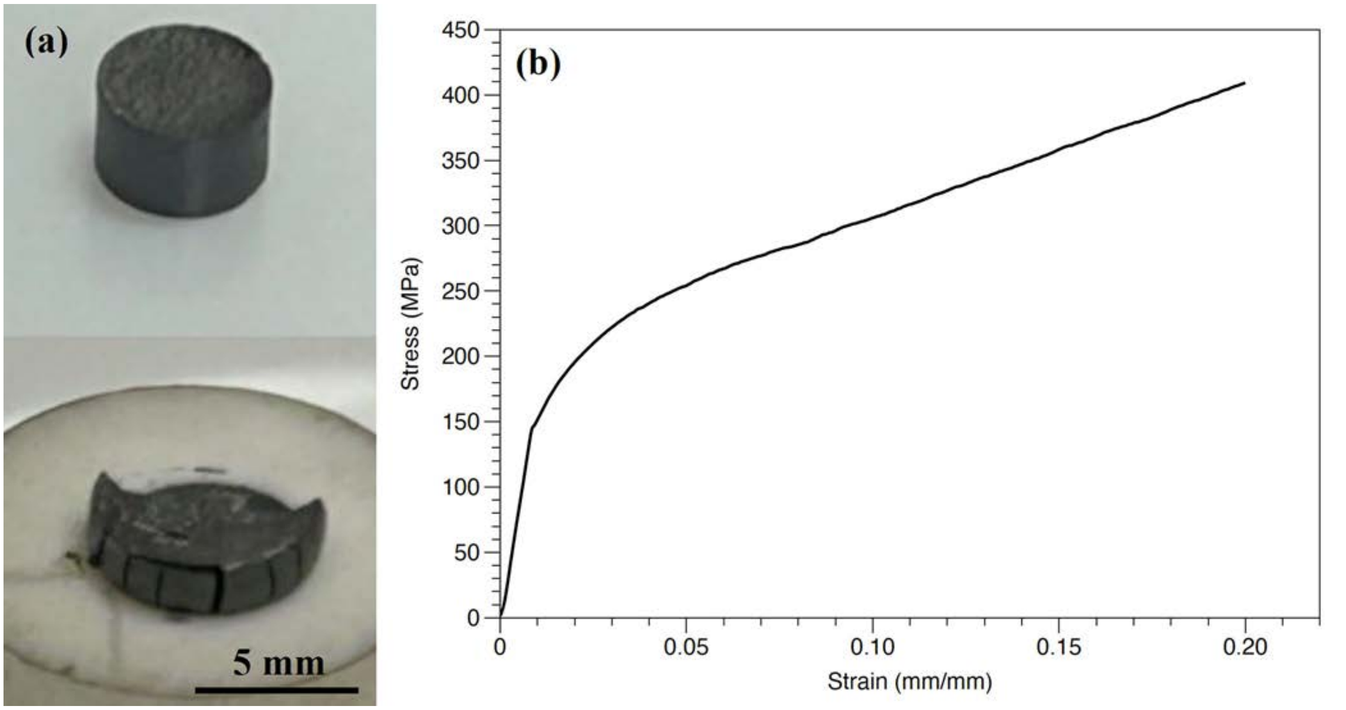



*Figure 3. (a) Side view of the pre- and post-creep test on $UO_2$ reference sample and (b) associated stress-strain response with a recorded pyrometer temperature of 950°C. It is noted that infrared absorption correction factors to the pyrometer temperature are being evaluated to correct the recorded temperature.*

In Figure 2a, the system is shown running with a setpoint of 950˚C, as registered by an external optical pyrometer. Oxygen sensors were installed at the inlet and outlet of the furnace with ultra-high purity, which measured <10 ppm $O_2$ during the entirety of the experiments, showing the purity of the gas environment. A sealed quartz tube contains the atmosphere. An internal $Al_2O_3$ tube supports a susceptor to couple with the coil midway up. In this case, a graphite susceptor was selected to run. A push rod is in contact with two alumina platens and a pellet that provides the load from the moving crosshead above through the push rod that feeds through an O-ring seal above. Figure 2b shows an external infrared (IR) camera that shows radiant heating, though only measures surface temperatures of the quartz tube due to light absorption in that spectrum. Current testing configuration has shown ramp rates on the order of 25 ˚C·s-1, which enables accident level ramp rates.

The stress strain response of $UO_2$ at a setpoint of 960 ˚C and associated representative pellet is shown in Figure 3. A defined elastic region and yield strength point can be observed and after the sample is deforming plasticly. The deformation occurred without deforming into fragments and the barreling of the sample also points to plastic deformation.

As plastic deformation of $UO_2$ is not seen at these temperatures it is likely that the pyrometer is not accurately calibrated for the emissivity of the susceptor. Work is on-going to change the single-color pyrometer to a dual-color pyrometer or use non-IR absorbing tubes (e.g., sapphire) to minimize errors in the temperature reading.

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# Cross-Cutting Advanced Ceramic Fuel Advances

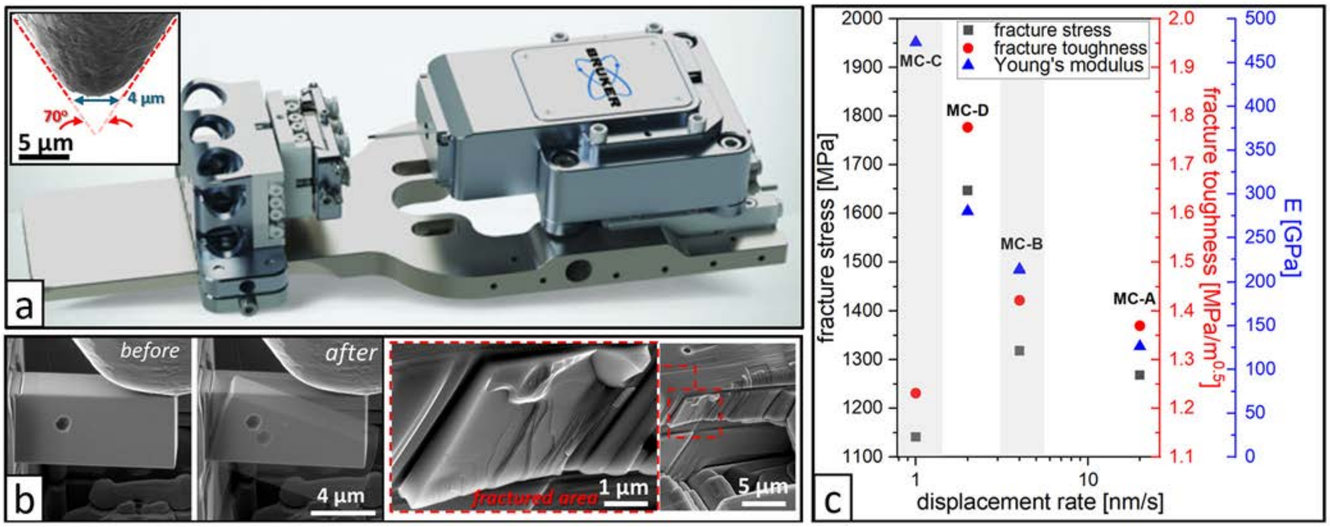
*Principal Investigator: Maria Kosmidou, Los Alamos National Laboratory (LANL)*
*Team Members/Collaborators: Nan Li, Erofili Kardoulaki (LANL); Andre Broussard, Kevin Yan, Nathaniel Cavanaugh, Jie Lian (All Rensselaer Polytechnic Institute [RPI]); Cihang Lu, Arantxa Cuadra (All Brookhaven National Laboratory [BNL]); Nicolas E. Stauff (Argonne National Laboratory [ANL])*

Advanced ceramic fuels offer significant advantages in nuclear reactors due to their high uranium density, excellent thermal conductivity, and superior melting points. These properties enhance fuel performance by allowing for higher operating temperatures and more efficient heat transfer. This work package aims to fill in critical data gaps, and in particular properties of Uranium Nitride (UN), an advanced ceramic fuel with high cross-cutting potential for multiple reactor application. The properties assessed here include creep deformation, strength, and fracture toughness under various stress and temperature conditions. Additionally, in this work we have assessed the neutronic performance of UN, specifically the relationship between N-15 and U-235 enrichment and how this affects in reactor performance.

## Project Description

This investigation focuses on two key areas:

1. Mechanical Property Assessment of UN: Experimental efforts include microcantilever (MC) bending tests at room temperature and bulk creep tests at elevated temperatures (1400–1500°C), conducted in collaboration with RPI. The MC bending tests reveal the material's ability to withstand mechanical stresses and resist cracking under operational conditions, ensuring the fuel's structural integrity and reliability throughout its life cycle. Creep tests provide essential data on how the fuel deforms over time under high temperature and stress conditions, helping to predict potential failures, ensure structural integrity, and maintain safe and efficient reactor performance. The study highlights the value of combining MC and bulk creep tests and calls for further research to address discrepancies in activation energy and enhance understanding of UN's behavior under extreme conditions, ultimately aiding in the optimization of nuclear fuel performance and safety.




2. Impact of UN Fuel Enrichment in Heat Pipe Microreactors (HPMRs): This part of the study, in collaboration with BNL, examines the effect of varying nitrogen-15 enrichment levels in UN fuel across two reactor designs: the Special Purpose Reactor (SPR) by LANL, representing fast-spectrum HPMRs with solid pellet fuels, and the HP-MR by ANL, a thermal-spectrum HPMR using TRi-structural-ISOtropic (TRISO) fuel compacts. The analysis includes reactor performance and safety characteristics such as excess reactivity, shutdown margins, reactivity temperature coefficients, and C-14 production, alongside econometric assessments of fuel costs. The study focused on microreactors as a potential cross-cutting reactor application since similar studies have already been conducted for light water reactors.

**Accomplishments**

Integrating cantilever bending and creep testing into the study of UN has yielded significant insights into its mechanical properties. Room temperature bending tests conducted at LANL, using a Hysitron/Bruker PI-89 pico-indent system (Figure 1a), have established key properties of UN and facilitated comparison with existing literature. By minimizing surface defects and utilizing a high-resolution nanoindenter with advanced imaging, subtle variations in two deformation and fracture properties are more accurately detected, improving on previous studies using Vicker's indentation. The bending test results align with expected values for the elastic modulus, confirming the reliability of this method. However, displacement rates below 2nm/s should be avoided to prevent significant errors. In-situ Scanning Electron Microscopy (SEM) images before and after testing

*Figure 1. (a) Hysitron/Bruker PI-89 pico-indent system used for MC bending tests. Inset shows the geometry of the conical indenter tip. (b) SEM micrographs for in-situ bending test and fracture surface of a cantilever. (c) Plot of fracture toughness and critical stress as a function of displacement rate for the tested beams.*

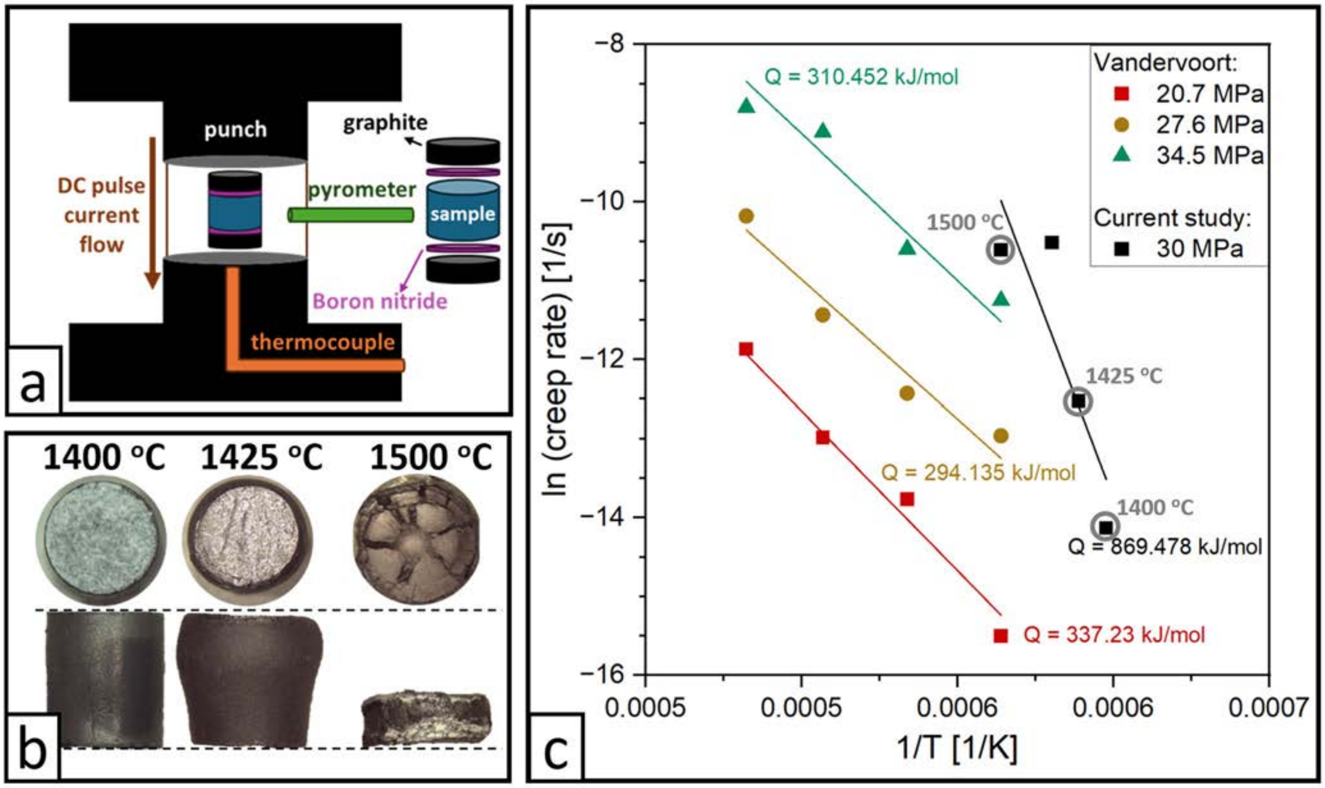



*Figure 2. (a) Schematic set up of the experimental creep testing with Spark Plasma Sintering, (b) Images of samples after creep testing performed at 1400°C, 1425°C, and 1500°C, (c) Activation energy deduced from creep measurements on UN.*

(Figure 1b), reveal a smooth, flat fracture surface, indicating brittle behavior. This is further supported by the load-displacement curves (Figure 1c), which confirm the lack of plastic deformation. Fracture toughness measurements from bending tests ranged between 1.3 and 1.88 MPa/m0.5, which is notably different from the broader range of up to 4 MPa/m0.5 reported with Vicker's indentation, highlighting the dependency on applied load. For $UO_2$ fuel, fracture toughness values from microcantilever bending and Vicker's methods were more consistent [1-6]. UN also exhibited higher fracture loads compared to $UO_2$ under similar displacement rates [1]. Bulk creep testing of sintered UN samples, developed at RPI, was conducted to evaluate creep behavior (Figure 2a). Creep tests revealed distinct temperature thresholds and pressure dependent behaviors during secondary creep, indicating different mechanisms critical for reactor performance.

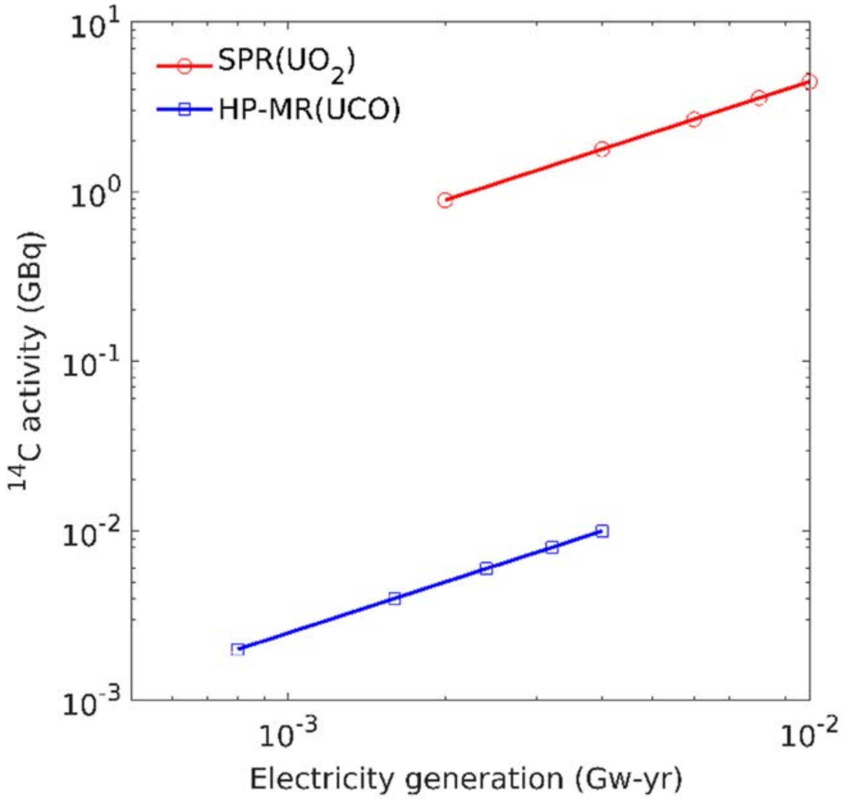



*Figure 3. Activity of the 14C buildup in the baseline SPR and HP-MR as a function of the generated electricity.*

The appearance of the pellets varies with temperature: at 1500°C, cracking is observed, while at 1425°C, the pellet's diameter increases to nearly 8 mm. At 1400°C, there is no significant change in shape or signs of compression. At 1500°C, pellets show significant compression and accelerated strain rates (tertiary creep), with height reduced from 7 mm to approximately 4 mm (Figure 2b). The stress exponent at 1500°C was measured at 7.97 under 30 MPa and 40 MPa, closely matching Vandervoort's value of 8.34 [7], as shown in Figure 2c. The lower average value of 4.5 suggested by Hayes's formula indicates that the stress exponent is sensitive to temperature variations [8]. Secondary creep thresholds were prolonged below 1450°C, while temperatures above this resulted in faster secondary creep at 30 MPa. This suggests that activation energy for creep is influenced by temperature-dependent behaviors and reported activation energies do not fully align with Hayes's estimates. The creep rates at 1500°C and 30 MPa matched literature findings, indicating that impurity levels have minimal impact on steady-state creep rates. Instead, microstructural changes like grain boundary gliding, along with factors such as grain size and porosity, play a significant role in deformation.

***This research is crucial for optimizing the performance of advanced ceramic fuels, and in consequence ceramic composites, for a variety of nuclear reactor applications by performing a comprehensive evaluation of their properties – the focus this year has been on mechanical and neutronic properties.***

Through neutronics calculations, the impacts of N-15 enrichment on C-14 production rate in UN-fueled HPMRs was investigated. Two microreactor designs are under consideration: the SPR, representing fast-spectrum HPMRs fueled with solid pellet fuels, and HP-MR, representing thermal-spectrum HPMRs fueled with TRISO fuel compacts. It was found that C-14 had a constant production rate throughout the lifetime of both MR designs, which can be negatively linearly correlated to the N-15 enrichment (Figure 3). HP-MR had a larger C-14 production rate than SPR with the same N-15 enrichment because of its softer neutron spectrum. Both HPMRs had a lower C-14 production rate than a typical PWR using UN fuels with 100 wt.% - enriched N-15, but a 420~750 times larger C-14 production rate if natural N (with a N-15 of 0.4 wt.%) is employed. Because HPMRs are sealed, the only potential pathway for HPMR produced C-14 to be released into the environment is during fuel reprocessing, which is not currently performed in the U.S. If the reprocessing of used HPMR fuels is sought, C-14-contaminated $CO_2$ from the reprocessing facilities needs to be treated properly to minimize or eliminate the releases of radioactive contaminants into the environment. This could involve either downstream through $CO_2$ capturing or upstream by using UN fuels with higher N-15 enrichments, both of which require more comprehensive studies on economic viability.

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# Accelerated Irradiation and Qualification of Ceramic Nuclear Fuels

*Principal Investigator: Scarlett Widgeon Paisner (Los Alamos National Laboratory [LANL])*
*Team Members/Collaborators: Joshua T. White, Meagan D. Wheeler (All LANL)*

Accelerated neutron irradiation testing is a component of accelerated qualification of new nuclear fuels for light water reactor, microreactors, and other special purpose reactors. The qualification and licensing of nuclear fuel is a lengthy process that can take 20-25 years to bring a new fuel into service. Accelerated fuel qualification combines both experimental and modeling work to expedite the total qualification time to 5-10 years' timeframe. The experimental aspect of this is accelerated irradiation aims to reduce the total time needed for neutron irradiation to achieve targeted burnup, which can take years using conventional irradiation profiles. The data that results from this irradiation testing can then be entered into BISON models to develop robust and reliable performance simulations to ensure safe operation under normal and off normal conditions. This milestone focused on the fabrication of test articles for accelerated irradiation testing at the Advance Test Reactor (ATR).

## Project Description

In order to implement accelerated fuel qualification, the experimental efforts must be tested and verified by first being carried out on a well know fuel, such as standard grain $UO_2$. Currently, irradiation tests are proposed at ATR, which provide high neutron fluxes to reduce the irradiation time. The test articles that were defined under this work package were standard grain $UO_2$, along with medium and large grain undoped $UO_2$. These samples will provide information regarding the fission gas release of standard grain $UO_2$ that can be compared with traditional irradiation testing for $UO_2$. The medium and large grain $UO_2$ test articles will provide additional information to elucidate the effect of grain size on the fission gas release. Large grain, $Cr_2O_3$-doped $UO_2$ was also of interest to compare with large grain, undoped $UO_2$. Several studies have reported the enhancement in grain size helps, which is proposed to retain fission gases during irradiation. Alternatively, there have also been modeling studies which suggest that Cr dopants can lead to increased

***Large grain, Cr-doped and undoped (standard, medium, and large grain) $UO_2$ pellets have been fabricated for Fission Accelerated Steady-state Test irradiations to investigate and separate the effects of grain size and dopant on the fission gas release under irradiation.***

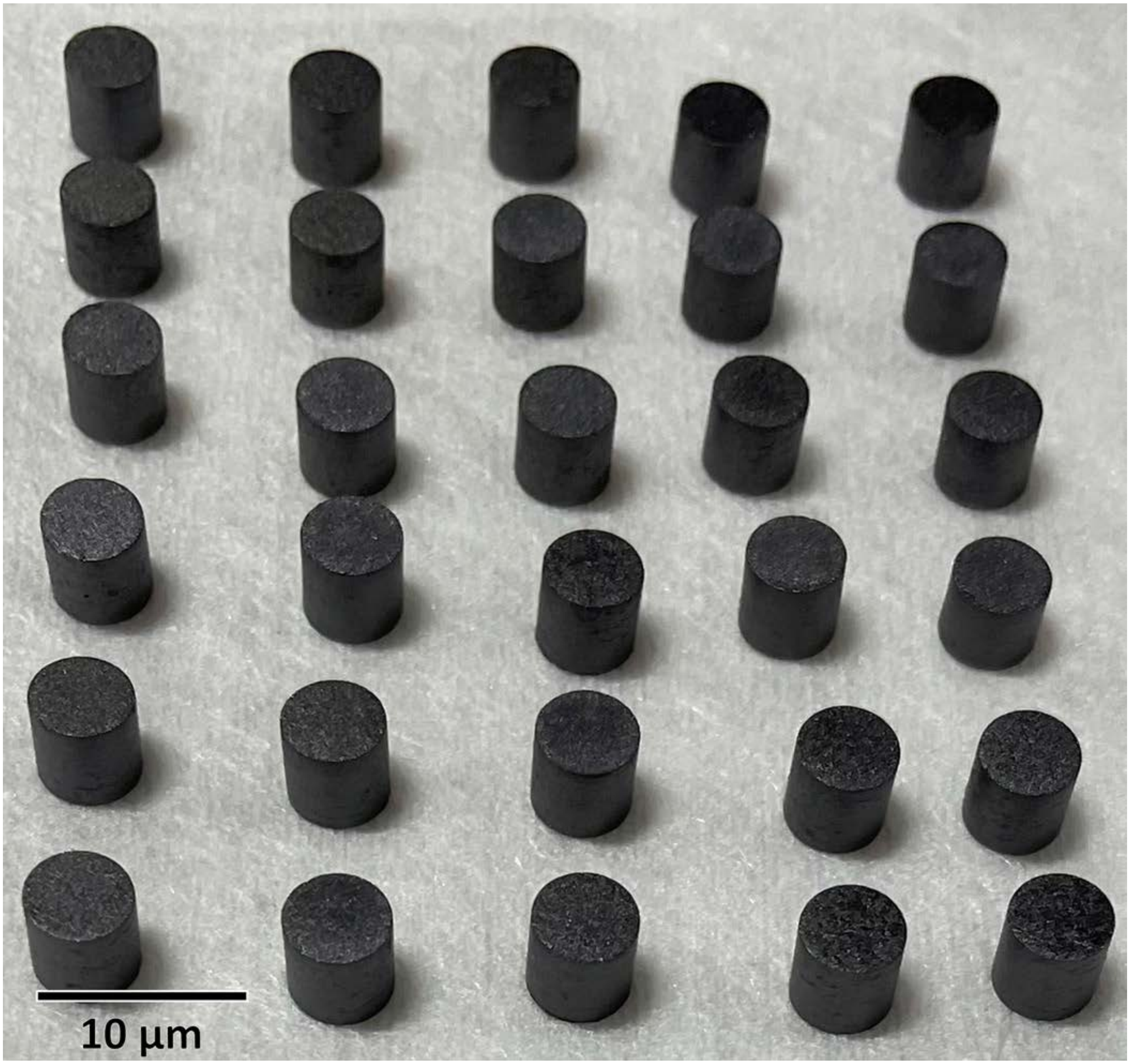



*Figure 1. Image of 30 large grain undoped $UO_2$ pellets after sintering.*

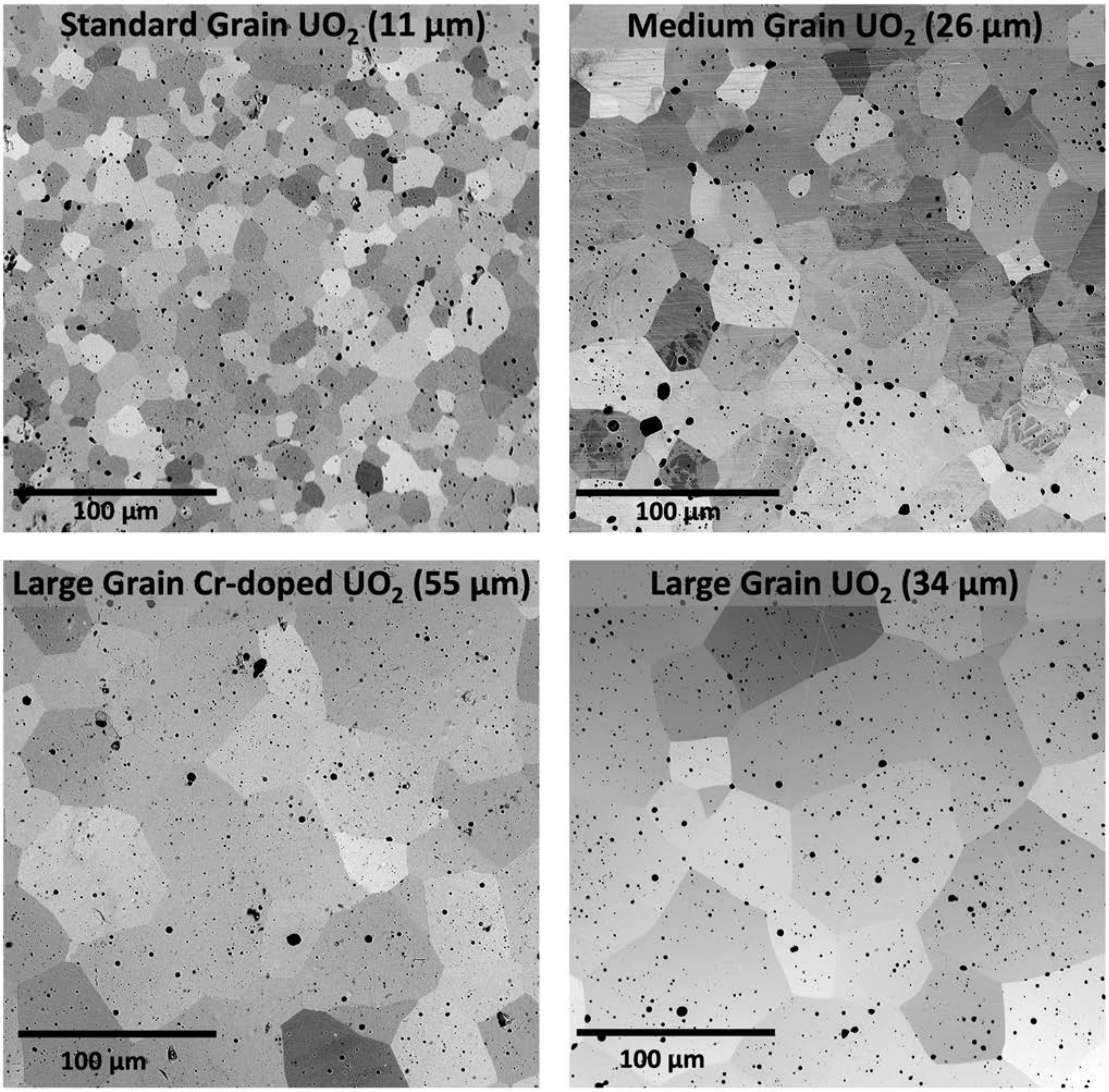



*Figure 2. Scanning electron microscopy micrographs of standard, medium, and large grain undoped $UO_2$ and large grain Cr-doped $UO_2$ showing grain size variation of the four $UO_2$ variants.*

diffusivity of such fission gases. The data that will be generated from the irradiation of these $UO_2$ variants will yield important information regarding fission gas retention as a function of grain size and the effect that Cr dopants have on diffusivity and release of fission gases. In addition, it will provide information about the validity of accelerated irradiation testing relative to traditional testing.

## Accomplishments

The goals for this work package focused on the fabrication of four $UO_2$ variants; undoped $UO_2$ (standard, medium, and large grain sizes) and large grain, Cr-doped $UO_2$. Approximately 30 pellets for each of the four $UO_2$ variants were fabricated in fiscal year 2024 using 19.9 % 235U $UO_2$ feedstock. The control of grain sizes was carried out by careful testing and examination of the oxygen potential during sintering of the fuel specimen. This testing included optimization of the grain sizes, pellet geometries, and density, which was first established using depleted $UO_2$, then transferred to high assay low enriched uranium $UO_2$. After significant testing, the final pellets were sintered and met the specified grain sizes. Figure 1 shows all 30 pellets for the large grain $UO_2$ variant after fabrication. The geometries and densities, as well as the average grain sizes were measured. The average linear grain sizes were determined on sintered pellets by analysis of scanning electron microscopy images, which is shown in Figure 2. The average grain sizes are indicated on the images, and show that the standard, medium, and large grain undoped $UO_2$ pellets have average grain sizes of 10.9, 25.6, and 33.9 mm, respectively, while the Cr-doped $UO_2$ samples have an average grain size of 54.5 mm.

This variation in grain sizes of the four variants will be important in the fission gas release data of the irradiated samples to determine how the grain size and inclusion of Cr dopants influenced this.

The feedstock powder and sintered pellets were characterized in anticipation of shipment to Idaho National Laboratory (INL) for planning of the insertion into ATR. The insertion date has not been identified at this time. The results have been documented in the L2 milestone report (M2FT-24LA020201031). This includes characterization of the oxygen to metal ratio, isotopic and impurity analysis, density, dimensions, and grain size. The shipment from LANL to INL is anticipated to occur in the first quarter of 2025.

# Internal Pressure Testing of Hydrided and Coated Cladding

*Principal Investigator: Aaron Colldeweih (Idaho National Laboratory [INL])*
*Team Members/Collaborators: David Kamerman, Malachi Nelson (INL); Caleb Massey, Nathan Capps, Mackenzie Ridley (Oak Ridge National Laboratory [ORNL])*

In collaboration, INL and ORNL have completed internal pressurization tests on cladding at 300°C to induce biaxial loading conditions representative of load following conditions in a reactor core. Accident tolerant fuel (ATF) claddings, namely Physical Vapor Deposition (PVD) Chromium (Cr) coated Zircaloy-4 were internally pressurized and subsequently characterized at INL while further testing on strained cladding was performed at the Severe Accident Test Station (SATS) at ORNL. Additionally, a study on the effectiveness of the strained coating against hydrogen uptake was performed.

### Project Description

The objectives of the research were to investigate the response of ATF Cr-coated claddings to load following conditions where elevated cladding strain may occur due to increased pellet cladding interaction. With increased electricity demands, load following becomes a more relevant procedure for nuclear power plants. The understanding of ATF claddings is primarily limited to accident conditions where the cladding starts from a pristine condition. However, the state of the cladding and coating may change during operations. Namely, load following conditions will result in increased strain on the cladding and potentially cause flaws in a Cr-coating.

A better understanding of the coating response to elevated strain levels will provide insight into the protectiveness of the coating against corrosion and hydrogen uptake. Subsequently, understanding the coating mechanical response to elevated strain will provide insight into the effectiveness and behavior of the cladding system during an accident condition. This insight of the cladding system will improve the understanding of the overall safety benefits of ATF cladding that contains potential flaws. Additionally,

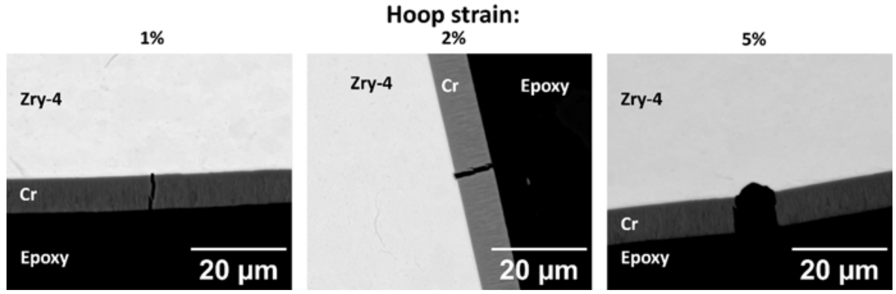
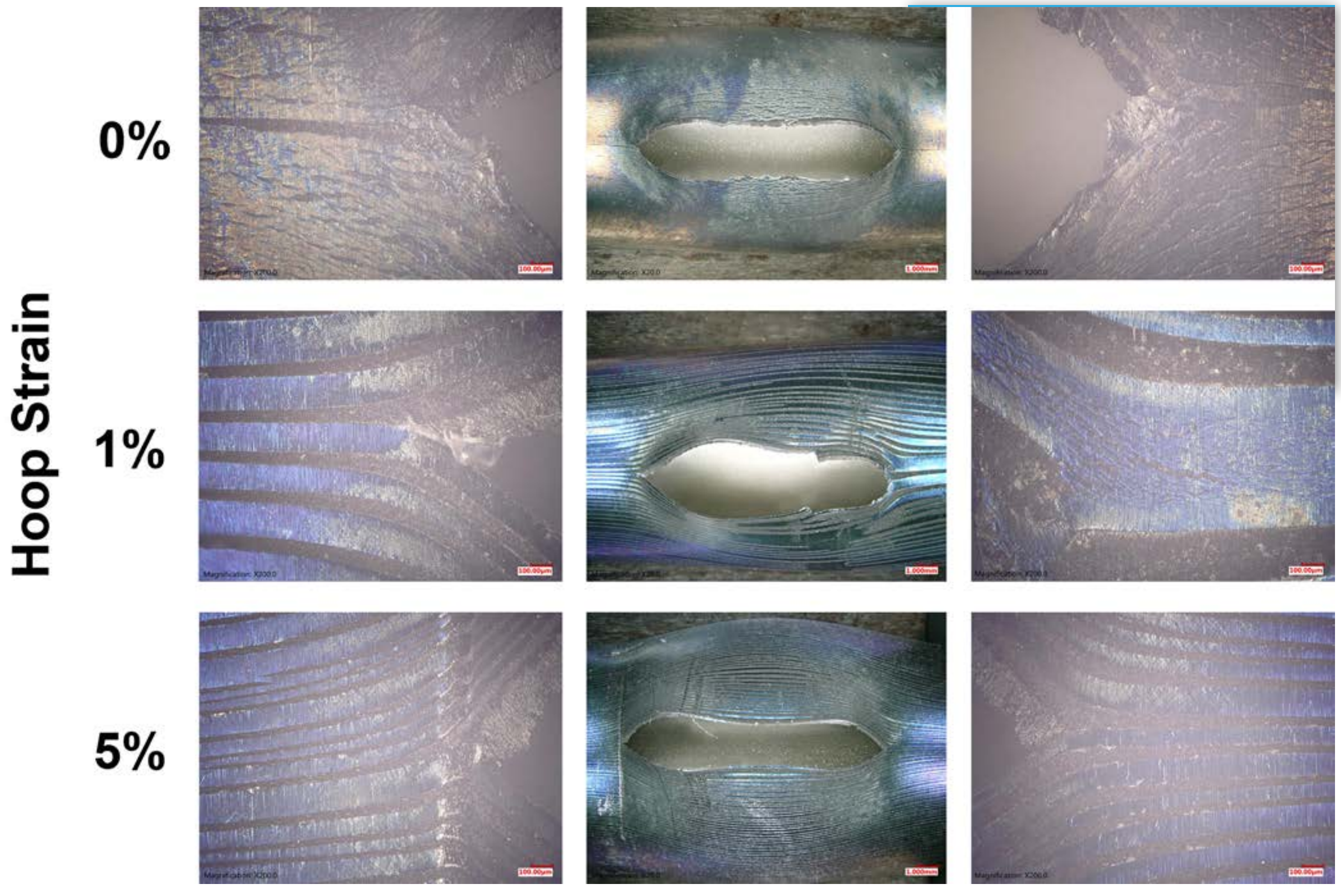



*Figure 1. Representative images of the coating appearance after varying levels of hoop strain (1, 2, and 5%). The cracking propagates through the coating thickness and arrests in the Zircaloy-4 cladding.*




knowledge of the protectiveness of a strained Cr-coating against hydrogen uptake will improve the decision-making of extending fuel lifetime. As the coating acts as a barrier against hydrogen uptake, the effectiveness of the coating may be altered when uptake pathways, such as cracks, develop.

## Accomplishments

The objective of the research was to determine how Cr-coated rods respond to varying amounts of hoop strains that are representative of strain levels during load following conditions. Subsequently, SATS testing was performed to determine how the pre-strained coating would respond to accident conditions. The results from internal pressurization tests showed that the PVD Cr-coated Zircaloy-4 tubes cracked in a through-coating thickness manner along the columnar grains of the coating (Figure 1). The crack density in the coating increased with increased hoop strain levels. However, the crack density did not increase linearly with strain. Characterization showed the average crack width increased with hoop strain. Given the increased average crack width, it is suspected that local strains in the cladding have developed at crack tips at the coating-cladding interface. While the majority of cracks extend though the thickness of the Cr-coating,

*Figure 2. Visual inspections of the burst openings of reference and pre-strain claddings tested in SATS at ORNL. The as-received materials has varying deformation pathways compared to the pre-strained cladding.*

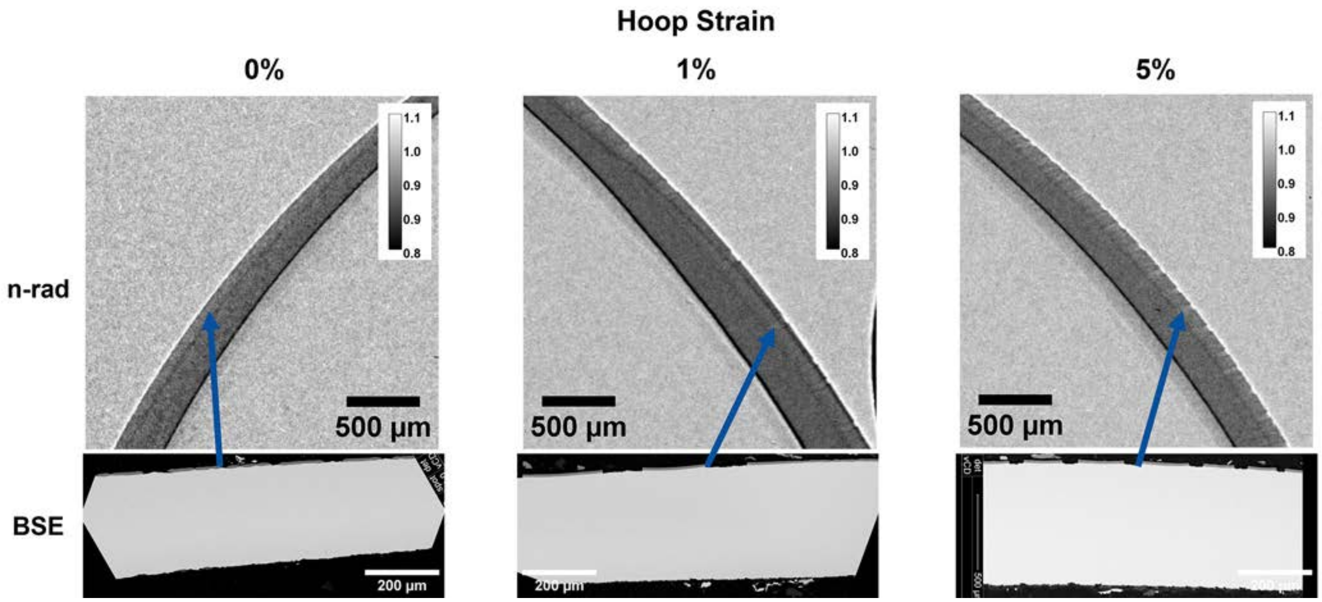
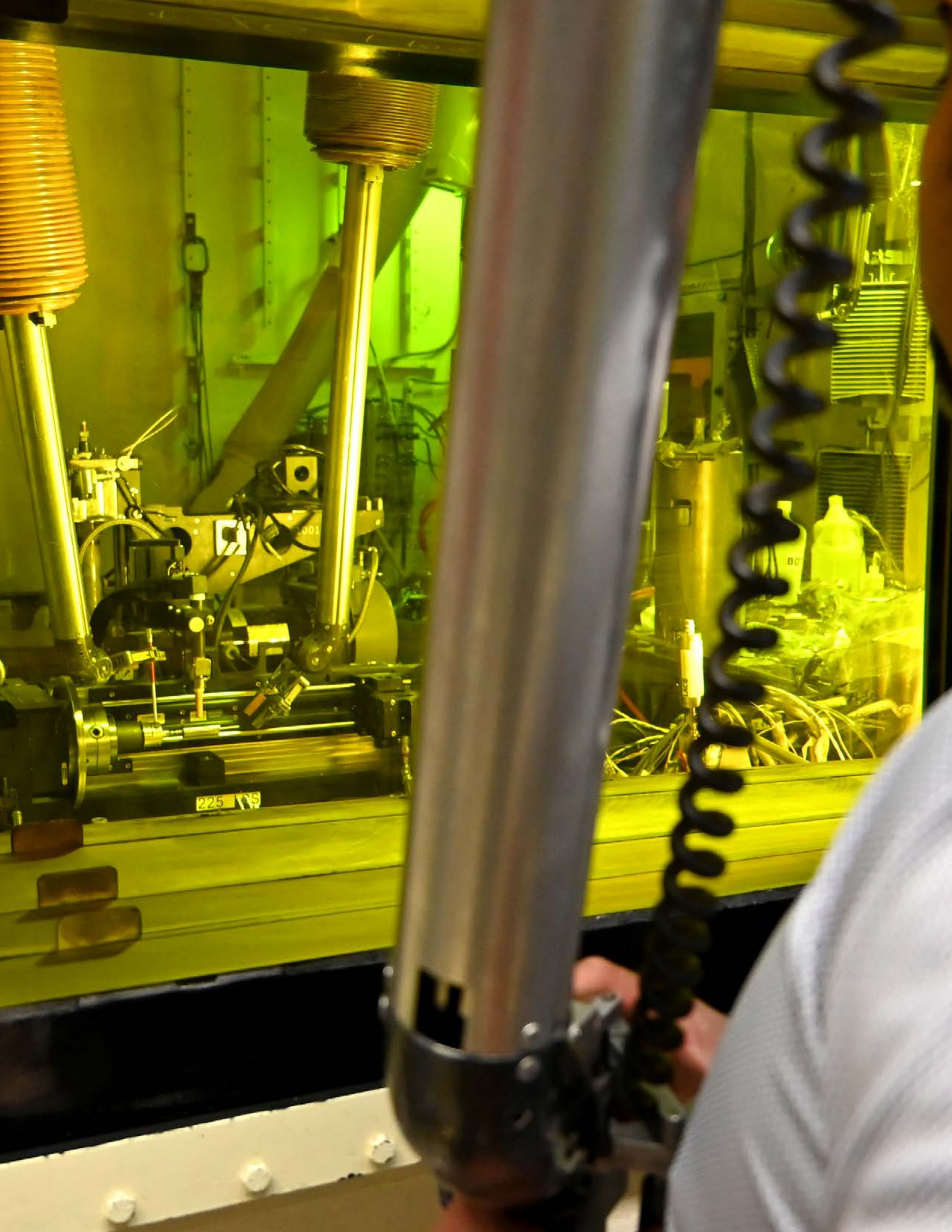



*Figure 3. Advanced characterization of the radial cross-section of the burst opening using neutron radiography and back-scatter electron (BSE) microscopy. The BSE images show the varying deformation of the coating between as-received and pre-strained conditions.*

they arrest within a few microns of coating-cladding interface in the cladding. Strained cladding was also subjected to hydrogen charging with known procedures for effectively charging cladding. The analysis using hot gas vacuum extraction and neutron radiography showed that the cracked coating was still effective at protecting the cladding against hydrogen uptake.

SATS tests showed that the deformation in the Cr-coating varied between the pre-strained coating and as-received coating. The pre-strained coating showed to further deform at the previously cracked locations while the as-received material showed plastic deformation in the Cr-coating (Figures 2 and 3). More specifically, the pre-strained coating did not exhibit plastic deformation such as necking, likely due to the lack of constraints caused by the cracks. The as-received material showed severe plastic elongation, specifically near the burst opening. The mechanical response, however, did not show any early burst from the pre-strained cladding tubes.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# Hydrogen Effects on Cladding Performance during Simulated Loss of Coolant Accident Testing

*Principal Investigator: Mackenzie Ridley (Oak Ridge National Laboratory [ORNL])*
*Team Members/Collaborators: Samuel Bell, Ben Garrison, M. Nedim Cinbiz, Nathan Capps (All ORNL)*

***Digital image correlation has identified evidence of both decreased creep strength at intermediate temperatures and decreased ductility at high temperatures for fuel cladding with elevated hydrogen concentrations during a simulated loss of coolant accident.***

Milestone report M2FT-24OR020202041 discusses the impact of hydrogen on the cladding rupture performance during simulated Loss of Coolant Accident (LOCA) testing as an effort to improve understanding of high burnup fuel cladding mechanical properties during postulated accidents. Unirradiated Zircaloy-4 cladding segments were charged with hydrogen to various concentrations prior to LOCA testing. The cladding rupture temperature decreased by 10 °C for every 100 wppm H added into the cladding. Hydrogen charged cladding also showed sharp rupture interfaces with evidence of decreased ductility compared to baseline material performance. Digital image correlation was employed to showcase hydrided cladding deformation behavior in situ, and the initial results provide a framework for materials separate effects assessment under transient conditions.

## Project Description

Targets for higher burnup fuels and fuel cladding can improve reactor efficiencies and decrease total spent fuel for the U.S. light water reactor fleet. Yet, a known challenge with high burnup fuel cladding is the increased propensity for hydrogen uptake from waterside corrosion. Hydrogen in fuel cladding can precipitate as zirconium hydride, which deteriorates the cladding mechanical properties and is often considered with fuel handling and storage. While modern fuel cladding are tailored to minimize hydrogen pickup in the cladding wall, there are still possibilities for hydrogen pickup that must be considered. First, higher burnup fuels have a higher propensity to fragment, and fuel fragmentation and relocation within the cladding can result in localized hot and cold spots. In most cases, hydrogen pickup from corrosion will occur at the highest temperature cladding region, yet hydrogen can diffuse and precipitate in the colder regions over time. A framework for assessing hydrogen impacts on LOCA rupture behavior was established in this work.

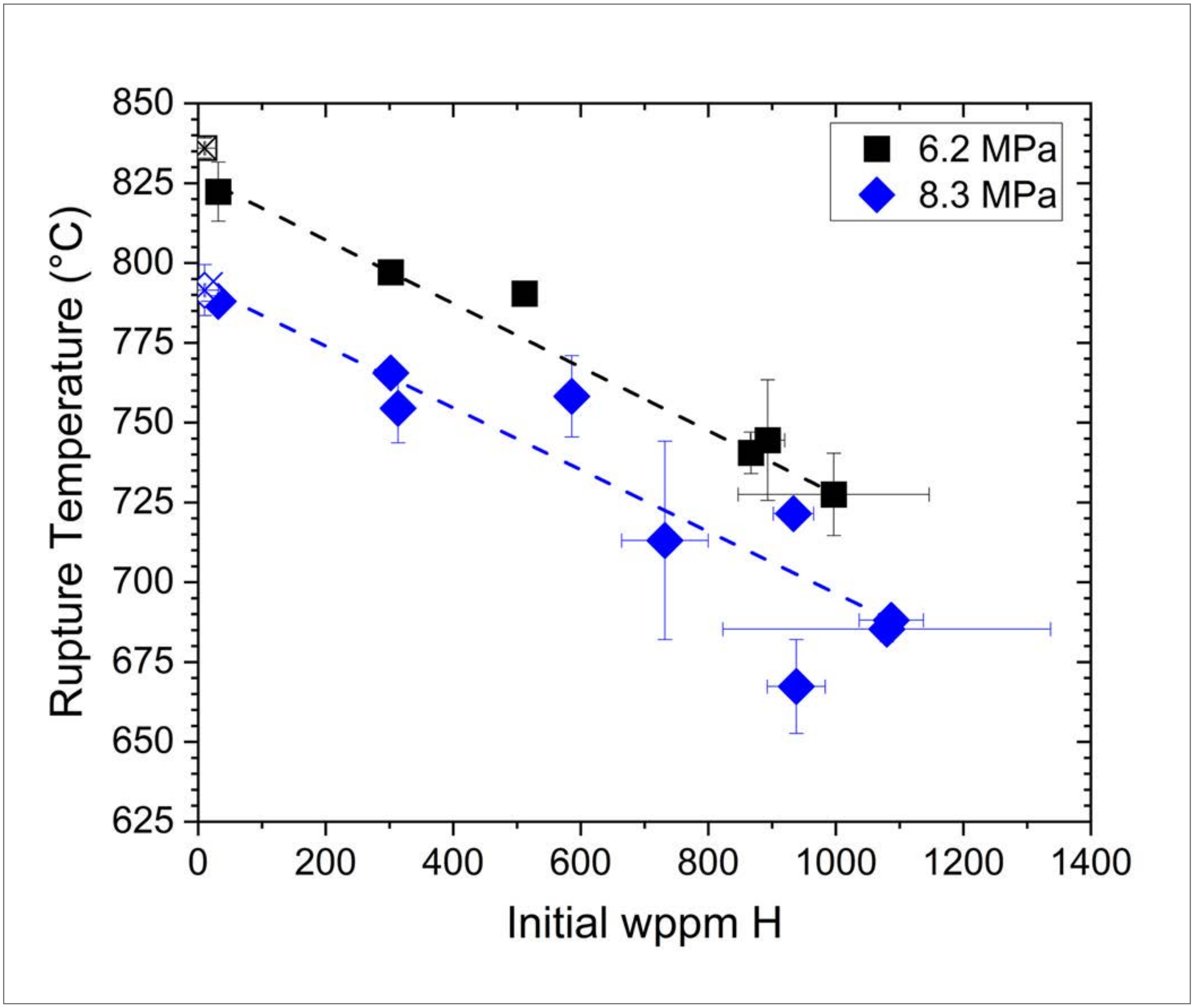
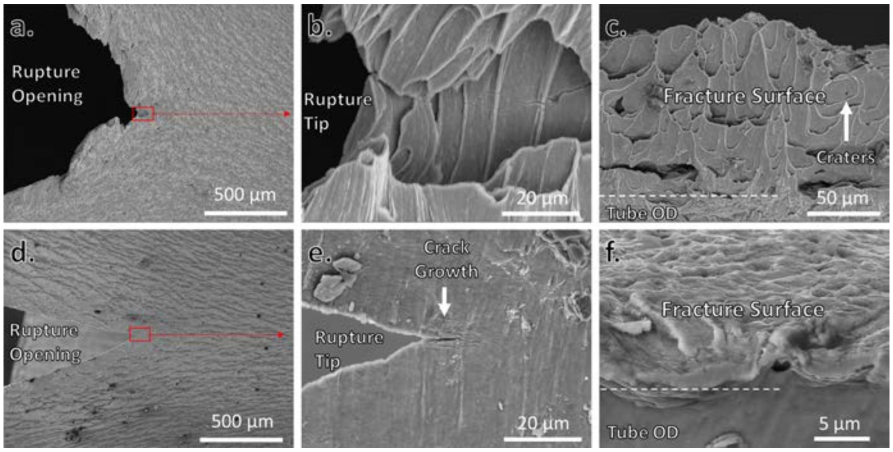



*Figure 1. Zircaloy-4 rupture temperature for 6.2 MPa and 8.3 MPa LOCA testing as a function of initial hydrogen concentration.*



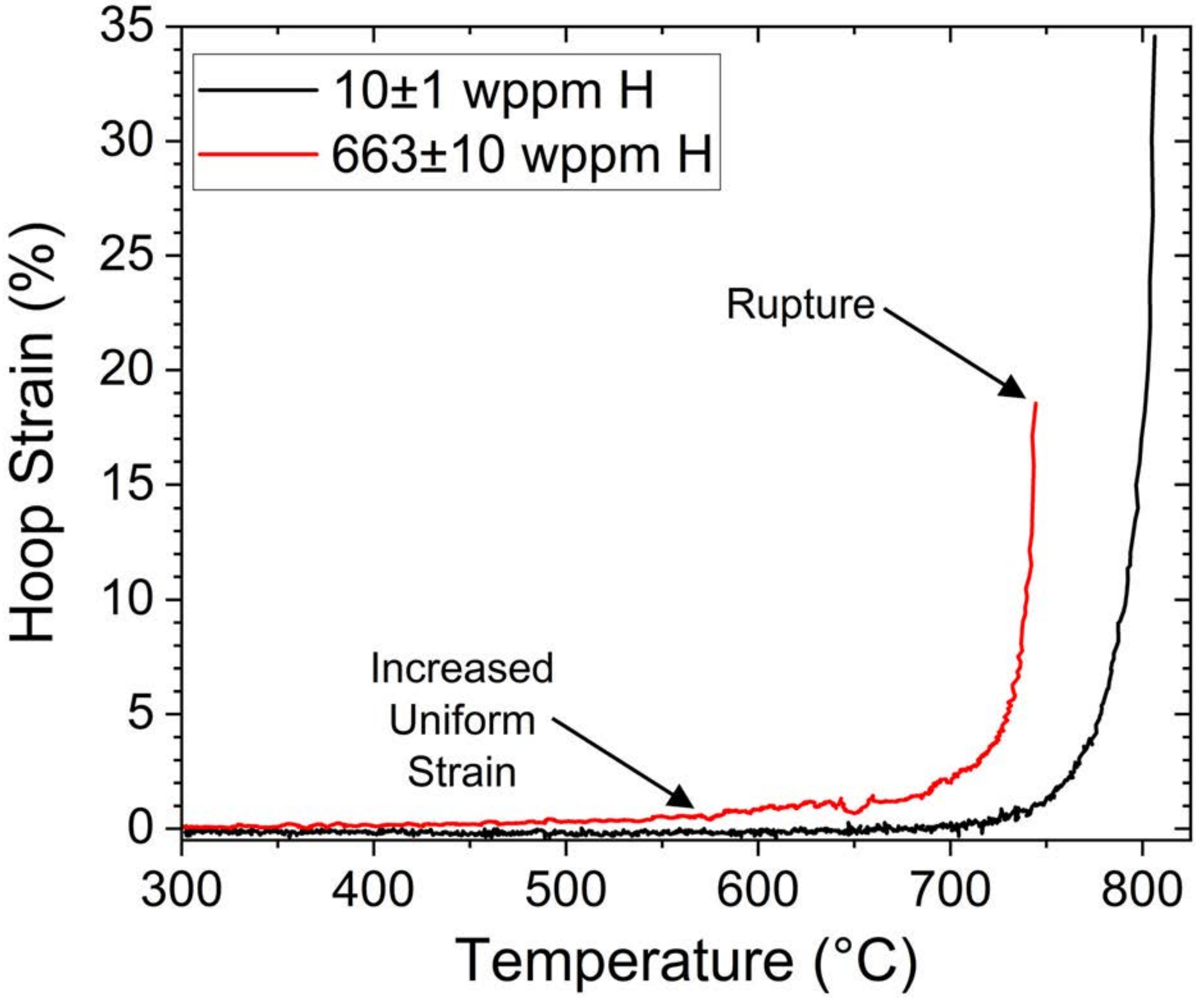

*Scanning electron microscopy of cladding rupture openings and fracture surfaces after 8.3 MPa LOCA testing: (a, b, c) as-received Zircaloy-4, and (d, e, f) Zircaloy-4 charged with 867 wppm H. Both image c) and image f) show fracture surfaces at the center of the rupture opening.*

This work attempts to provide knowledge regarding the impacts of hydrogen on Zirconium alloy cladding performance during unexpected reactor conditions, such as a LOCA. The zirconium – hydrogen phase diagram showcases a strong temperature dependence for the solubility limit of hydrogen, which suggests that there would be both dissolution and precipitation of zirconium hydride phases within fuel cladding during a LOCA transient. This work involves establishment of a framework for understanding relationships between equilibrium phase diagrams, transient accident testing, and mechanical properties for zirconium alloys with elevated hydrogen concentrations. The first objective of this work was to develop hydrogen charged fuel cladding segments with uniform hydrogen concentrations. Then, LOCA tests were performed with 6.3 or 8.2 MPa internal pressures and a 5°C/s heating rate. Upon cladding rupture, the segments were quenched to retain the high-temperature rupture microstructure. Characterization was performed to assess relationships between rupture geometry and initial hydrogen concentration. In situ deformation was measured for the first time on a hydrogen charged Zircaloy-4 cladding segment during a simulated LOCA event.

### Accomplishments

Unirradiated Zircaloy-4 claddings were charged with hydrogen from 300 – 1200 wppm Hydrogen and subjected to simulated LOCA conditions to assess the impact of hydrogen on the cladding rupture performance. Two internal cladding pressurization cases were completed: 6.2 and 8.3 MPa. The cladding rupture temperature decreased by 10 °C for every 100 wppm H added into the system for both internal pressurization cases (Figure 1). Post-test analysis such as in Figure 2 showed a loss of ductility with increasing hydrogen content, evident through decreased balloon sizes and sharper rupture interfaces. A peer reviewed publication is underway to discuss the impacts of hydrogen on LOCA rupture performance of unirradiated fuel cladding.

Digital image correlation was used to map the cladding strain in situ for two cladding segments. A small region of interest that highlighted the balloon location was used for hoop direction strain analysis on both a baseline and hydrogen charged cladding segment, results shown in Figure 3 as a function of cladding temperature. An increased intermediate temperature uniform




strain on the order of 1-2% was measured for the hydride cladding, followed by decreased ductility during cladding ballooning and decreased final rupture temperature. The increased uniform strain during the middle stage of the LOCA heating profile suggests that soluble hydrogen, zirconium hydride, or a combination of the two archetypes decrease the creep strength of the cladding. The variable thermal stabilities of Zr alloys and Zr Hydride as a function of temperature and hydrogen concentration are thus likely responsible for the uniform deformation behavior at intermediate temperatures. Additional hydrogen charged cladding LOCA experiments with digital image correlation are needed to fully elucidate initial findings and to quantify the impact of hydrogen on thermal creep of zirconium alloys at high temperatures.

*Figure 3. Hoop strain measured by digital image correlation near the rupture area for both a high and a low hydrogen concentration Zircaloy-4 cladding segment.*

# Quantifying Residual Stresses in Cr-coated Cladding

*Principal Investigator: Tim Graening (Oak Ridge National Laboratory [ORNL])*
*Team Members/Collaborators: Mackenzie Ridley, Jesse Werden, Kory Linton, Nathan Capps (All ORNL)*

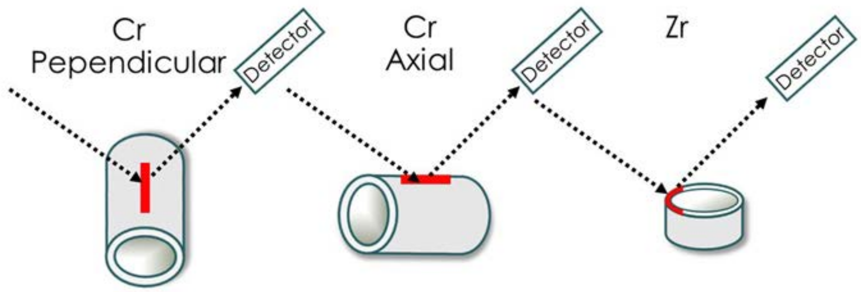



*Figure 1. Orientation of tested tube samples for residual stress measurements.*

Accident-tolerant fuel (ATF) concepts for light-water reactor applications have been developed and tested in diverse research programs around the world. Industry teams have developed coated cladding concepts using different coating application methods to achieve various thicknesses. The microstructure and mechanical properties of Cr-coated Zircaloy-4, were investigated as coated condition, and the results suggest an increase in performance in an unirradiated state. However, it remains unclear what phenomenon drove this performance. One explanation to this performance is associated with the coating process imparting a residual stress. Residual stress analysis was performed on coatings produced with various process parameters to understand how a thin layer of Cr impacts the mechanical properties after coating application and after simulated operating conditions. The gathered information is intended to address safety concerns or support margin identification related to ATFs and to support subsequent modeling and simulation efforts.

## Project Description

As an ATF concept, Cr coatings on Zircaloy-4 were produced via physical vapor deposition with a thickness of 6 to 9 microns. Several different processing parameters were investigated regarding the introduced residual stress using X-ray diffraction methods to provide understanding how the initial processing impacts performance in operation. Simulating operating conditions, tubes were exposed in prototypic pressurized water reactor conditions in the absence of irradiation. Additionally, to investigate the effect of a damaged coating on the overall performance, a laser engraving method was developed to reproducibly remove parts of the coating without altering the bulk material. The tubes were welded shut before they were introduced in the autoclave for testing. The heat affected area was determined

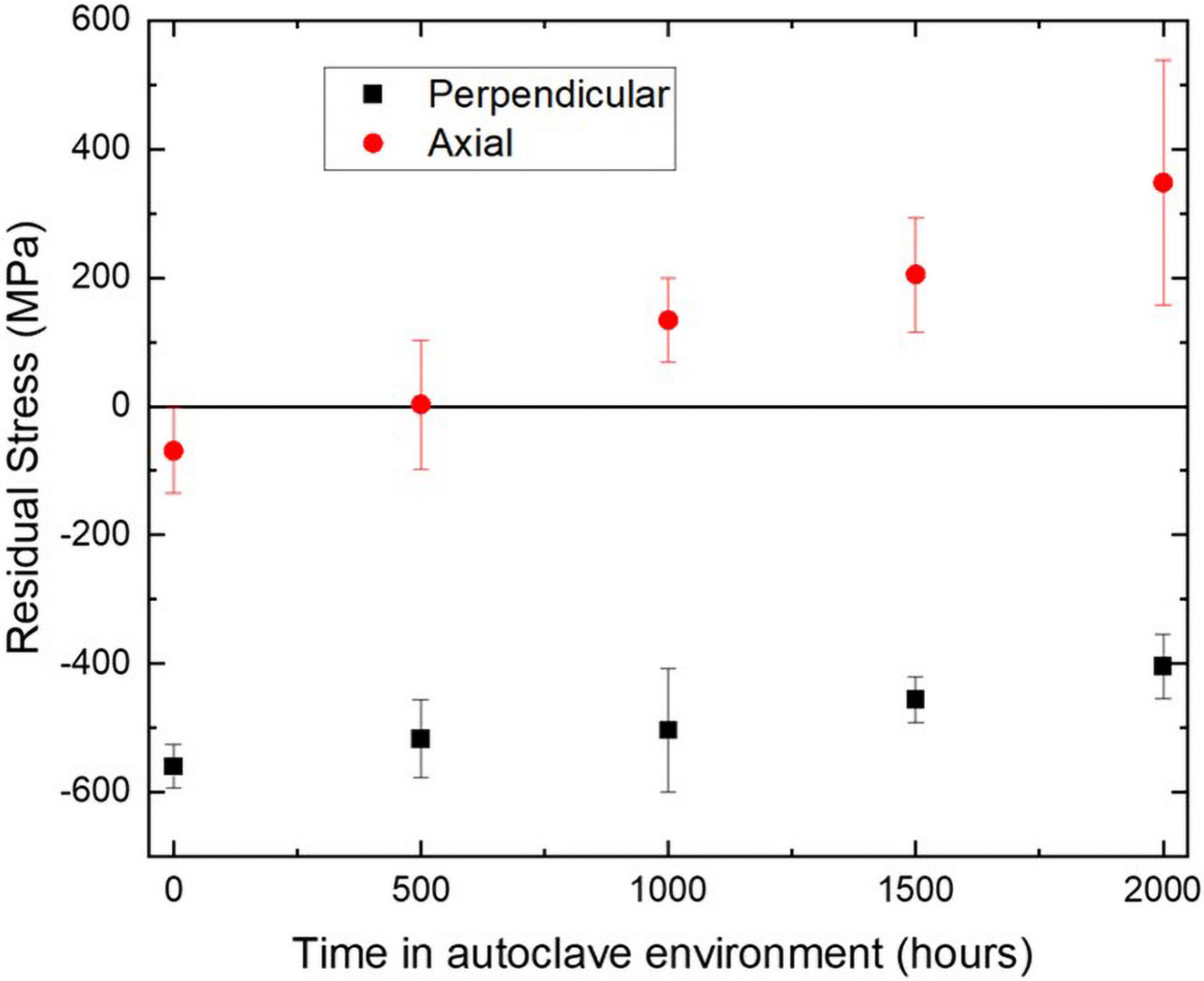



*Figure 2. Determined residual stress in perpendicular and axial measuring condition.*

beforehand to exclude that area from microstructure characterization. Each sample condition was exposed to 500, 1000, 1500, and 2000 hours in a 316 SS basket at 15.6 MPa and 330°C. Continuous flow deionized water was filtered to remove residual impurities, and hydrogen gas was introduced to the water supply to achieve a measured dissolved oxygen concentration below 5 parts per billion. X-ray and microstructure characterization was performed afterwards in two different orientations on the surface of the tubes as shown in Figure 1.

### Accomplishments

Understanding the effects of coating application parameters and simulated operating conditions on the Cr-coating is required to predict how a thin coating affects the bulk properties of the cladding material. Here, residual stress introduced during the coating application process on cladding tubes is investigated. That information can inform simulation efforts, when microstructural and mechanical data are combined to perform lifetime and accident scenario predictions.

Here, we conducted a first step in that direction by using the coated material batches produced over several years in a comparative study. Residual stresses are the sum of elastic and plastic deformation of crystallites across multiple scales: (1) macroscopic, where stresses are averaged across multiple grains (σI), (2) mesoscopic, as the average stress inside a single grain (σII), and (3) microscopic, as the fluctuation of stress inside a single grain (σIII). Mechanical properties are defined mostly by the macroscopic stresses, which can get evaluated using x-rays analysis assuming a uniform and isotropic distribution of crystals. Stresses inside a sample lead to changes of the interplane distances dhkl, which depends on the orientation of the grain inside the polycrystal. In general, a clear trend for all coated samples was found: Residual stress in the axial direction was found to be slightly compressive in the range of 20 to 170 MPa, while stress values of -500 to -620 MPa were calculated from the sin2ψ method. The samples which were exposed to the simulated pressurized water conditions for extended times also show a clear trend, which is highlighted in Figure 2. During extended exposure, the perpendicular compressive stress reduces, while the axial stress becomes a tensile stress within the coating, clearly indicating a change

*The surprising mechanical testing results of Cr-coated materials can now be better understood by providing residual stress as an input parameter for simulation efforts.*

within the bulk material. The results show that the annealing process reduced the large compressive stress value in one measurement direction, alleviated by a slight showing of a tensile stress in the perpendicular direction. These reported values can be directly employed in simulation efforts for residual stress values $\sigma 11$ and $\sigma 22$. Due to those results, burst tests of an annealed sample of a coated material were performed to investigate the reduction of compressive stress impact on the mechanical properties. Previously, burst tests of unannealed coated sampled showed an increase in burst temperature. However, it was found that annealed samples did not show any difference between coated and uncoated material. This behavior challenges the notion that residual stress, as influential as it certainly on the interface and cladding material, has an insignificant contribution under effects such as neutron irradiation and long-term annealing under service conditions. These results push for a detailed study on neutron-irradiated material to help elucidate the interconnected mechanisms.

# Stress Relaxation Experiments of Cr-Coated Zr to Support Modeling

*Principal Investigator: Benjamin Eftink (Los Alamos National Laboratory [LANL])*
*Team Members/Collaborators: T.M. Kelsy Green (Formerly LANL, now Antares Industries); Aditya Shivprasad (Formerly LANL, now Electric Power Institute [EPRI]); Tim Graening (Oak Ridge National Laboratory [ORNL]); Carl Cady, Tarik Saleh, Laurent Capolungo (LANL)*

***The mechanical tests will provide critically valuable data to the NEAMS program, modeling and simulation efforts are currently under way with the grand objective to capture the mechanical response of Zircaloy-4 under complex transient loading scenarios as a function of the microstructure of the material.***

Predicting the creep behavior of Zirconium based cladding has important implications to performance. Models are being developed which require experimental data for validation, in the case of Zircaloy, during dislocation activity and relaxation. In this study, Zircaloy-4 with a Cr coating is investigated during stress-relaxation tests in an operating and an off-normal temperature regime.

## Project Description

The objective of this research is to determine both the stress-relaxation response and the Cr coatings response to a strain at temperature for Cr coated Zircaloy nuclear fuel cladding. Stress relaxation rates were investigated at both operating and off-normal temperature regimes. Microstructure will be different at the two conditions, cold-worked stress relieved at the lower temperature and fully recrystallized at the higher temperature. The mechanical tests will provide critically valuable data to the Nuclear Energy Advanced Modeling and Simulation (NEAMS) program. Indeed, modeling and simulation efforts are currently underway with the grand objective to capture the mechanical response of Zircaloy-4 under complex transient loading scenarios as a function of the fingerprint of the material. During these transients, plastic flow is mainly accommodated by dislocation activity and relaxation. Both the rate of relaxation and the amount of relaxation and their overall temperature and stress dependence are therefore critical to model validation efforts in NEAMS. Better predictive capabilities result in safer, more reliable and economic reactors.

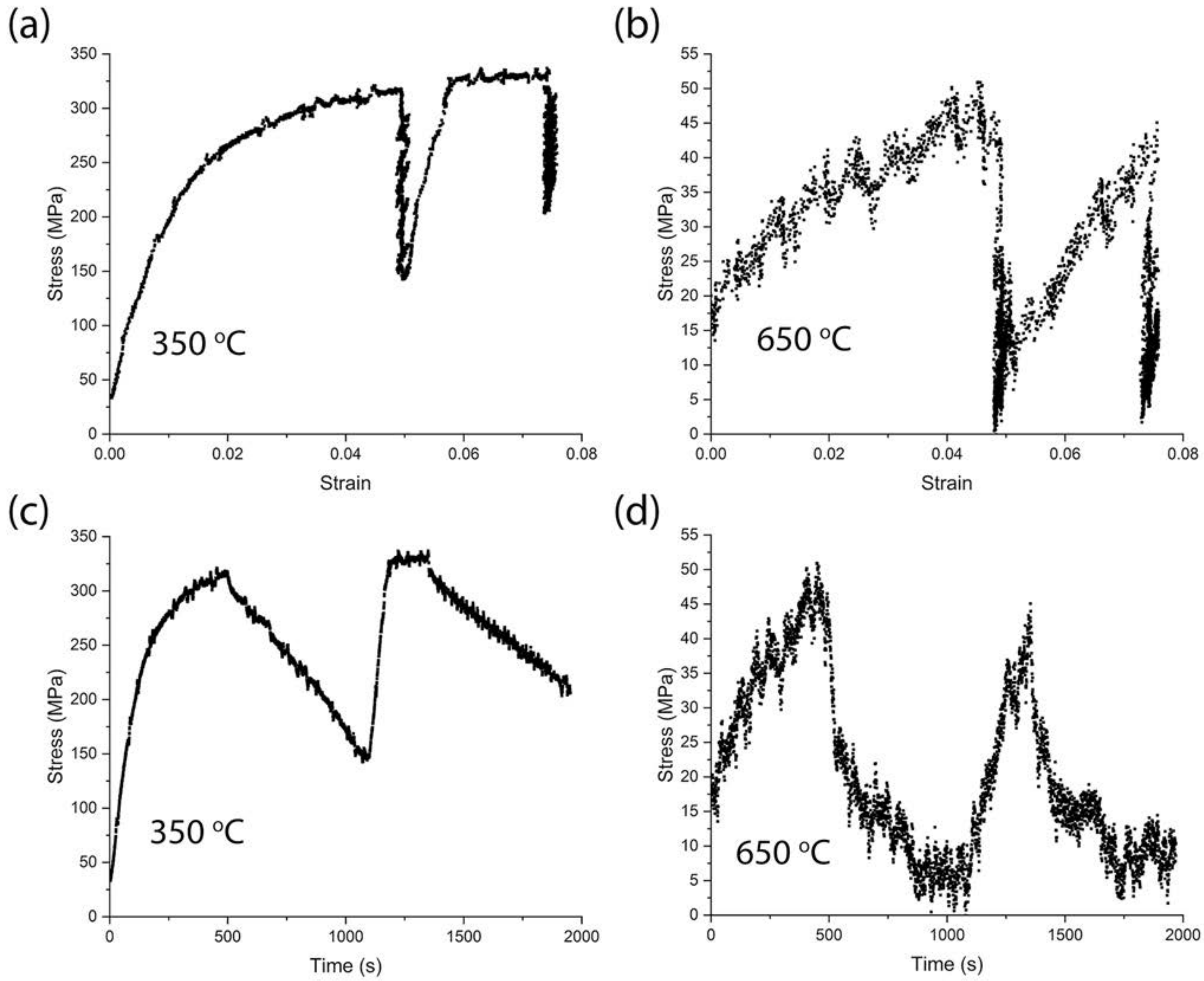



*Figure 1. Stress and strain curves for tests conducted on Cr coated Zircaloy-4 at (a) 350°C and (b) 650°C Stress as a function of time is presented in (c) and (d).*

## Accomplishments

The technical goals of this work were to measure stress relaxation response of Cr coated Zircaloy-4 cladding near the expected service temperature and also at an elevated temperature. To accomplish this, stress relaxation creep testing was performed on Cr coated Zircaloy-4, provided by Tim Graening at ORNL. High temperature tensile tests were performed near 350 and 650°C, including stress relaxation creep tests and 10-4 s-1 tensile tests to strain values of 5 or 7.5 %. Stress-strain and stress-time curves for tensile creep tests are presented in Figure 1. Stress relaxation experiments were performed at around 5 and 7.5% strain. Strain was held constant while stress was allowed to decrease. At the two temperatures, 350 and 650°C, the rate of stress relaxation increased with temperature. At 350°C, the decrease in stress with time is near linear (Figure 1c) as compared to a nonlinear decrease in stress at 650°C (Figure 1d). At 350°C the rate of stress decrease during the entire duration of stress relaxation decreased from -0.27 MPa s-1 to -0.18 MPa s-1 as the holding strain was increased from 5 and 7.5 % strain.

Further quantification of the stress relaxation rates can be obtained by fitting the relaxation curves. The fitting curves also aid interpretation of the noisy stress data because of the high temperature testing. Of the tests, the 350°C curve was fitted with linear fits while the 650°C test was fitted with exponential fits. Figure 2 shows the fitted curves on the experimental data. When comparing the fitting equations, the higher temperature has a greater exponential coefficient of -0.0077 and -0.0078 compared to -0.0023. When comparing strain levels, at the higher temperature the exponential coefficient was nearly the same for the 5 and 7.5% strain levels. In the case of 350°C, the relaxation rate was reduced by a third when going from 5 to 7.5 % strain levels. The difference between the lower and higher temperatures is likely caused by work hardening between the holds at the lower temperature and softening between the holds at the higher temperature.

The behavior of Zircaloy-4 under stress relaxation testing was examined. The analyzed stress-strain data shows that Zircaloy-4 displays temperature-dependent stress relaxation behaviors at 350 and 650°C. Transmission electron microscopy showed that the Cr coating maintained good adhesion with the Zircaloy-4 at 350°C away from cracks. Further work will investigate the microstructure of the 650°C samples. Additionally, the stress relaxation results will be provided to the modeling team to assist fitting their model on Zircaloy-4 creep.

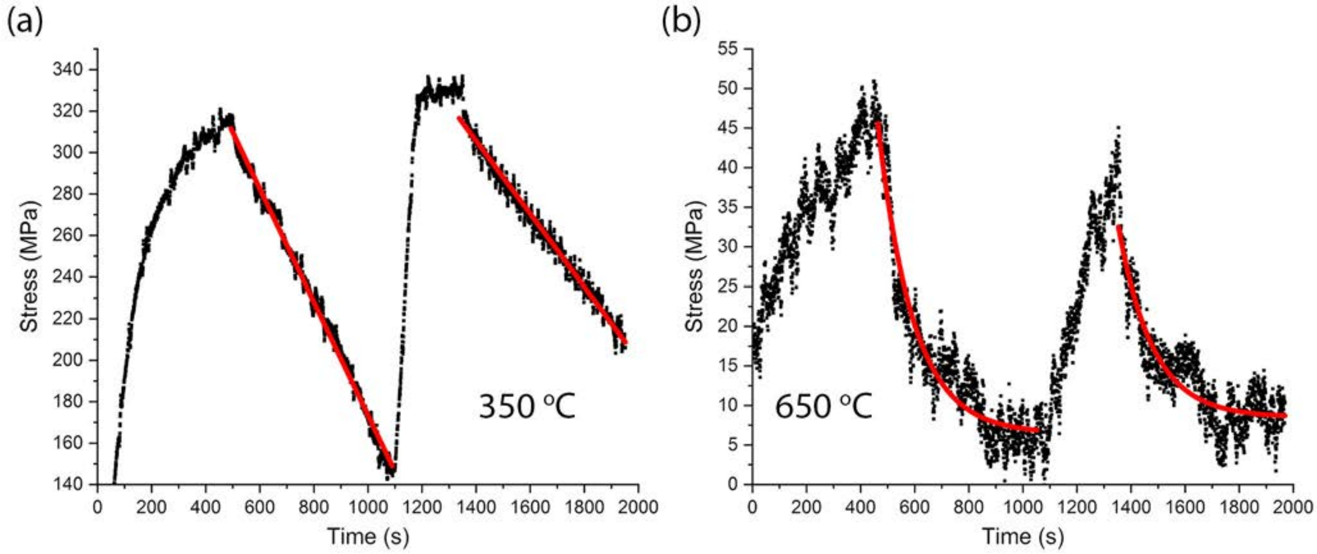



*Figure 2. Stress-time curves for Cr coated Zircaloy-4 at (a) 350°C, (b) 650°C. The black symbols are the experimental data while the red curves are the fitting curves.*

# Post-irradiation Examination of Wrought and Oxide Dispersion Strengthened FeCrAl Alloys

*Principal Investigator: Caleb P. Massey (Oak Ridge National Laboratory [ORNL])*
*Team Members/Collaborators: Annabelle Le Coq, Kory Linton, Jesse Werden, Samuel Bell, Yukinori Yamamoto, David Hoelzer, Sebastien Dryepondt (All ORNL)*

Over the past decade, the Advanced Fuels Campaign (AFC) has invested heavily (and in partnership with industry) to accelerate the development of Fe-based alloys for accident-tolerant fuel cladding. These alloys were initially in wrought form and had primary alloying constituents of Fe, Cr, and Al, with minor alloying elements of Mo, Si, and Y, respectively. Although the oxidation and burst properties of these FeCrAl alloys were superior to those of Zr-based alloys in accident scenarios, their neutron irradiation revealed severe irradiation hardening and embrittlement in light-water reactor (LWR) operating environments. This project aimed to investigate the resistance of new low-Cr oxide dispersion strengthened (ODS) FeCrAl alloys irradiated in the same conditions up to (and beyond) the end-of-life doses expected in LWRs.

## Project Description

In fiscal year (FY) 19, Oak Ridge National Laboratory (ORNL) fielded a High Flux Isotope Reactor (HFIR) irradiation titled the Advanced Manufacturing, ODS, and Wrought (AMOW) campaign. The AMOW irradiation campaign included capsules irradiated within the LWR (i.e., ~300°C) temperature regime at doses spanning damage levels of 8 and 16 displacements per atom, or dpa. Within the AMOW experimental matrix, two variants of wrought FeCrAl alloy (C26M), with varied levels of prior deformation during processing, were evaluated against multiple variants of FeCrAl-ODS alloys produced over the past 7 years at ORNL. Ultimately, the data generated from this irradiation campaign will inform alloy processing and optimization efforts for the next phase of accident-tolerant fuel technology as part of the next-generation fuels portfolio.

***The use of advanced ODS materials could alleviate irradiation hardening and embrittlement concerns seen in the first phases of accident-tolerant fuel cladding development in AFC.***

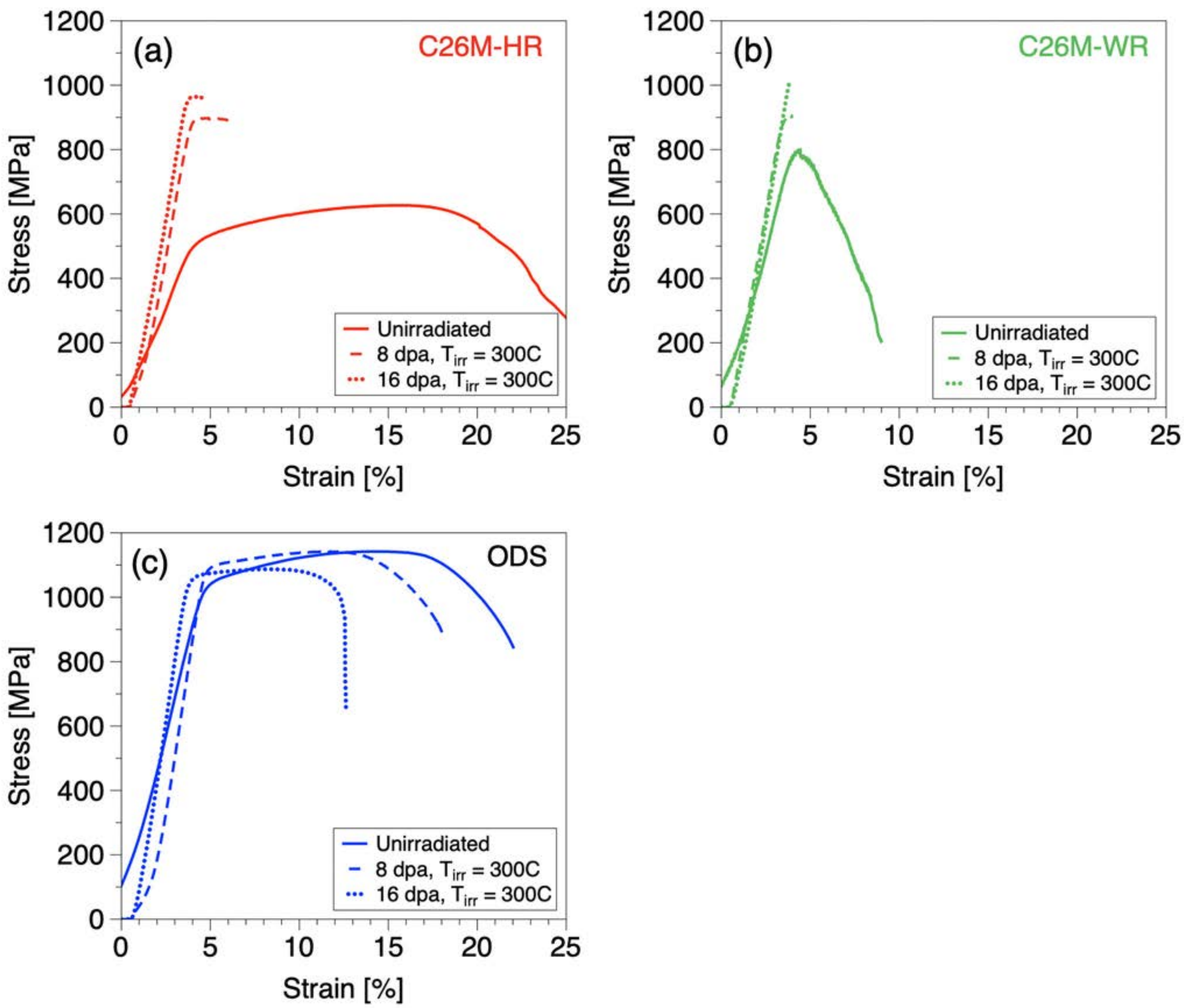



*Figure 1. Room-temperature tensile curves of (a) hot-rolled C26M, (b) warm-rolled C26M, and (c) FeCrAl-ODS following irradiation in HFIR to 8 and 16 dpa at a target irradiation temperature of 300°C.*

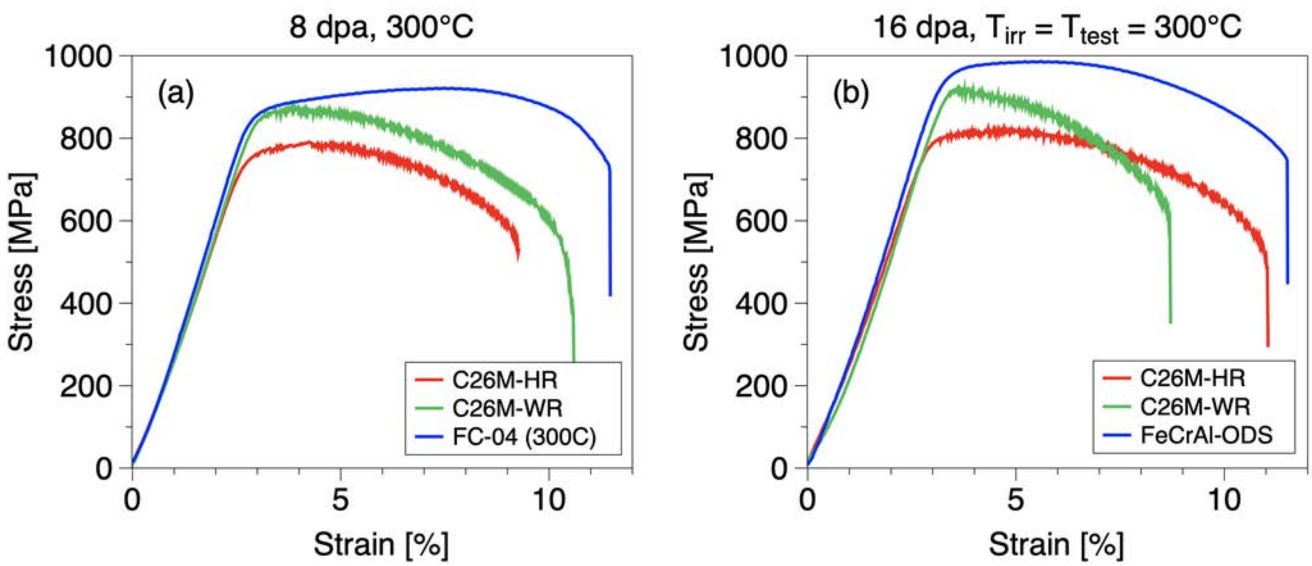



*Figure 2. Elevated-temperature (300°C) tensile curves of hot-rolled C26M, warm-rolled C26M, and FeCrAl-ODS following irradiation in HFIR to (a) 8 and (b) 16 dpa at a target irradiation temperature of 300°C.*

**Accomplishments**

During FY24, mechanical tests were conducted on both wrought C26M and ODS FeCrAl materials following irradiation to 8 dpa (capsule AMOW01) and 16 dpa (capsule AMOW02) in HFIR at a target irradiation temperature of 300°C. Experimental irradiation temperatures for these capsules were estimated at 324±16°C and 340±32°C, respectively, based on passive SiC thermometry specimens co-located within each capsule. Room temperature tensile tests of these specimens are summarized in Figure 1.

For these alloys to be potentially utilized as fuel cladding, even if the uniform elongation is low, at least some ductility following onset into plastic instability is needed to prevent sudden catastrophic failure of the cladding during handling, transportation, and storage following in-reactor use. The hot-rolled (HR) C26M (Figure 1(a)) represents the normal response of wrought FeCrAl alloys following neutron irradiation. With increasing irradiation dose, the initial high ductility is eroded while the mechanical strength increases by ~400 MPa. By performing warm rolling (WR) processes on the C26M FeCrAl alloy, the initial dislocation density can be increased through work hardening of the material, albeit sacrificing initial ductility in the process. Figure 1(b) shows that although the C26M-WR material has a higher initial strength, this alteration of microstructure was insufficient to prevent the significant irradiation hardening and ductility loss for the alloy: even less ductility was measured for this alloy in comparison with the reference C26M-HR material. For the FeCrAl-ODS material with nominal composition Fe-10Cr-6Al-0.3Zr+0.3$Y_2O_3$, a fundamentally different irradiation response was observed. Although the material lost approximately half of its ductility following the maximum irradiation dose, Figure 1(c) shows over 10% ductility in the highest dose condition. It is important to note that all three of these materials were irradiated in the same capsule, so differences in irradiation response are a function of the processing differences rather than subtle differences in irradiation conditions.

Similar trends were observed for the same alloys tested at elevated temperatures relevant for LWR operation. Figure 2 compares the tensile curves tested at 300°C for each alloy at 8 dpa and 16 dpa, and the same trends can be seen for both conditions. The HR material has the lowest strength, and it has only a 2% strain hardening capacity following irradiation. The WR material is slightly stronger, but it has only a 1% strain hardening capacity prior to necking and eventual failure. The ODS material has significantly more strain hardening capacity at both irradiation conditions compared with the wrought FeCrAl alloys. Thus, the FeCrAl-ODS materials developed in the AFC program show promise in solving the irradiation hardening/embrittlement problem for conventional wrought FeCrAl alloys.

## Byron Fuel Shipment Receipt and Unloading

*Principal Investigator: Aaron Colldeweih (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Jake Stockwell, Alex Hanson (All INL)*

***The successful receipt and unload of this shipment of commercial fuel from the Byron commercial power plant marks the first shipment of commercial fuel to the INL for PIE since the Idaho Settlement Agreement was put in place.***

This effort received and unloaded the first shipment of commercially irradiated (and commercial length) research nuclear fuel since the initiation of the Idaho Settlement Agreement in the early 2000's. While the INL (specifically, the Hot Fuel Examination Facility (HFEF)) had previously received commercial length nuclear fuel in the past, it had been nearly two decades since the last shipment, so the facility's ability to complete this work was tested.

### Project Description

The objective of this research was to ensure that HFEF could handle full length commercial nuclear fuel specimens. This allows Accident Tolerant Fuel (ATF) Concept commercial lead test rods to be examined at the INL. Examination of these rods will help provide scientific evidence to extend the lifecycles of existing fuel designs as well as prove and qualify the new upgraded ATF fuel concepts.

### Accomplishments

The goal of this effort was to receive and unload the full 25-rod shipment of commercial-length fuel from the Byron reactor. The team was able to successfully receive and unload the Byron shipment of 25 commercial fuel rods from the Byron reactor and demobilize the shipment cask. This proved that HFEF could continue to handle commercial-length irradiated fuel and was a viable facility for completing the post irradiation examination (PIE) on these commercial specimens to qualify them for use. The 25-rod shipment of fuel was unloaded in mid-January and the cask was demobilized later that month. This effort required close coordination between a very large team made up of: Erik Woolstenhulme and Ed Mai in the INL's Nuclear Science and Technology department, Ernesto Pitruzella and the rest of the team at Westinghouse, team members at the Byron Nuclear Generating Station, Fabiola Cappia, James Angell, Jadin Frongner, Dennis Wahlquist, Jake Stockwell, and Aaron Colldeweih to coordinate the work in the HFEF. The unload went nearly perfectly, with only one equipment failure

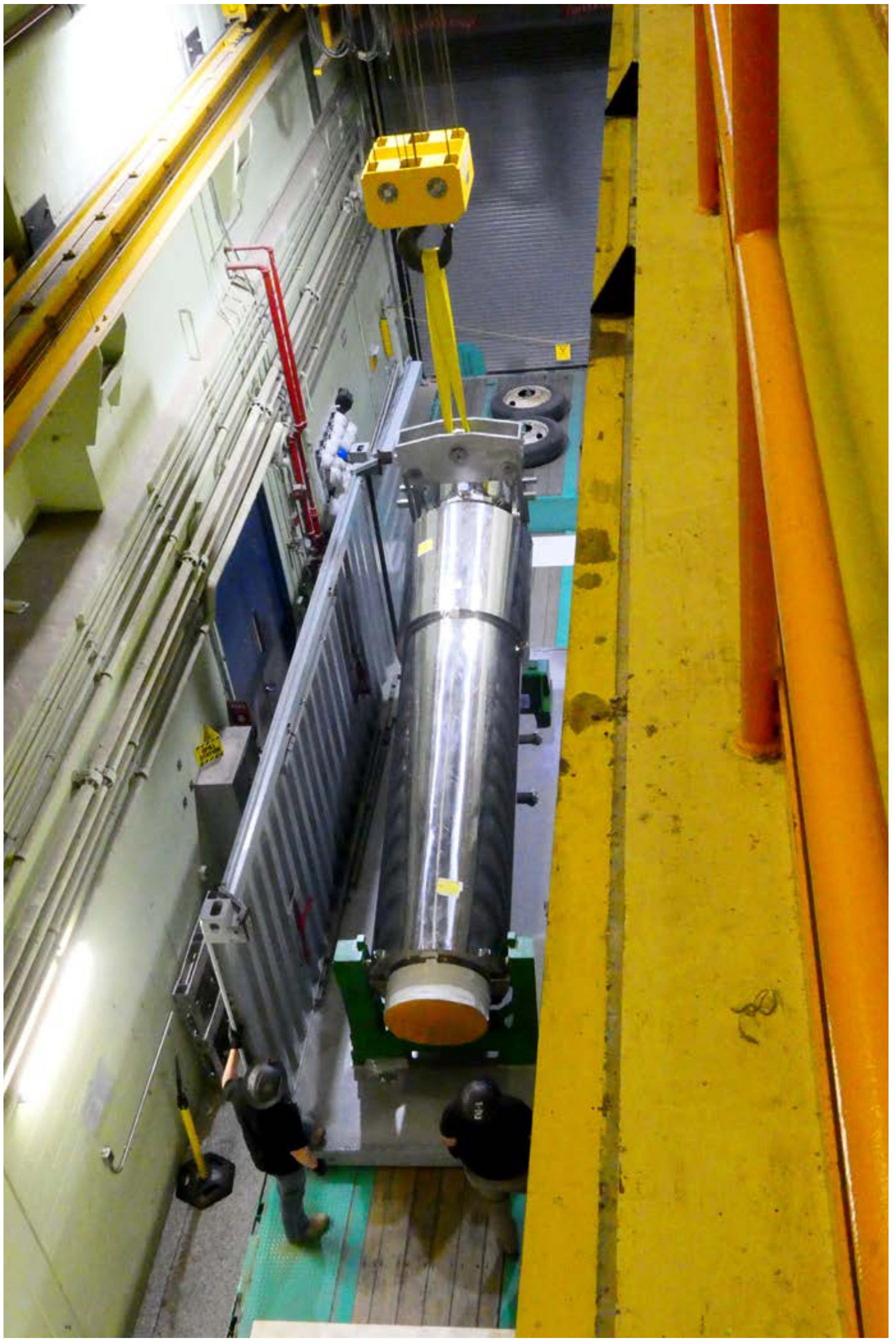


*Figure 1. Cask upending in HFEF.*

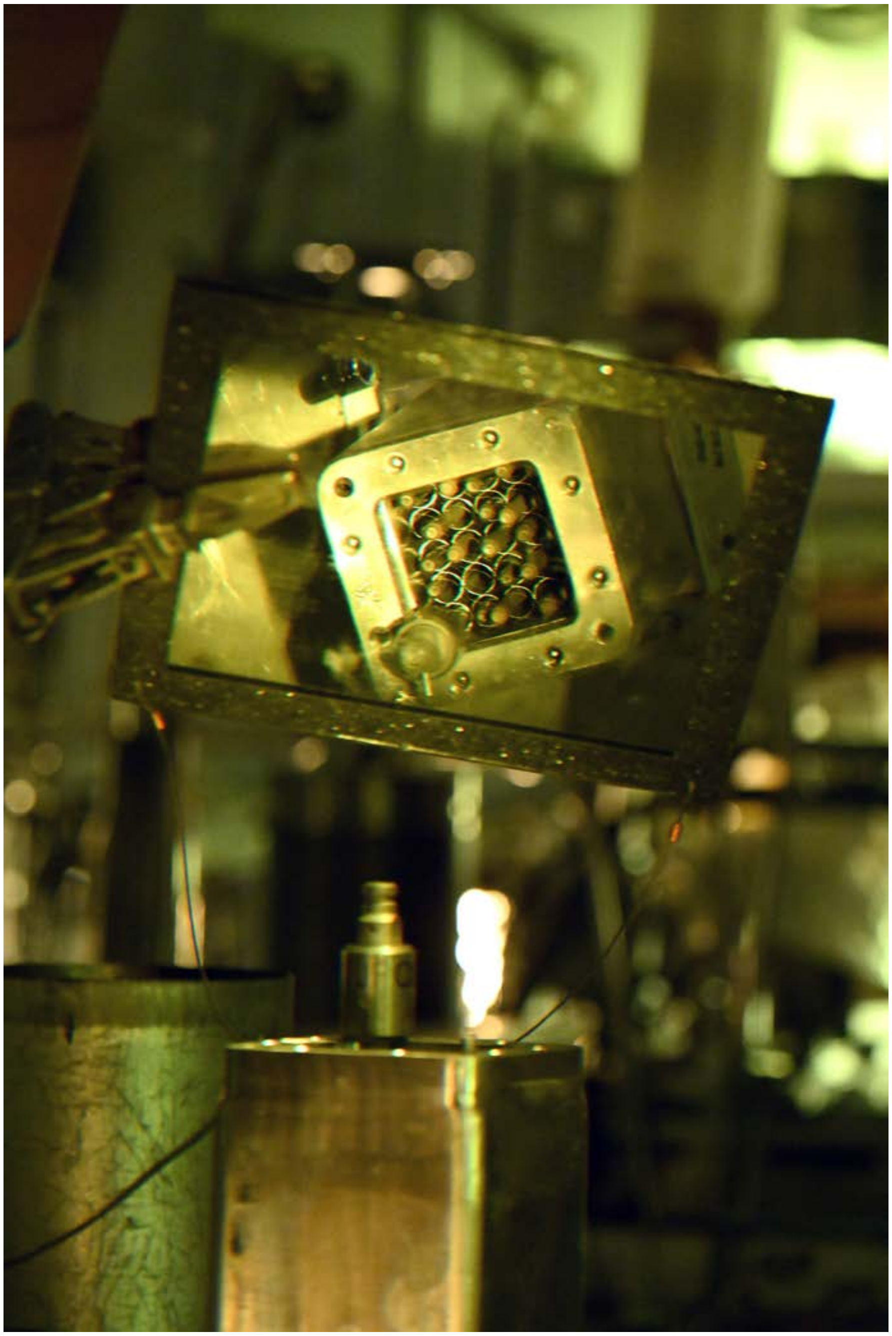


Figure 2. Looking down into NAC cask with partial payload.

which caused a delay. This unload required the team to re-fabricate and requalify old equipment that had been broken or lost over the 20 years of layup in the HFEF Main Cell; the team completed these preparations on-time to complete the unloading effort.

The unload process is pictured in Figures 1 through 3. Figure 1 shows the cask in the HFEF truck lock, in the lifting process called “upending”. Figure 2 shows the NAC cask payload, as seen in a mirror from above the cask (as the operators would see the cask during the unload activities). An end fitting can be seen on the top of one of the rods. Figure 3 shows the HFEF operators removing a rod from the cask, after it has been mated to the cell.


*Figure 3. Operators removing a fuel rod from the NAC cask.*

# Non-Destructive Examination and Sectioning of Byron Commercial Fuel

*Principal Investigator: Aaron Colldeweih (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Jake Stockwell, William Alex Hanson (All INL)*

***Completion of the NDE, sectioning, and beginning of DE on the commercial fuel from the Byron commercial power plant marks the first PIE on commercial fuel at the INL for PIE since the Idaho Settlement Agreement was put in place.***

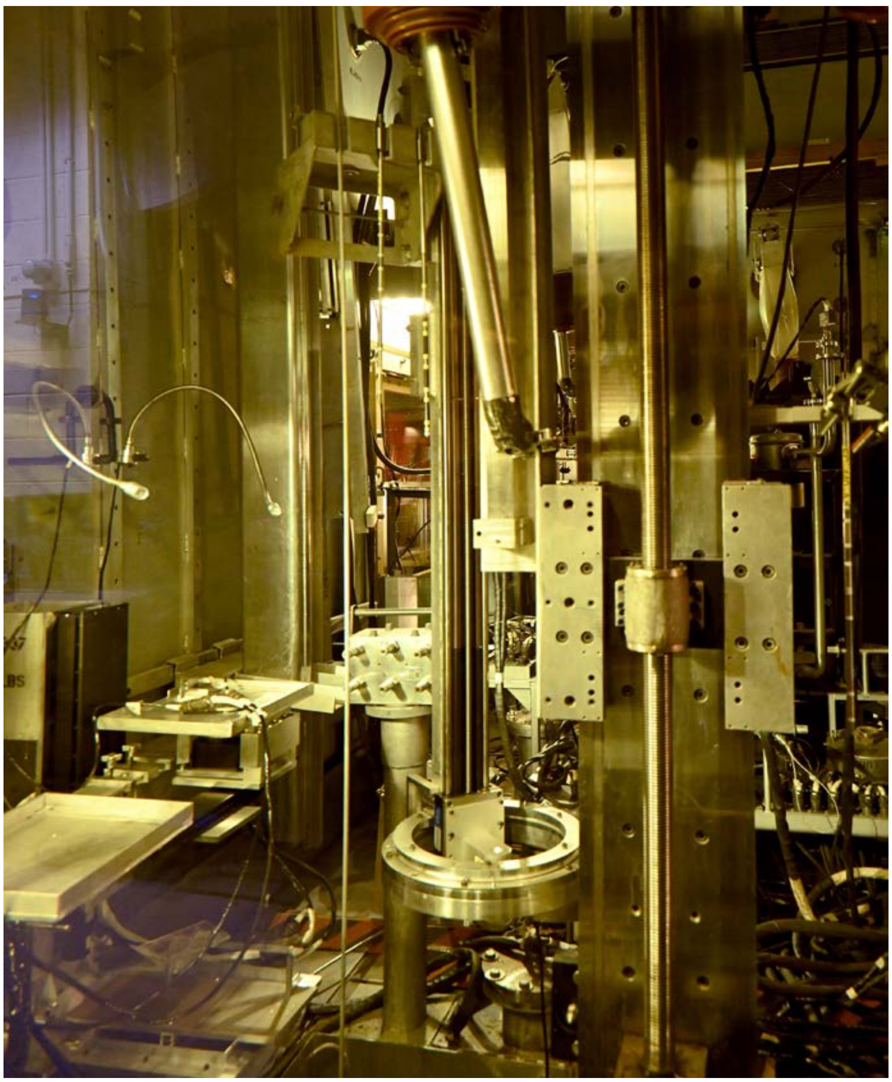
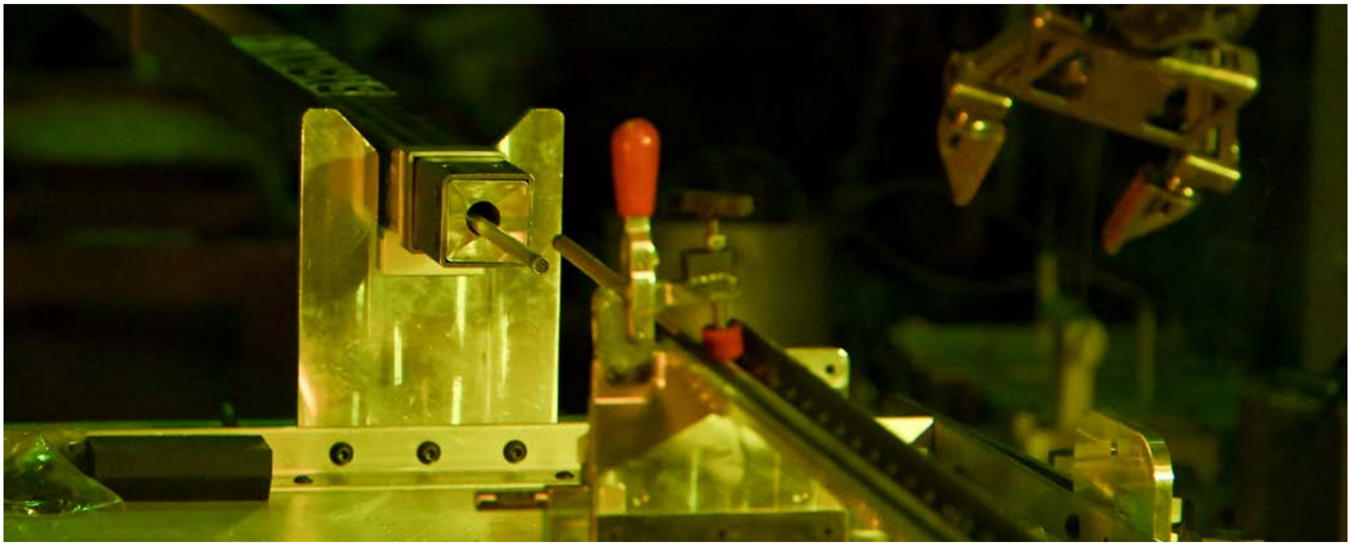


*Figure 1. Byron rod hanging in the Visual Examination Machine.*

This effort completed the first parts of post irradiation examination (PIE) on the first shipment of commercially irradiated (and commercial length) research nuclear fuel since the initiation of the Idaho Settlement Agreement in the early 2000's. While INL (specifically, the Hot Fuel Examination Facility (HFEF)) had previously completed PIE on commercial length nuclear fuel in the past, it had been nearly two decades since the last shipment so the facility's ability to complete this work was tested.

### Project Description

The objectives of this research were two-fold; first, this research aimed to begin to examine the rods, then cut and prepare pieces of the rods for subsequent testing in Transient REActor Test Facility (TREAT). Second (as a byproduct of the first goal) the research aimed to ensure HFEF 's aging infrastructure and engineers could successfully handle full length commercial nuclear fuel specimens to generate the TREAT test specimens. The purpose of preparing the TREAT tests on these rods is to show the performance of the cladding under Reactivity Initiated Accident (RIA) or Loss of Coolant Accident (LOCA) scenarios, via the TREAT reactor at higher burnup levels. These tests will help the allow Accident Tolerant Fuel program proven the effective-ness of the new cladding concept. Specifically, these high burnup



(HBU) tests will help to show the baseline performance of existing approved cladding concepts for commercial light water reactors in the RIA and LOCA scenarios. This sets the stage for future LOCA and RIA TREAT tests of the new ATF concepts from various vendors and to prove and qualify the new upgraded ATF fuel concepts.

### Accomplishments

The goal of this effort was to complete the Non-Destructive Examinations (NDE) of 3 of the rods designated for the High burnup examination program, to section the first rod, to begin crucial Destructive Examination (DE) efforts, and to prepare a segment of this rod for subsequent RIA testing in the TREAT reactor (a separate work package). At the end of the year, the team was able to complete the NDE on all 5 of the HBU designated rods. The Visual examination step is shown in Figure 1, with a rod hanging in the Visual Examination Machine. The completion of NDE on all the rods enables the subsequent sectioning steps to be completed whenever the team is ready in fiscal year 2025. The team was also able to section the first rod, as shown in Figure 2, and prepare the rodlet section for refabrication (a separate work package). Finally, the team initiated crucial destructive PIE activities. The team completed the first of many advanced characterization activities including preparation of stubs for advanced microscopy, as well as cladding hydrogen pickup measurements and fuel burnup analysis via analytical chemistry techniques in the MFC Analytical Laboratory.

*Figure 2. Byron rod rough sectioned.*

# First Results from Water Loop Corrosion Tests in MITR

*Principal Investigator: David Carpenter (Massachusetts Institute of Technology [MIT])*
*Team Members/Collaborators: Arunkumar Seshadri, Koroush Shirvan (All MIT)*

The Massachusetts Institute of Technology Reactor (MITR) water loop experiment commenced in May 2024 following an extended shutdown for repairs. The loop was successfully operated for 53.9 days at high temperature in a normal water chemistry environment, including 33.8 days at full reactor power. This cycle included coated accident-tolerant fuel (ATF) tubular specimens from several vendors, such as Westinghouse, General Electric, and Framatome. It also featured ATF tubular specimens developed through university partnerships, incorporating both chromium-based and a novel chromium-niobium coating.

Specimens were placed in three different locations within the MITR, all designed to mimic typical light water reactor (LWR) temperature and pressure conditions: the in-core location (which experiences neutron/gamma flux similar to that of a commercial LWR), the upper reflector region (which has significant gamma flux), and an external autoclave connected to the


*Figure. 1. MITR water loop: capsule assembly.*

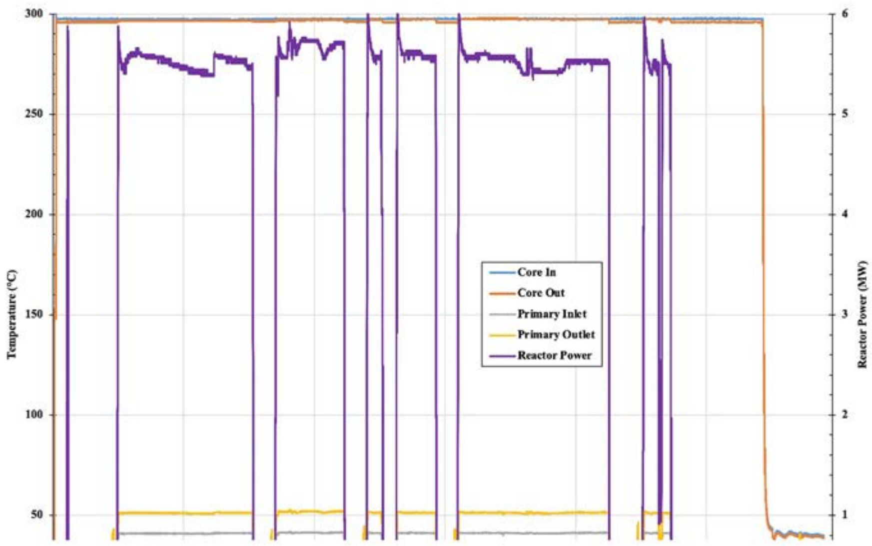



*Figure. 2. MITR water loop: power curve.*

water loop (which has no neutron/gamma flux). The in-core capsules were removed after 33.8 effective full power days and are scheduled for reinsertion in October for an additional 10-week radiation cycle. The samples in the other two locations continue to be irradiated. Post-irradiation examinations (PIE) are planned for Q2 of 2025, which are anticipated to yield crucial insights into the combined irradiation and corrosion behavior of various ATF coated cladding, facilitating their commercial deployment.

### Project Description

The U.S. Department of Energy's Office of Nuclear Energy has dedicated over a decade to developing ATF claddings, investing significantly in various U.S. industries for this essential advancement. ATF aims to substantially improve the reliability and safety of nuclear fuel, as well as the economics of nuclear reactor operations. Their enhanced heat tolerance leads to greater reactor safety and security while also allowing for improved economic performance by enabling more flexible reactor operation within safety margins.

To support the licensing and eventual deployment of ATF, comprehensive irradiation and corrosion testing of ATF candidates is crucial. Few facilities worldwide can conduct such testing. MITR features a versatile 6 MW reactor and boasts innovative systems like the High-Temperature Water Loop (HTWL) and the upcoming High-Pressure Water Loop (HPWL). These capabilities are complemented by high-fidelity simulations, specialized irradiation vehicles, and adaptable in-situ electric heaters, all aimed at accelerating nuclear fuel research and development. Currently, MITR operates the HTWL, an in-core pressurized water loop located in one of its experimental zones. This setup simulates typical boiling water reactor (BWR) conditions, heating specimens up to 300°C, constrained by the autoclave's

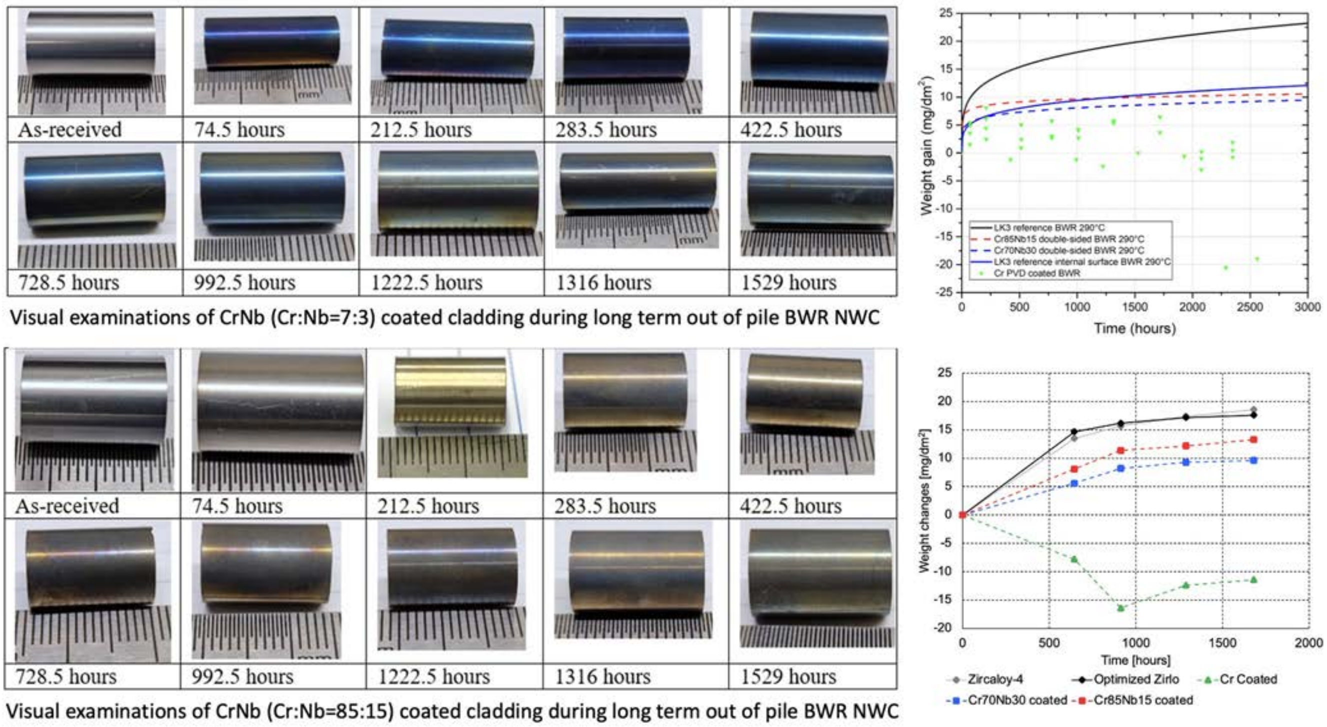



*Figure. 3. Out of pile corrosion tests demonstrate the stability of CrNb coating.*

pressure rating of approximately 10.3 MPa. All operations, including flow, heating, and sampling, occur outside the primary reactor vessel, while samples are placed in areas with peak neutron flux or in gamma-only regions above the core. The HTWL is extensively used for researching materials' corrosion, irradiation behavior, and real-time instrument testing, particularly for ATF claddings. This project focuses on several key aspects:

- Developing a HPWL capability to extend temperature limits to 350°C and pressure limits to 150 bar, facilitating simulated pressurized water reactor testing for ATF candidates.
- Investigating BWR coating survivability under prototypical hydrothermal corrosion, addressing the challenges faced by the ATF program in oxidizing and radiation environments. The current project utilizes MITR's HTWL capabilities to enhance the maturity of BWR candidate coatings.
- Collaborating with various industry partners and universities, including Westinghouse, Framatome, GE-Hitachi, and General Atomics, to test multiple novel coatings.
- Leveraging advanced PIE characterization capabilities at Idaho National Laboratory for detailed insights into damage mechanisms.
- Understanding the combined effects of fouling and corrosion on Zircaloy and ATF candidates.

## Accomplishments

A test section featuring over 40 specimens developed by Westinghouse, GE Research, Framatome, Czech Technical University, and MIT was inserted into the water loop, undergoing preliminary testing for over 30 days in a radiation environment and more than 50 days in normal water chemistry, simulating prototypical BWR temperature and pressure conditions. These specimens included various types of coated zircaloy cladding, FeCrAl, and composite SiC-based materials, and the radiation cycle comprised both open rings and samples with end plugs, with the sample assembly and power profile illustrated in Figures 1 and 2, respectively. Specimens were positioned in three distinct locations within the MITR to replicate typical LWR conditions: the in-core location (exposing specimens to neutron/gamma flux similar to a commercial LWR), the upper reflector region (experiencing significant gamma flux), and an external autoclave connected to the water loop (with no neutron/gamma flux). The in-core capsules were removed after 33.8 effective full power days and are set to be reintegrated in October for an additional 10-week radiation cycle, while samples in the other two locations will accumulate at least 25 weeks of irradiation. PIE of all samples is scheduled for Q2 of 2025, anticipated to provide critical insights into the combined irradiation and corrosion behavior of various ATF coated cladding, aiding their commercial deployment.

Additionally, Phase 2 of the new HPWL began in September, and the project is on track to commence HPWL operations by April 2025.

MIT developed a new CrNb coating for BWRs as a replacement for chromium coatings and tested in out-of-pile corrosion facilities. The challenge with Cr-based coatings is that Cr oxides formed on Cr or CrN exhibit significant solubility under BWR-normal water chemistry (NWC) conditions. Thermodynamic analysis using a Pourbaix diagram at 300 °C shows that the equilibrium between $Cr_2O_3/HCrO_4^-$ is approximately 80 mV (SHE), while the Electro-Chemical Potential of stainless steel in BWR-NWC conditions ranges from 100 to 200 mV. The analysis with CrNb revealed stable Nb oxide formation, prompting the development and testing of this concept. Cr-Nb coatings on zircaloy were produced through unbalanced magnetron sputtering with two different Cr:Nb ratios (7:3 and 85:15 by mass) and were tested in both static and high-flow corrosion loops, with the latter simulating MITR water loop conditions. Preliminary results for this coating have been promising, demonstrating significant stability compared to previously developed Cr-based coatings, as illustrated in photographs and graphs in Figure 3. CrNb coatings have also been inserted into the water loop, and the PIE is expected to yield further insights for advancing this coating for use in BWR environments.

***MITR corrosion tests provide valuable insights for the development of ATF cladding for BWRs.***

# Progress and Status of ATF2 Irradiations

*Principal Investigator: David Kamerman (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Brian Durtschi, Matilda Aberg-Lindell, Seth Kilby (All INL)*

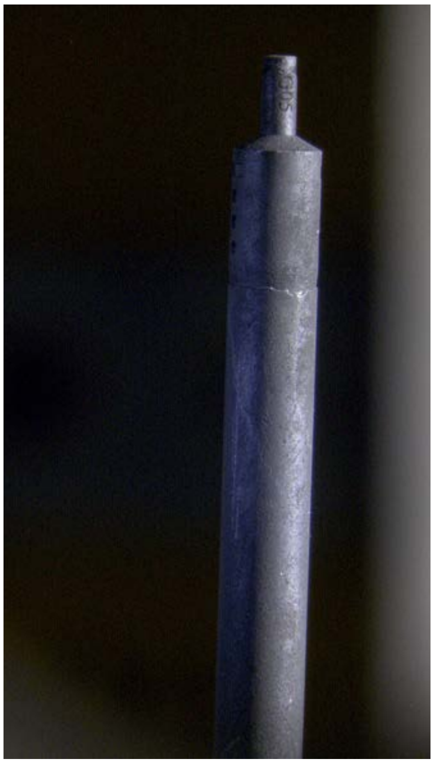


*Figure 1. Silicon Carbide Test Pins Irradiated in ATF2 under Examination in the Hot Fuels Examination Facility.*

The Accident Tolerant Fuels 2 (ATF2) Irradiation Project continues to be the world leading pressurized water reactor irradiation experiment. Irradiations of ATF concepts began in 2018, underwent a significant design change in fiscal year (FY)22 and FY23 and continued irradiation completing three additional 60-day cycles in FY24.

## Project Description

The ATF2 irradiation project takes place in the center flux trap of Advanced Test Reactor (ATR) in the nation's (and western world's) only fully prototypic pressurized water loop available for civilian research purposes. The irradiation test has been testing domestically produced near term ATF concepts with prototypic radial dimensions since 2018. It has completed seven cycles of irradiation prior to the ATR core internal change out which lasted two years from mid FY21 to mid FY23. In late FY23, ATR restarted with a brand new ATF2 test train which included instrumented leads in the first internationally sponsored ATF irradiation in a collaboration between the Japan Atomic Energy Agency and INL. The new test train also included for the first time unfueled Silicon Carbide composite cladding test pins from both General Atomics and Framatome.

## Accomplishments

In FY24 the new test completed nearly three additional cycles of irradiation. The fuel temperature sensors in the test train showed good performance for the first two cycles of irradiation but in the third cycle they have not produced meaningful data. The plenum pressure sensors have not been working in the test. The first batch of the Silicon Carbide composite test pins was shipped to HFEF (Figure 1) for PIE with initial indications showing that the pins remained hermetic during their two cycles of irradiation.

Additionally, in FY24 the ATF2 team neared completion of a third design iteration for the ATF2 experiment with a focus on standardization of the test pins sizes and an investment in poolside nondestructive examination (NDE) equipment in the ATR canal to support the generation of more data from the test sooner in a way that is more reliable than the in-situ instrumentation currently in the test. The new design uses a series of carefully placed Hafnium shrouds in each of the four tiers of the test train. This achieves a more flat axial power profile in each test pin as seen in Figure 2.

TABLE II. Final configuration of the ATF-2D shroud components.

| Tier | Lower shroud Material | Lower shroud Thickness | Main shroud, bottom Material | Main shroud, bottom Thickness | Main shroud, top Material | Main shroud, top Thickness | Upper shroud Material | Upper shroud Thickness |
|---|---|---|---|---|---|---|---|---|
| 1 (bottom) | Zr | 26 mil | Zr | 10 mil | Hf | 10 mil | Hf | 36 mil |
| 2 | Hf | 26 mil | Hf | 20 mil | Hf | 20 mil | Hf | 36 mil |
| 3 | Hf | 36 mil | Hf | 20 mil | Hf | 20 mil | Hf | 26 mil |
| 4 (top) | Hf | 36 mil | Hf | 10 mil | Zr | 10 mil | Zr | 26 mil |

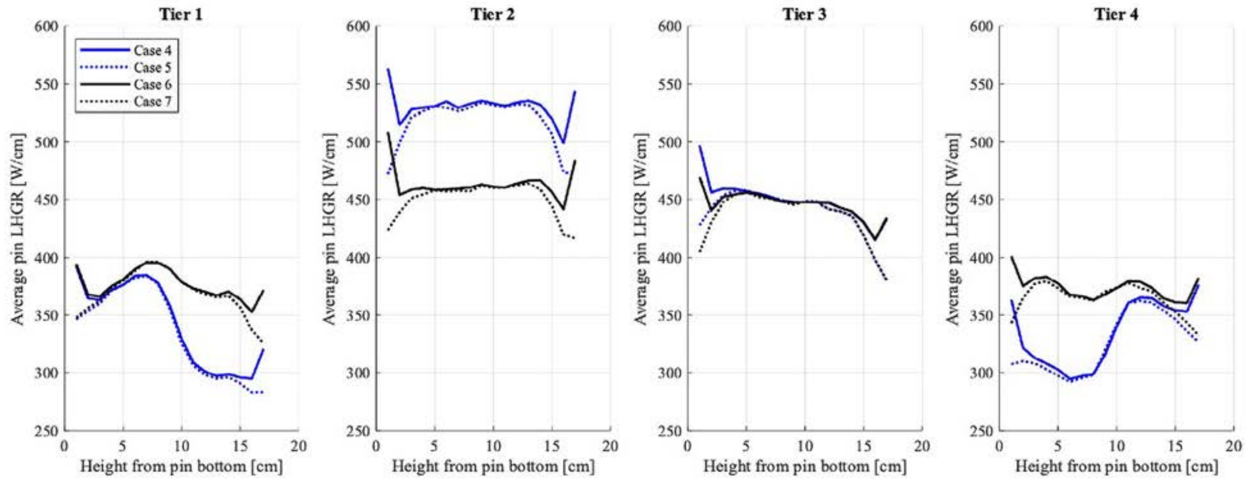



*Figure 2. Heat Rates of Standard ATF2 Test Pins under Different Hafnium Shroud Geometries.*

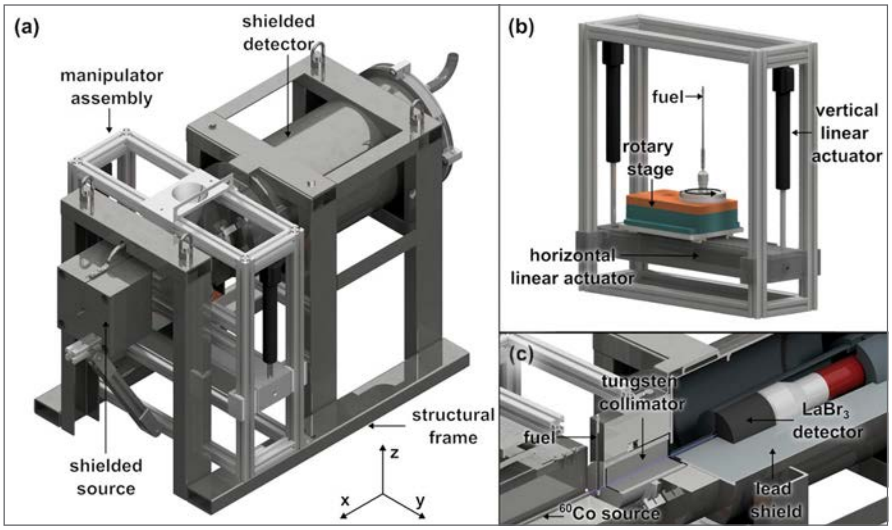



*Figure 3. CAD Rendering of the ATF2 Gamma Scanner.*

The poolside NDE will consist of a detailed dimensional scan using a Newton Labs underwater laser scanner capable of generating a 10-micron resolution cad image of the test pins between irradiation cycles and prior to shipment. Following the dimensional scan the test pins will be interrogated with a gamma spectrometry / gamma tomography system (Figure 3) which will provide detailed axial power and burnup determinations, indications of fuel stack growth, fuel cladding gap distance/closure, and concentration of Kr-85 in the plenum to indicate fission gas release and determine plenum pressure. Fabrication of the gamma scanner is nearly complete in FY24 with commissioning planned in FY25.

***The ATF2 Irradiation project continues to provide reliable irradiations of near term and long-term Accident Tolerant Fuel concepts from domestic and international fuel suppliers while innovating on the best ways to provide important fuel performance data in the most timely manner.***


N

# Lead Test Rod Receipt at National Laboratories – Clinton Rods at Oak Ridge National Laboratory

*Principal Investigator: Jason Harp (Oak Ridge National Laboratory [ORNL])*
*Team Members/Collaborators: Casey McKinney, Peter Doyle, Yong Yan (All ORNL)*

Two full length boiling water reactor (BWR) rods were received at ORNL for post-irradiation examination (PIE) from the Clinton Power Station. PIE data generated in collaboration between ORNL and General Electric (GE) will be used to prepare licensing topicals for the ongoing use of conventional Zircaloy-2/$UO_2$ fuel and future Accident Tolerant Fuel (ATF) and High Burnup (HBU) applications.

## Project Description

Generation of fuel performance data from currently operating nuclear power plants is important to understand conventional approaches to reactor operation. Establishing a baseline for current generation fuel performance is needed to meet U.S. Department of Energy (DOE) and industry goals to improve the performance of U.S. commercial nuclear reactors. GE and its joint venture partner Global Nuclear Fuel (GNF) are investigating new ATF concepts for BWRs in collaboration with ORNL and other DOE national laboratories. GE is pursuing a variety of activities to enable economically favorable use of ATF technology via the introduction of low-enriched uranium plus and burnup extension beyond the currently used Zircaloy-2/$UO_2$ fuel licensing basis, which is applicable to a variety of ATF technologies. To help achieve

*Figure 1. NAC-LWT Cask containing two full length BWR rods from Clinton Power Station is unloaded at Oak Ridge National Laboratory Hot-Cell Facility.*

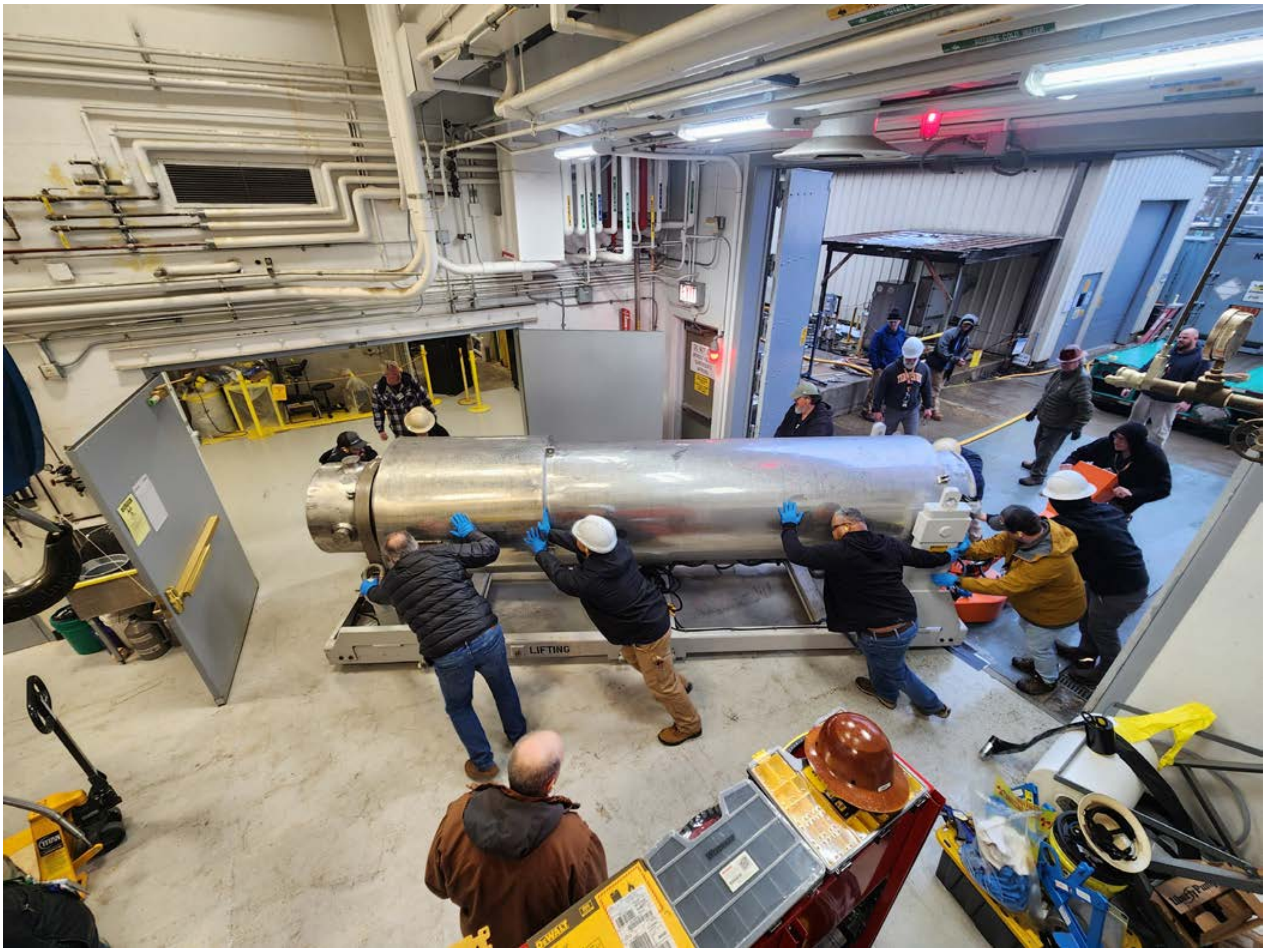
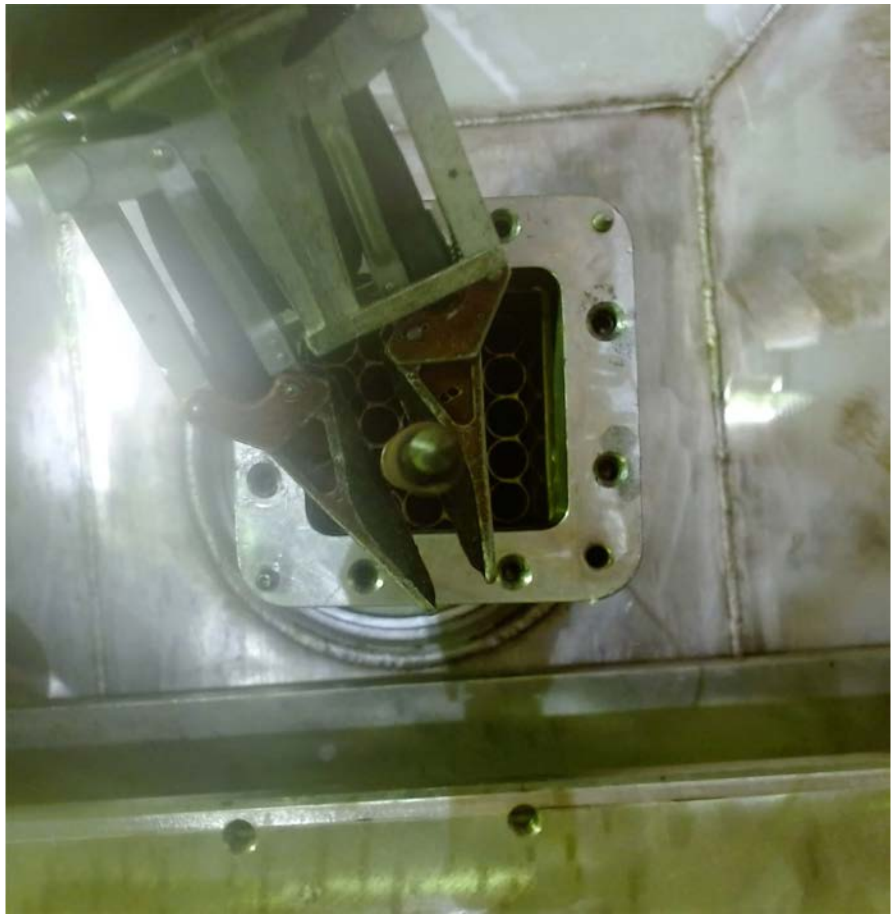



*Figure 2. NAC-LWT cask is maneuvered into the hot-cell facility.*

the HBU aspect of this objective, conventional Zircaloy-2/$UO_2$ based fuel was retrieved from Clinton Power Station, a GE-designed BWR Type 6 (BWR/6) design located near Clinton, Illinois, USA. The focus of the HBU work is, primarily, to obtain basic integral fuel performance and $(U,Gd)O_2$ / Zircaloy-2 property data to support HBU fuel licensing. The HBU data obtained will be applicable to both the ongoing use of the currently used conventional Zircaloy-2/$UO_2$ fuel system and new ATF fuel systems.

ORNL experience with conventional light-water reactor fuel and its hot-cell expertise are recognized by GE. The history of collaboration between GE and ORNL continues with the PIE of conventional fuel irradiated at Clinton Power Station. The PIE includes two conventional-

Figure 3. The first full length BWR rod is removed from the basket of the NAC-LWT cask and placed in the ORNL hot-cell.

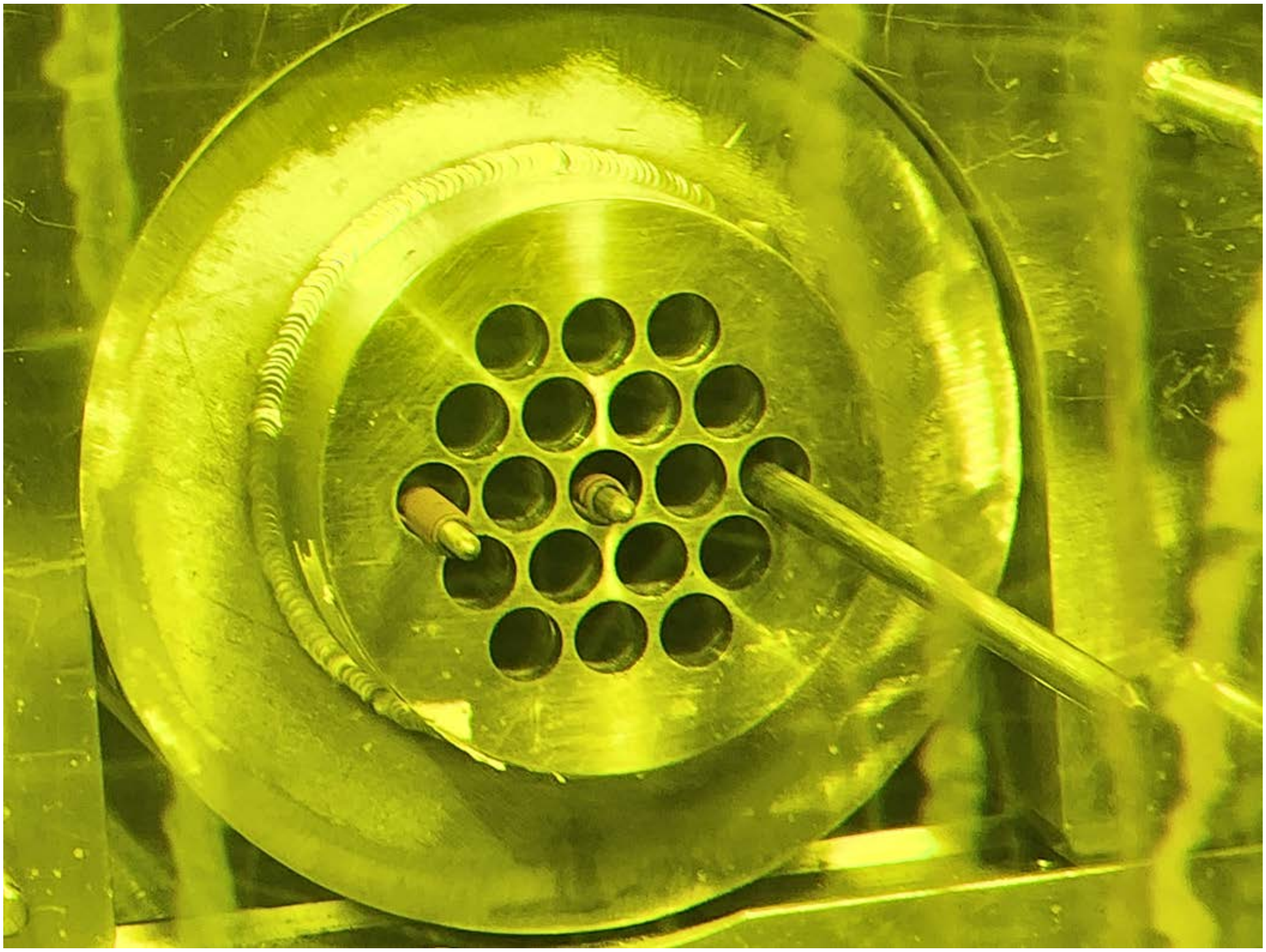


fueled Zircaloy-2 barrier full-length rods containing both $UO_2$ and $UO_2$ + 4 wt% $Gd_2O_3$. PIE data generated as a result of collaboration between ORNL and GE will be used to prepare U.S. Nuclear Regulatory Commission licensing topicals for the ongoing use of conventional Zircaloy-2/$UO_2$ and future ATF fuel systems in HBU applications.

**Accomplishments**

The receipt of nuclear fuel requires the careful coordination of many different parties. For this shipment relevant subject matter experts from GE, ORNL, Constellation (the operator of Clinton Power Station), DOE Headquarters staff, local DOE site office staff, STS Nuclear and NAC International. All these parties worked together to enable delivery of these 2 rods in January 2024 (see Figure 1). These full-length rods were shipped in a

*This fuel shipment will allow for the examination of high burnup BWR material at a U.S. facility for the first time in several decades including examination of Gd doped fuel for the first time in the U.S. in several decades.*



NAC-LWT cask. This is cask was successfully received at the ORNL hot-cell (Figure 2) where the rods were loaded into the hot-cell and placed in storage before PIE was initiated (Figure 3 and Figure 4).

Overall, this process was without incident and benefited from ORNL experience recently receiving the NAC-LWT cask to support other programs. PIE of this material will begin early in FY25.

*Figure 4. The two BWR fuel rods from Clinton (center and left-center) are visible in the fuel rod storage shield inside the ORNL Irradiated Fuel Examination Laboratory.*

## Recent High-Burnup Loss of Coolant Accident Testing at Oak Ridge National Laboratory

*Principal Investigator: Nathan Capps (Oak Ridge National Laboratory [ORNL])*
*Team Members/Collaborators: Yong Yan, Jason Harp, Mackenzie Ridley, Robert Salko Jr. (All ORNL)*

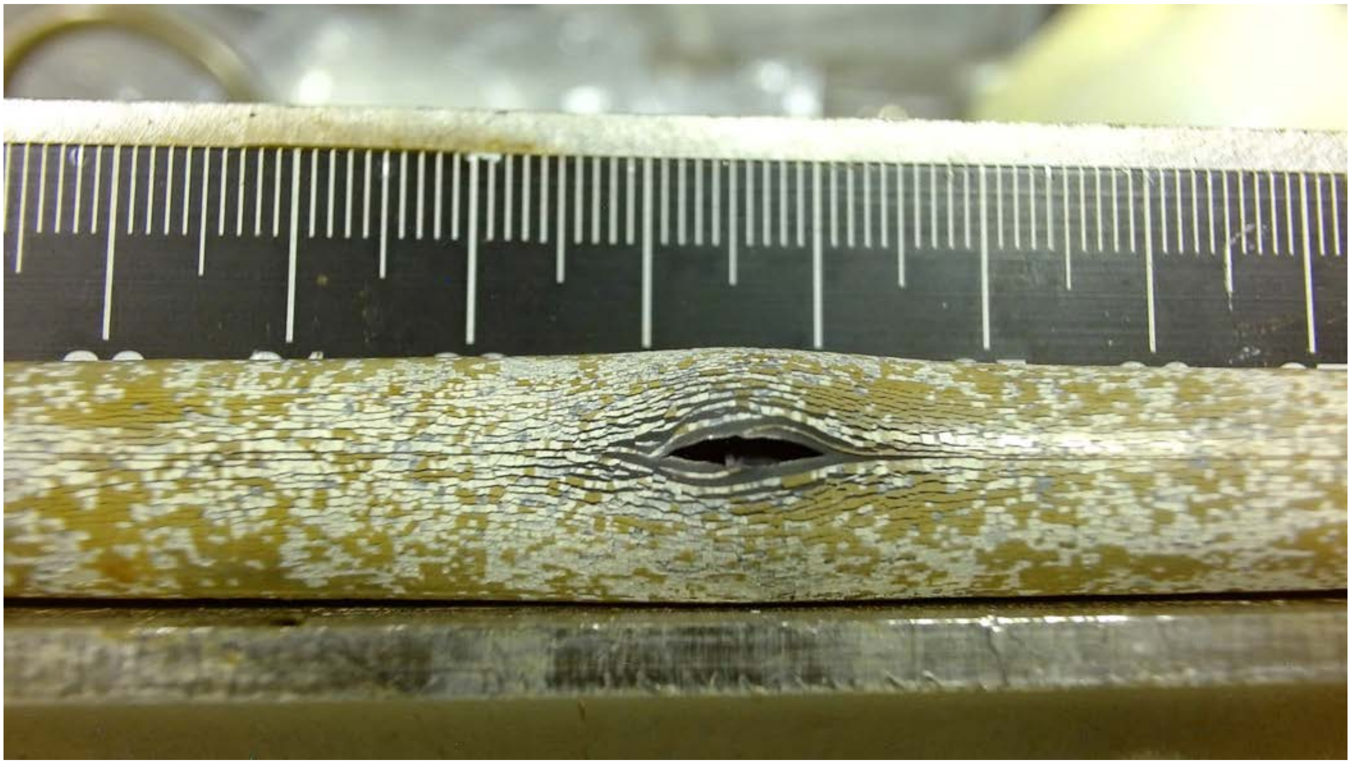
***In collaboration with industry stakeholders, these LOCA test were designed to address critical data gaps in the existing data sets and support stakeholder topical report submittals.***

This summary details two high-burnup Loss of Coolant Accident (LOCA) tests designed to evaluate the effects of terminal temperature and grid spacers (or cladding restraints) on fuel fragmentation and relocation susceptibility. In addition, BISON fuel performance modeling and out-of-cell benchmark testing were conducted to better interpret the in-cell test results. These tests were developed in collaboration with fuel vendors to ensure that the results provide valuable data for topical reports and support the Nuclear Regulatory Commission's (NRC) review.

### Project Description

To improve fuel cycle economics, increasing the fuel burnup limit in light-water reactors requires a solid technical foundation. Observations from experiments at Halden and Studsvik highlighted severe fuel fragmentation during LOCA conditions, indicating a need for further technical considerations. ORNL performed two tests specifically aimed at addressing critical data gaps in the current high-burnup LOCA data set. Historically, LOCA testing has replicated the high-temperature oxidation tests conducted by Argonne National Laboratory. These tests involved heating cladding segments from 300°C to 1000–1200°C at a rate of 5°C/s, simulating the "hot rod" conditions during a large break LOCA. However, high-burnup fuel operates at lower powers and is unlikely to reach temperatures comparable to those in hot rod tests. Testing high-burnup fuel rods under these traditional conditions could yield overly conservative results. Consequently, the first LOCA test documented in this report focuses on terminal temperatures that are more representative of high-burnup LOCA conditions.

The second LOCA test described in this report examined how cladding structural features influence cladding deformation and how this deformation affects fuel fragmentation and relocation. Recent analytical results suggest that cladding deformation during a LOCA is locally reduced near grid spacers or mixing veins. This behavior is attributed to the reduced local power due to these structural features and the enhanced mixing they promote. As a result, temperatures in these regions are lower, which limits cladding deformation. The analysis also indicates that these assembly structural features may impose additional mechanical constraints, further reducing deformation and potentially limiting fuel fragmentation and relocation. The purpose



of this experiment was to validate these analytical findings and provide essential data to support high burnup safety assessments. The final LOCA test intends to look at the impact of double heating on LOCA fuel fragmentation relocation and dispersal (FFRD) behavior.

### Accomplishments

Three loss-of-coolant tests were conducted at ORNL's Severe Accident Test Station facility to address data gaps and key questions concerning the industry's high-burnup safety case. The first test aimed to evaluate the impact of terminal temperature on FFRD. For this test, material from the same rods was used, the only variation being the length of the rodlets: one was 30 cm long, and the other was approximately 20.5 cm. In the tests, HBR#1 was heated to a terminal temperature of 1000°C, whereas HBR#2 was heated to 900°C. HBR#2 exhibited significantly less FFRD compared to HBR#1, despite the shorter rodlet length (see Figure 1). However, HBR#2 also had a rupture temperature of approximately 100°C higher than that of HBR#1. BISON simulations were employed to investigate the differences in cladding performance between the two samples. It was found that HBR#2 experienced gas communication issues between the rodlet and the gas in the lines where the pressure transducer was placed. This issue led to a reduced gas volume during the LOCA test and ultimately resulted in an increased cladding rupture temperature for HBR#2. Consequently, it remains uncertain whether the observed differences

*Figure 1. HBR#2 post-test images a) complete sample, b) bottom section post sectioning, and c) top section post sectioning.*

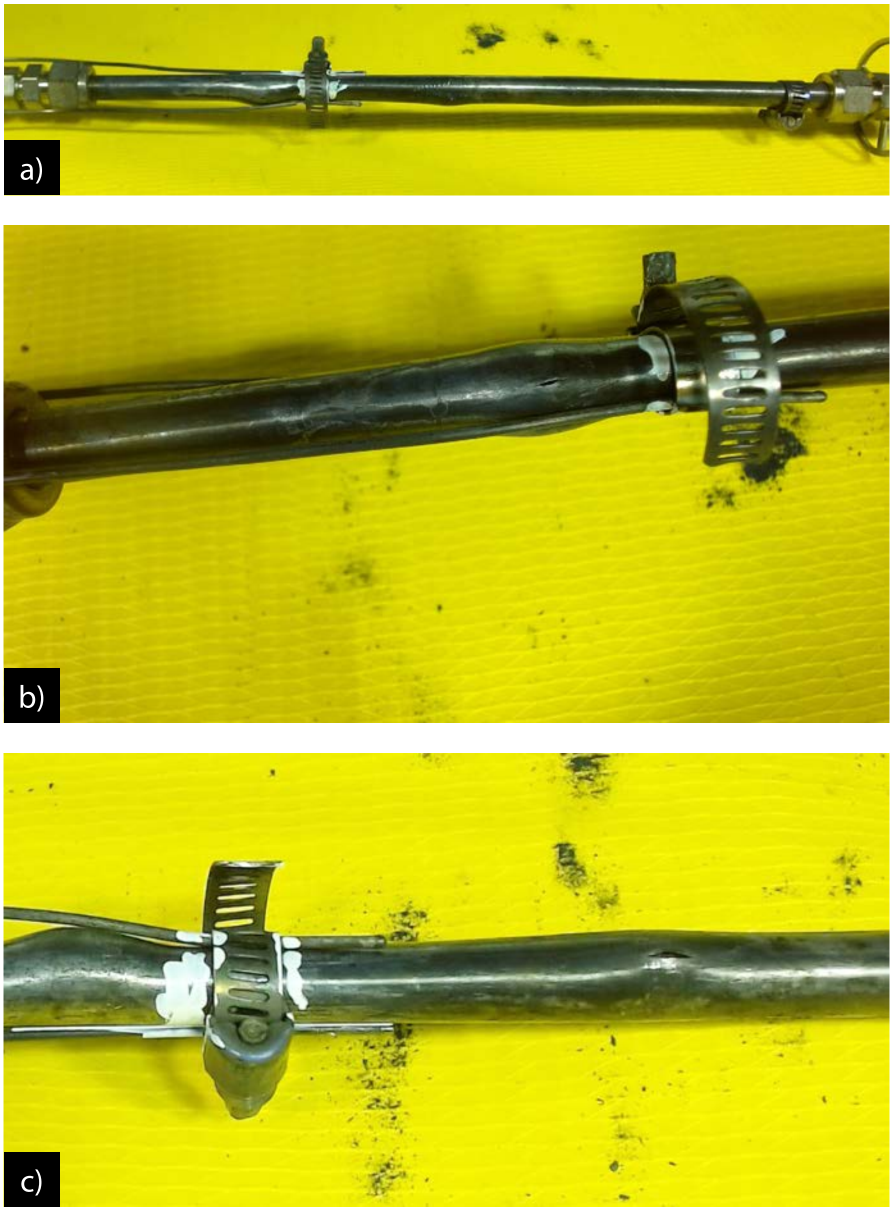


Figure 2. Post LOCA images of the NA grid spacer. a) The whole rod indicating a balloon and rupture on each side of the clamp, b) close up image of the top rupture, and c) close up image of the bottom rupture.

in FFRD behavior were due to the terminal temperature or the increased rupture temperature.

The second test was one of the most complex LOCA tests conducted to date. Its purpose was to evaluate how assembly structural features influence FFRD behavior. These structural features are designed to provide mechanical stability and enhance heat transfer. To ensure that the test is representative, out-of-cell experiments were conducted to evaluate various methods for simulating a grid spacer and to confirm that cladding conditions at and around the grid spacer matched expectations. The results from these out-of-cell tests were compared with experimental data and Cobra-TF (CTF) simulations, and the measured cladding temperatures fell within the ranges reported by both the thermal hydraulic experiments and CTF simulations. Consequently, a ring clamp was selected for the in-cell test. The in-cell test, which included the complex LOCA test train with the ring clamp, also recorded time-dependent transient fission gas release (tFGR) at the moment of burst. These data are intended to help the industry assess how tFGR might impact cladding burst.

The in-cell test was successfully conducted under conditions similar to those reported previously, see Figure 2. As before, the North Anna (NA) grid spacer ruptured at a higher-than-expected temperature. Two rupture openings were observed, and three balloons were ultimately measured. tFGR measurements were successfully obtained and appeared to be lower than those

reported in the NRC FFRD Research Information letter [1] under similar burnup conditions. Fuel sieving was performed in multiple stages to evaluate dispersal at the time of rupture, and fuel was removed from both the upper and lower segments of the rodlet to assess the impact of the double rupture. Post irradiation examination (PIE) will continue with the objective of studying fuel behavior in the vicinity, which will involve optical microscopy. A section diagram has been developed to guide the completion of PIE on this rodlet.

The final test evaluated the effects of pre-heating on FFRD (Figure 3). The sample, obtained from the 6XV Byron reactor that had been previously irradiated, was heated to 1000°C without internal rod pressure to investigate whether tFGR would lead to cladding deformation, ballooning, or rupture. No such deformation was observed, and the cladding's outer diameter remained consistent with measurements taken before heating. Following this, the sample was pressurized to 11 MPa and heated until rupture occurred. The rupture event caused significant bending of the sample, likely due to the high pressure. Due to time constraints, not all post-test data could be collected, but further data collection is anticipated and will be detailed in a future peer-reviewed publication.

*Figure 3. Image of sample 6XV-5E following LOCA testing.*

## References


[1.] Bales, M., Chung, A., Corson, J., Kyriazidis, Lucas. (2021). Interpretation of Research on Fuel Fragmentation, Relocation, and Dispersal at High Burnup, NRC: RIL 2021-13, https://www.nrc.gov/docs/ML2131/ML21313A145.pdf.

# Silicon Carbide Modeling Supporting Tube Bowing Evaluations

*Principal Investigator: Chris Petrie (Oak Ridge National Laboratory [ORNL])*
*Team Members/Collaborators: Peter Doyle, Takaaki Koyanagi (All ORNL)*


*Figure 1: Close-up view of one irradiated SiC/SiC tube specimen.*

The deployment of Silicon Carbide (SiC)-based accident-tolerant fuel cladding for light-water reactor (LWR) applications requires validated material models to reliably predict the fuel and cladding performance. One concern with any SiC-based cladding is the potential bowing that may result from thermomechanical and irradiation-induced loads distributed nonuniformly both axially and azimuthally across a single fuel rod during normal operation. These non-uniform loads could result in distortion of some of the rods within the fuel assembly, which could affect coolant flow and/or the ability to insert control rods or control blades.

Bowing is driven by irradiation-induced swelling of SiC, which depends strongly on temperature and is also a function of dose up to the saturation point near 1 displacement per atom (dpa). Although irradiation-induced thermomechanical property changes have been incorporated into previous models, irradiation-induced bowing of SiC/SiC composite tubes has not been experimentally verified, and the model predictions have not been confirmed.

## Project Description

The overarching goal of this research is to develop experimentally validated fuel performance models that predict the bowing of SiC/SiC composite cladding under neutron flux gradients to increase the confidence in deploying SiC/SiC composite fuel cladding that could increase the safety and reliability of the current LWR fleet. This work compared the modeled and experimentally measured bowing of two SiC/SiC composite tubes (provided by General Atomics) that were irradiated with intentional neutron flux gradients of varying magnitudes within the High Flux Isotope Reactor (HFIR). The specimens were measured before and after neutron irradiation using basic dimensional measurements (e.g., calipers), high-resolution profilometry via a light curtain, and a digital camera coupled with a translation stage to map local displacements between laser-engraved markings.

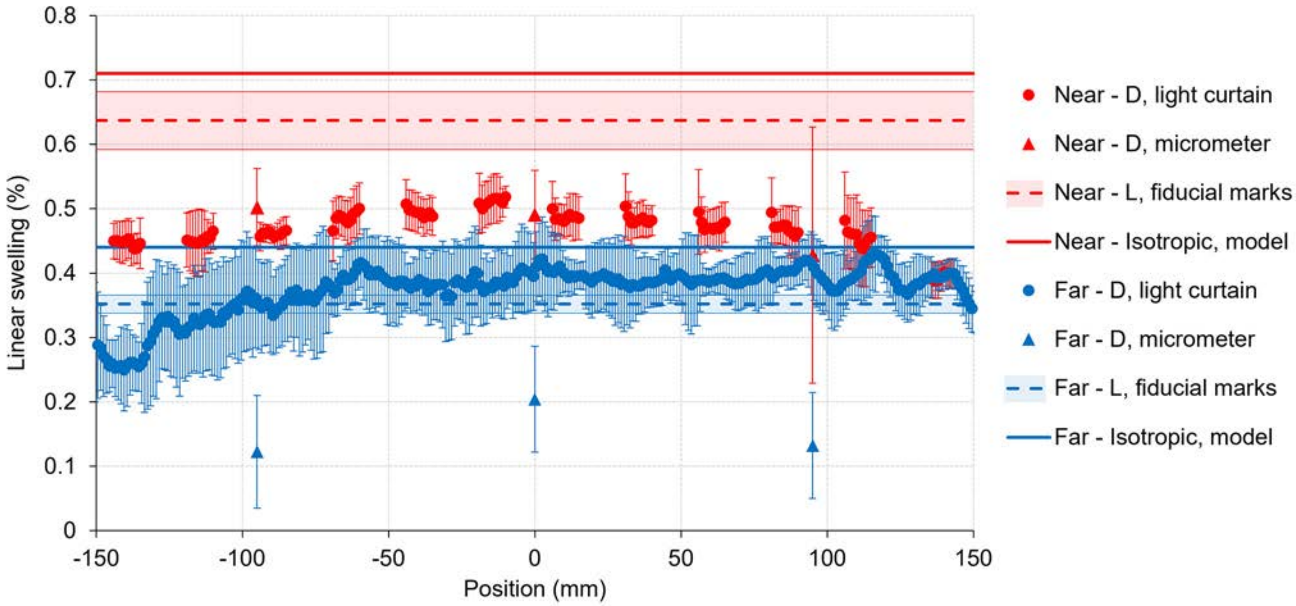



*Figure 2: Measured radiation-induced linear swelling of the specimen lengths (L) and diameters (D), as determined using the techniques specified in the legend. Isotropic swelling calculated using the neutronics model and literature expressions to convert from dose to swelling are provided as a reference.*

In parallel, three-dimensional Monto Carlo neutronics calculations were performed to predict the distribution of neutron flux and displacement damage throughout all specimen materials during the ~50 days of irradiation in HFIR. The specimens were irradiated at HFIR coolant temperatures (~60 °C) to allow for unconstrained deformation that would otherwise be more challenging to accomplish in a dry experiment that relies on properly sized gas gap for temperature control. The dose levels were in the range of 0.04 to 0.12 dpa to prevent amorphization that can occur at these temperatures when the dose levels are closer to 1 dpa. The dose distributions were used as inputs to both analytical equations and three-dimensional finite element analyses (FEAs) to calculate the radiation-induced dimensional changes and bowing, which were compared with the experimental measurements. Additional parametric evaluations were performed to calculate the bowing when adjusting the dose distributions to match the measured swelling to gain insights into potential gaps in the material models that were used as inputs to the simulations.

## Accomplishments

The two irradiated tube specimens were successfully extracted from the irradiation vehicle and showed no signs of damage or surface cracking (see Figure 1). Post-irradiation dimensional inspection (Figure 2) showed a few interesting trends. First, the specimen that was irradiated closer to the HFIR core ("Near," with higher neutron flux gradients) exhibited statistically significant swelling anisotropy,

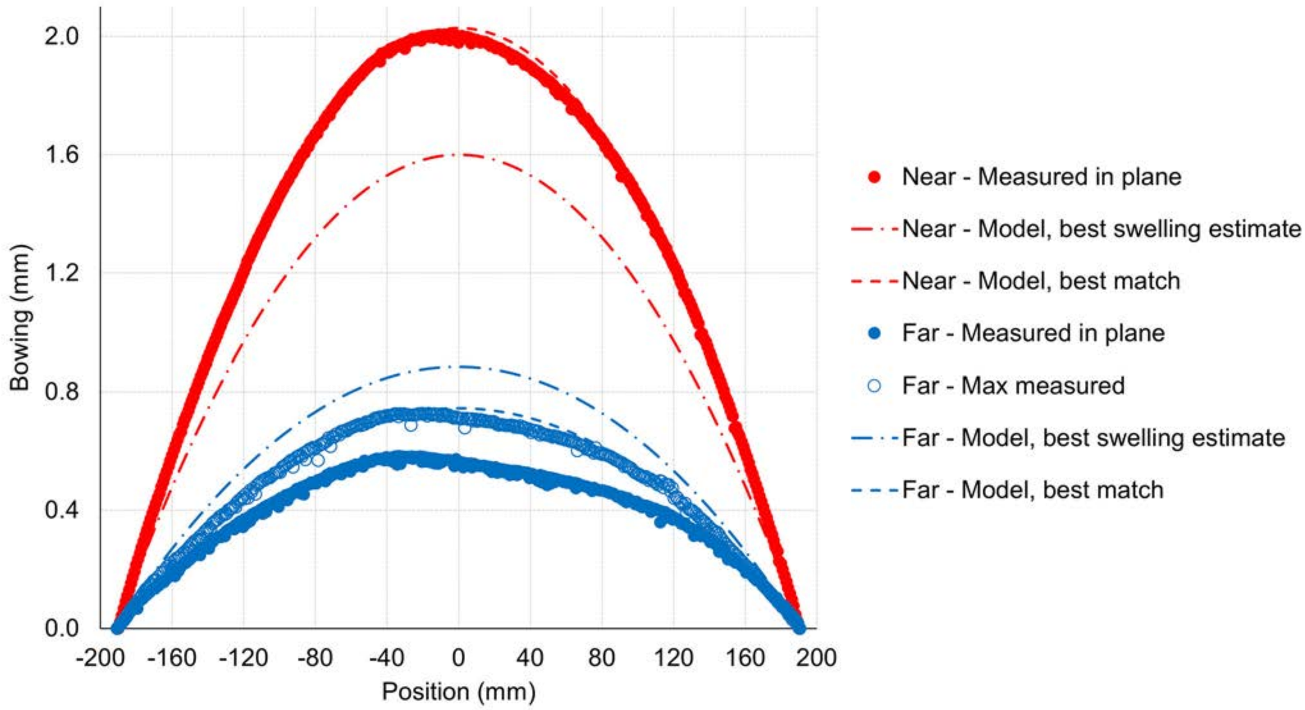



*Figure 3: Measured and calculated (analytical model) bowing in the near and far specimens. Measured data are shown within the plane in which neutron flux gradients were calculated to be maximized and in the plane in which bowing was measured to be highest (45° away) in the far specimen. Model data are shown using dose distributions that results in the best agreement with the measured swelling as well as other cases that match the swelling within experimental uncertainties but also provided the best match to the measured bowing.*

***For the first time, bowing of SiC/SiC components under intentional neutron flux gradients has been measured and used to inform modeling efforts that are required to ensure that bowing of SiC/SiC fuel cladding will not interfere with coolant flow and/or the ability to insert control rods or control blades in operating LWRs.***

with higher length swelling vs. diameter swelling. Swelling anisotropy was not observed in the specimen that was irradiated farther from the HFIR core ("Far," with lower neutron flux gradients), at least when comparing the length measurements with the diameter measurements made using the light curtain. The diameter swelling measurements performed using a micrometer were considered to be less reliable due to the high surface roughness of the specimens. The fact that both specimens exhibited swelling that was slightly less than that predicted by the neutronics model (assuming isotropic swelling) is reasonable considering the uncertainties in neutron transport out to the permanent reflector positions in HFIR, displacement damage cross sections, and dose-to-swelling models that were extracted from the literature. The swelling measurements and their uncertainties were used to adjust the input dose distributions to the FEA models to determine whether the models could capture the measured bowing of each specimen. Length swelling values were used because these values exhibited the lowest uncertainty and because bowing is expected to be driven primarily by differential axial swelling of tube geometries.

Figure 3 shows the measured bowing compared with the values predicted by an analytical model (nearly identical to the FEA results, not shown). The near specimen exhibited reasonable bowing behavior with the maximum bowing occurring in the plane with the highest calculated neutron flux gradients. The magnitudes were also within the range of values calculated using the analytical model when considering the uncertainties in the measured swelling. The best match was obtained when using the maximum near specimen swelling and the minimum far specimen swelling (i.e., the highest differential swelling). In contrast, the far specimen showed the most significant bowing 45° away from the plane within which the maximum swelling was expected to occur and was lower than any of the calculated values. The best match for the far specimen was found when minimizing the differential swelling, which is the opposite of what was found when comparing measured vs. modeled bowing of the near specimen. Moreover, prior to irradiation the far specimen exhibited bowing in some planes as high as 0.33 mm that completely relaxed after irradiation. The lack of agreement between the measured and modeled bowing in the far specimen could be due to contributions from residual stresses that relax (creep) during irradiation, unexpected variations in neutron flux gradients caused by adjacent experiments (not accounted for in the model), anisotropic swelling (also not considered in the model), or a combination of these factors. Future work is underway to repeat these irradiations at more representative LWR temperatures and determine whether the bowing reaches a maximum value near 0.1 dpa (~8 days in a pressurized water reactor) and subsequently decreases.

# Byron Fuel Performance Predictions for Research Applications

*Principal Investigators: Charles Folsom, Robert Armstrong (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Colby Jensen, David Kamerman (All INL)*

***The Byron fuel performance modeling predictions are providing the tools and capabilities to help inform PIE of the parent rods and detailed final design of the planned LOCA and RIA experiments in TREAT.***

High burnup (HBu) fuel rods from the Byron Nuclear Generating Station (BNGS) were recently received at INL to support a variety of planned nuclear energy fuel cycle research and development (R&D) objectives ranging from fuel performance, fuel recycle and spent fuel research topics. Included in these R&D activities is fuel safety testing of HBu fuel where these fuel rods will be the subjects of multiple in-pile experiment programs at the Transient Reactor Test (TREAT) facility as well as detailed characterization and testing in the hot cells at INL and Oak Ridge National Laboratory (ORNL). Specimens refabricated from these fuel rods will be used to support HBu reactivity-initiated accident (RIA) and loss-of-coolant accident (LOCA) experiments to support burnup extension in commercial reactors. The TREAT RIA experiments are planned for the Nuclear Energy Agency Framework for Irradiation Experiments (FIDES) Joint Experimental Program called High burnup Experiments in Reactivity Initiated Accident (HERA) program. TREAT and ORNL-furnace LOCA experiments are part of the Department of Energy (DOE) Advanced Fuels Campaign (AFC) program U.S. consensus LOCA test plan called Loss of Coolant-High Burnup (LOC-HBu).

**Project Description:**

The purpose of this work was to perform fuel performance computational simulations to support the first two LOC-HBu and the first HERA experiments. All three of these experiments will be refabricated from the same parent fuel rod, termed 9EU with a rod average burnup of 62.7 GWd/MTU. The BISON fuel performance code was used to simulate commercial irradiation of the 9EU fuel rod, as well as the TREAT transients to be performed on the refabricated segments. Detailed computational assessment of the full-length fuel rod commercial irradiation and the TREAT experiments inform all stages of ongoing research activities, from post-irradiation examinations (PIE) and refabrication on the full-length rods to the TREAT experiment design, thus ensuring that the targeted conditions are reached.

Due to its inherent structure, BISON cannot directly transfer the as-irradiated condition of the fuel rod segment from the simulation results of the full-length rod.

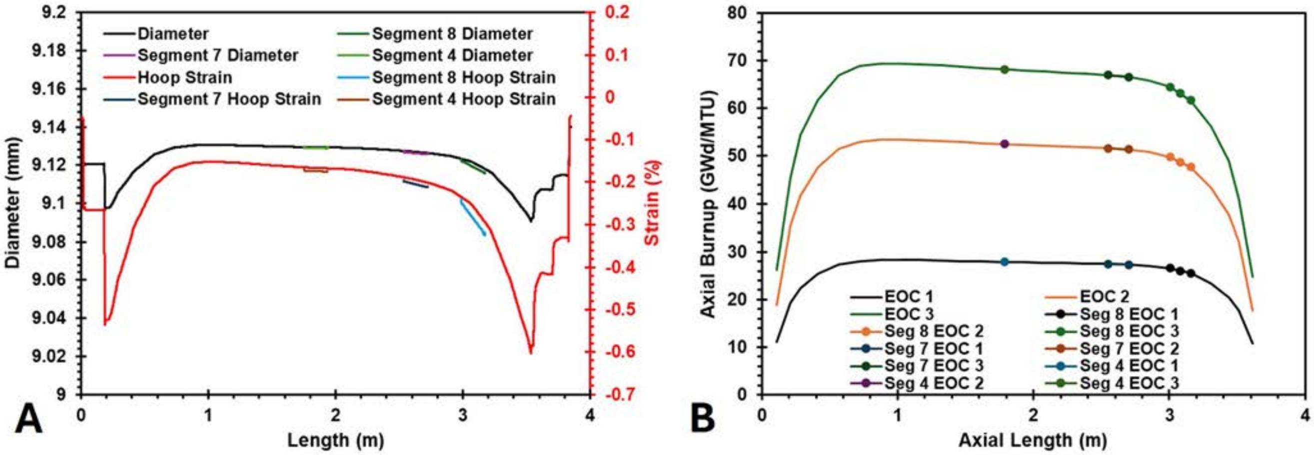



*Figure 1. BISON predictions of 9EU segments 4, 7, and 8 compared against the full-length rod for a) EOL final cladding diameter and hoop strain and b) axial burnup at the end of each cycle.*

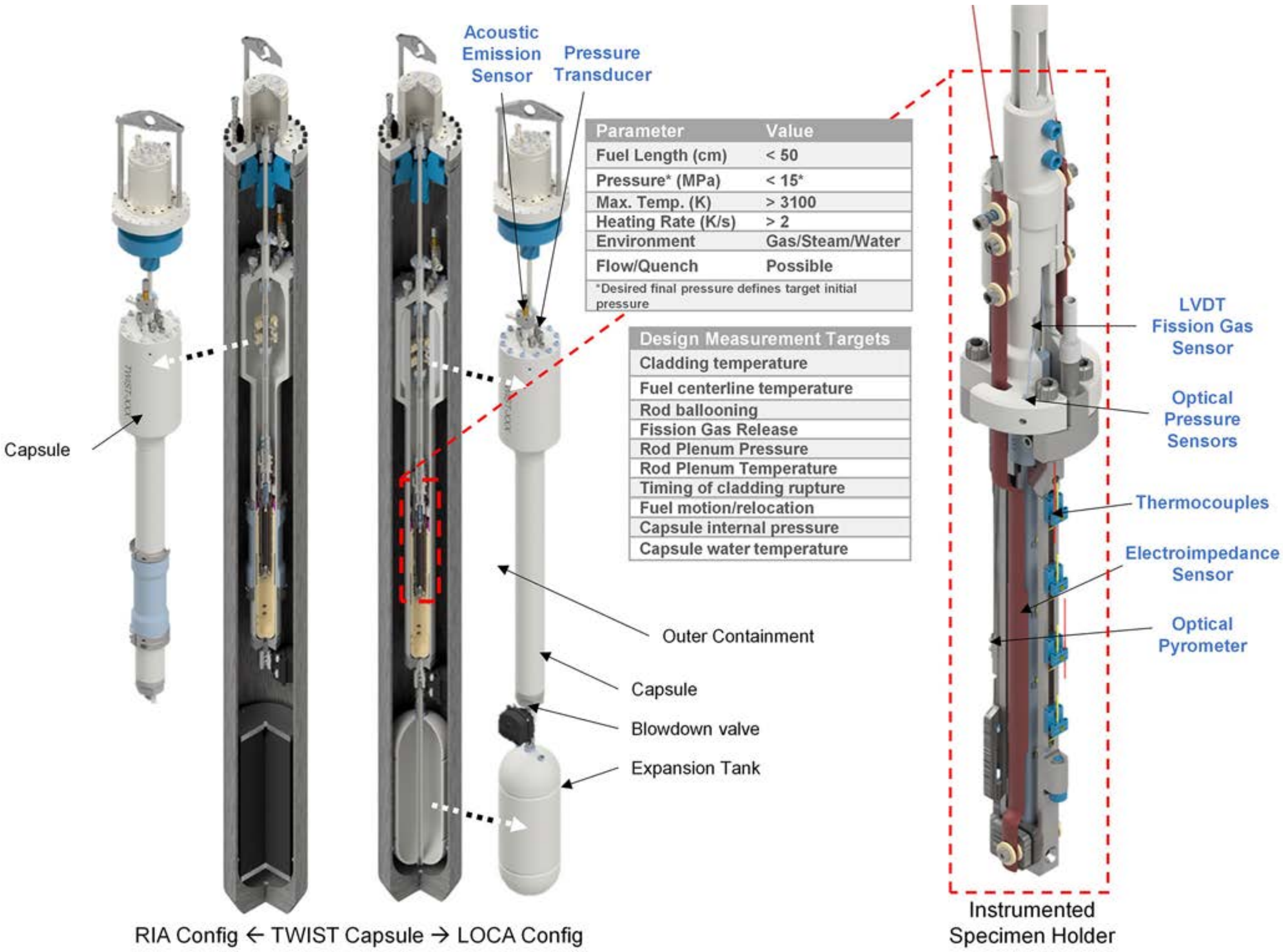
Thus, the following methodology was developed to best transfer the effects of the commercial irradiation performed on the full-length rod over to the test segment so as to simulate the transient experiment behavior on a segment with initial conditions that represent the HBu condition of the rod:

- Simulate the commercial irradiation of the full-length fuel rod in order to capture all life history boundary conditions for the test segments of interest.
- Simulate the commercial irradiation on the test segment length, using the corresponding boundary conditions obtained from step 1.
- Simulate the transient experiment on the as-irradiated condition of the test fuel segment.

The primary objective of the LOC-HBu and HERA experiment campaigns is to study HBu fuel performance under LOCA and RIA conditions, respectively. A primary objective of the LOC experiments is to explore fuel fragmentation, relocation, and dispersal. The HERA experiments are intended to investigate fuel failure thresholds for HBu fuels with several different cladding types, ranging from moderate to low hydrogen pickup.

### Accomplishments

A methodology was developed for obtaining an accurate description of the test segments' end-of-life (EOL) conditions—one that accounts for full length rod effects. This entailed a two-step method to model the full-length rod and then apply the calculated conditions from that simulation to a model devoted solely to the rodlet segment that would eventually be tested in TREAT. The method proved effective at capturing the necessary fuel performance data on the test segments at the start of the TREAT tests. The final EOL cladding diameter and hoop strain for each segment (Figure 1a) and the burnup at the end of each cycle for each segment (Figure 1b) is compared to the full-length rod model with good agreement. The BISON model was also used to predict fission gas release and cladding oxide thickness




| Parameter | Value |
|---|---|
| Fuel Length (cm) | < 50 |
| Pressure* (MPa) | < 15* |
| Max. Temp. (K) | > 3100 |
| Heating Rate (K/s) | > 2 |
| Environment | Gas/Steam/Water |
| Flow/Quench | Possible |

*Desired final pressure defines target initial pressure

| Design Measurement Targets |
|---|
| Cladding temperature |
| Fuel centerline temperature |
| Rod ballooning |
| Fission Gas Release |
| Rod Plenum Pressure |
| Rod Plenum Temperature |
| Timing of cladding rupture |
| Fuel motion/relocation |
| Capsule internal pressure |
| Capsule water temperature |

*Figure 2. Schematic overview, key specifications, instrumentation measurement targets, and test train overview of the TREAT TWIST device showing comparison between RIA and LOCA configurations.*

that can be compared against PIE data to further refine and improve the model.

The LOCA and RIA experiments will be performed within the Transient Water Irradiation System for TREAT (TWIST) capsule. The TWIST capsule is meant to support irradiations simulating LOCAs and RIAs in TREAT. It was designed, deployed, and commissioned using a series of experiments with fresh fuel specimens, called the LOC Commissioning experiments. In the LOCA configuration, the TWIST device functions as a static-water blowdown capsule with two primary test volumes for simulating thermal conditions in the test fuel, from pre-through-post blowdown LOCA conditions. The RIA configuration of TWIST is similar, except that the expansion tank is removed, and a neutron flux collar is added around the capsule exterior to limit the neutron flux to the test specimen under high-energy power pulses in TREAT. Figure 2 gives a schematic overview of the TREAT TWIST capsule design, with tables listing its capabilities.

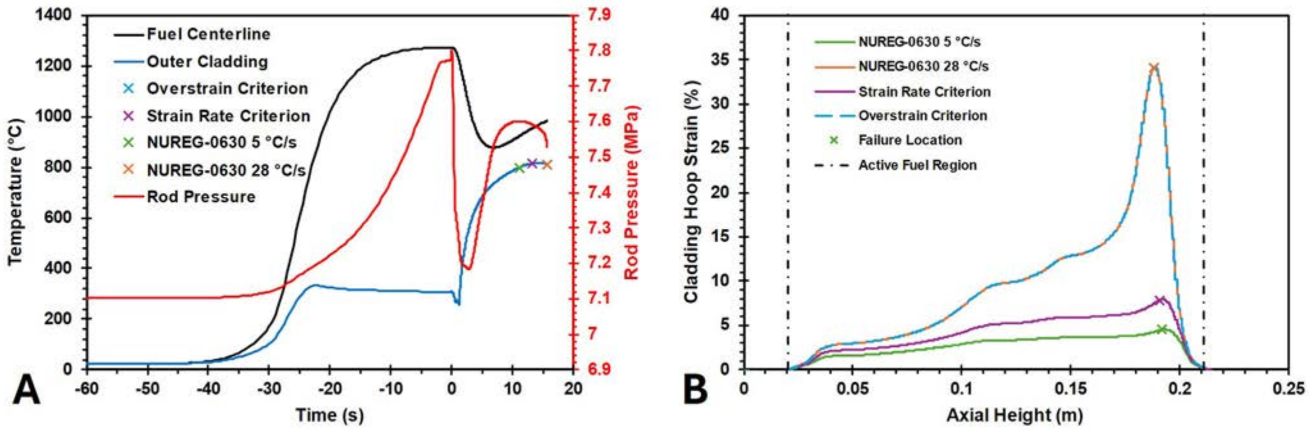



*Figure 3. BISON predictions of the LOC-HBu-2 experiment with initial rod plenum pressure of 7 MPa for a) maximum fuel centerline and outer cladding temperatures, rod pressure, and rupture predictions. Blowdown occurs at time zero. b) BISON prediction showing cladding hoop strain and failure location at predicted time of failure for each failure criterion.*

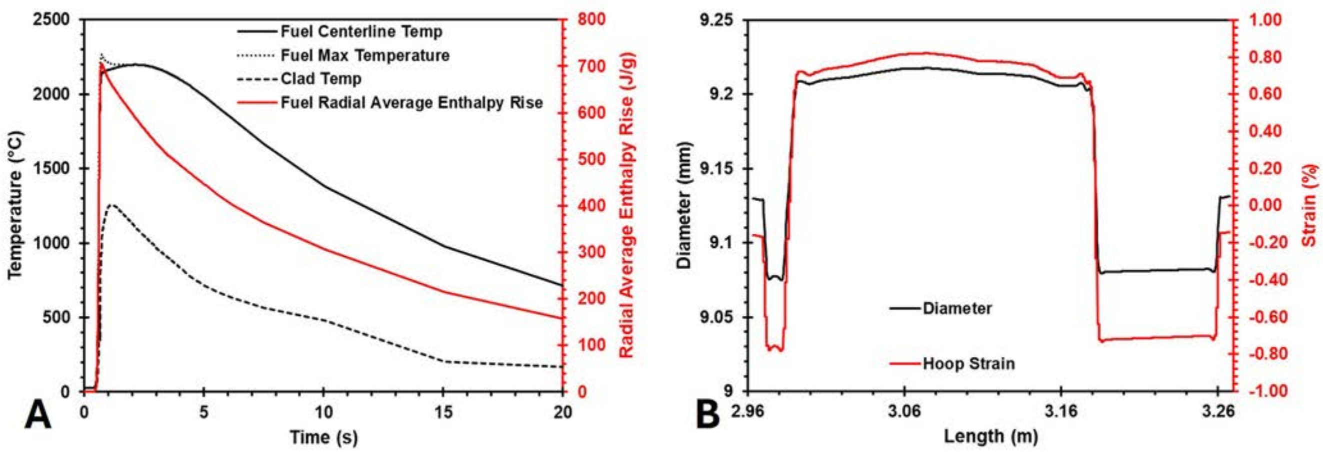



*Figure 4. BISON predictions of the HERA-HBu-1 experiment with initial rod plenum pressure of 1 atm. a) Fuel centerline, max fuel temperature, cladding temperature, and fuel radial average enthalpy rise predictions at the peak axial power location. b) Final rodlet diameter and hoop strain along the length of the rod following the TREAT transient.*

LOC-HBu-2 will be the first HBu LOCA experiment performed to simulate pre-through-post blow-down thermal conditions consistent with those experienced by a light water reactor (LWR) fuel rod at operating conditions transitioning to a LOCA. The experiment configuration begins with the TWIST capsule being filled with water pressurized to 3.5 MPa. In the first 45 seconds, TREAT is brought up to a steady-state power to create a radial temperature profile through the fuel that is consistent with that of an LWR fuel rod during operation. At this point, the controllable valve on TWIST opens to the expansion tank, and the TREAT power is simultaneously decreased. Opening the valve quickly depressurizes and removes liquid water in the upper capsule. With the fuel rod in a steam environment, the stored energy within the fuel redistributes, rapidly increasing the cladding temperature and decreasing the fuel centerline temperature. BISON, along with thermal-hydraulic boundary conditions provided by RELAP5-3D simulate the experiment to provide data on predicted failure time, temperature, and cladding strain. The predictions of fuel centerline, cladding temperature, and rodlet plenum pressure with failure predictions based on different failure criteria in BISON is shown in Figure 3a. The cladding hoop strain at failure is shown in Figure 3b. These simulations showed that the balloon and failure is predicted to occur near the top of the fuel rod, which is not ideal and led to a redesign adding flux shaping features to the TWIST design to reduce the end peaking near the top of the fuel rod.

The first planned RIA experiment is HERA-HBu-1, which will be refabricated from segment 8 of the 9EU rod. The transient was selected to target a peak fuel radial average enthalpy of 650 J/g. The BISON simulation predicted the peak fuel radial average enthalpy rise to be 704 J/g (Figure 4a), which was higher than the targeted 650 J/g. This over-prediction was due to the axial peaking in TREAT. Figure 4a also shows the predicted fuel and cladding temperatures. The peak fuel temperature occurs in the periphery of the fuel due to the HBu structure and higher Pu content in the outer rim. BISON also predicts the post-transient final clad hoop strain at 0.8% (Figure 4b) with a peak cladding hoop strain occurring during the transient of 1.7%, and a peak cladding hoop stress of 220 MPa.

This modeling exercise and capability developments are providing valuable tools used to design the experiment conditions and transients necessary to target the desired conditions and possible outcomes for the experiments. This work will continue for the other BNGS fuel rods and LOC-HBu and HERA experiments.


LINEAR BEARING CONNECTOR PLATE

Nuclear Energy University Project (NEUP) Award

# Post-departure from Nucleate Boiling Thermo-mechanical Behavior of Near-term Accident Tolerant Fuel Designs in Simulated Transient Conditions

*Principal Investigator: WooHyun Jung (University of Wisconsin, Madison [UWM])*
*Team Members/Collaborators: Hwasung Yeom (Pohang University of Science and Technology [POSTECH]); Kumar Sridharan, Michael Corradini (All UWM); David Kamerman, Robert Armstrong (All Idaho National Laboratory [INL]); Raul Rebak; Andrew Hoffman (All General Electric [GE])*

The Cr-coated Zr-alloy cladding is the most promising near-term accident tolerant fuel (ATF) cladding, which is being actively developed to provide additional safety margin by its excellent corrosion/oxidation resistance under accident conditions and to provide economic benefits by allowing for longer fuel cycle or power uprate. Thus, this research aims to experimentally evaluate the thermo-mechanical performance of Cr-coated cladding with simulated high burnup fuels in the reflood phase of loss of coolant (LOCA) accident. The coolant quenches the hot cladding during the reflood, and the quench temperature is a key parameter determining whether the peak cladding temperature is below the safety guideline. Furthermore, the post-quench ductility is another important parameter representing the integrity of the cladding during the accident, and even after the accident, which is critical in terms of transporting the post-accident fuel rods. This work will extend our knowledge of the Cr-coated cladding under LOCA in both thermal-hydraulic behavior and post-quench ductility.

## Project Description

The objective of the research is to investigate the thermal-hydraulic and thermo-mechanical response of Cr-coated ZIRLO under high-temperature reflood conditions up to 1200 °C, in comparison to uncoated cladding. The objective is achieved by bottom reflood tests on the uncoated, cold spray (CS) Cr-coated, and physical vapor deposition (PVD) Cr-coated ZIRLO tubes followed by the ring compression tests to evaluate the post-quench ductility. Previous research on reflood quenching experiments was conducted with Cr-coated cladding under a limited initial temperature range below 800 °C, despite the possibility of higher peak cladding temperature during the design basis accident up to 1200 °C. Thus, thermal-hydraulic response and thermo-mechanical behavior of post-quenched Cr-coated sample at high temperature reflood conditions were experimentally investigated in this project.

Reflood tests have been performed using the single rod reflood facility at the UWM to study the quench temperature and post-quench ductility of Cr-coated cladding compared to the uncoated cladding. The quench tests were performed

**Cr-coated ZIRLO cladding shows an improved thermal-hydraulic and thermo-mechanical response compared to uncoated cladding under the oxidative environments and thermally-cycled oxidation conditions in terms of both quench temperature and post-quench ductility.**

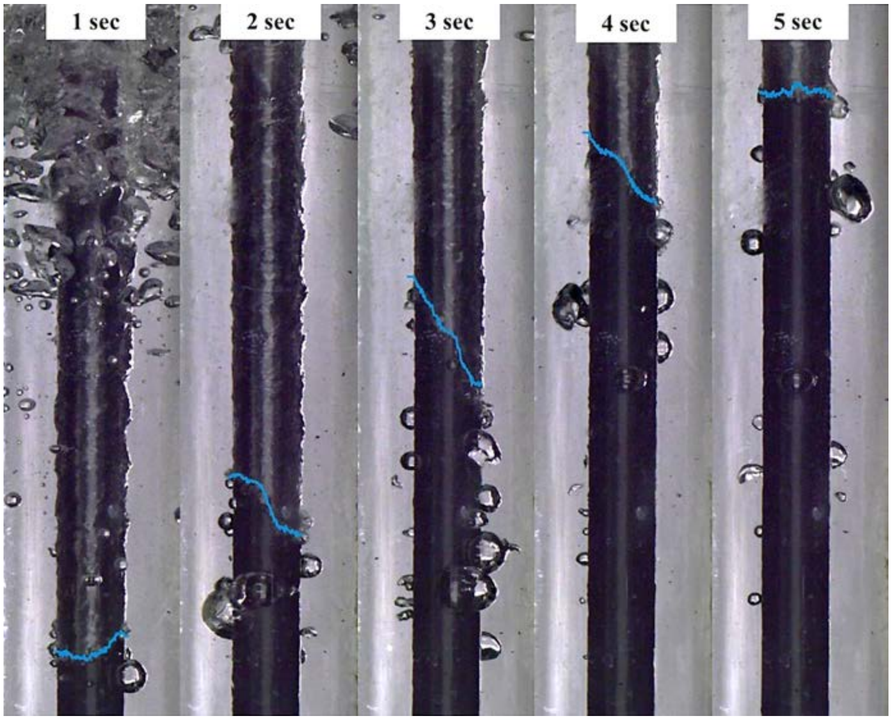



*Figure 1. Quench front position at several timesteps during reflood with ZIRLO cladding.*

under various initial temperatures (600-1200 °C) by measuring the cladding temperature using thermocouples inserted inside the cladding at four different axial locations. Moreover, repeated quench tests were performed with the uncoated and CS Cr-coated claddings to examine the Cr-coated cladding under thermally-cycled oxidation conditions for the purpose of observing the evolution of quench temperature in relation to increasing surface oxidation and examining the mechanical integrity of Cr-coating under severe oxidation and thermal cycle. The result of this project will be used for anticipated licensing applications to the U.S. Nuclear Regulatory Commission, thereby accelerating the deployment of Cr-coated ATF concepts in U.S. commercial light water reactor (LWR) fleets.

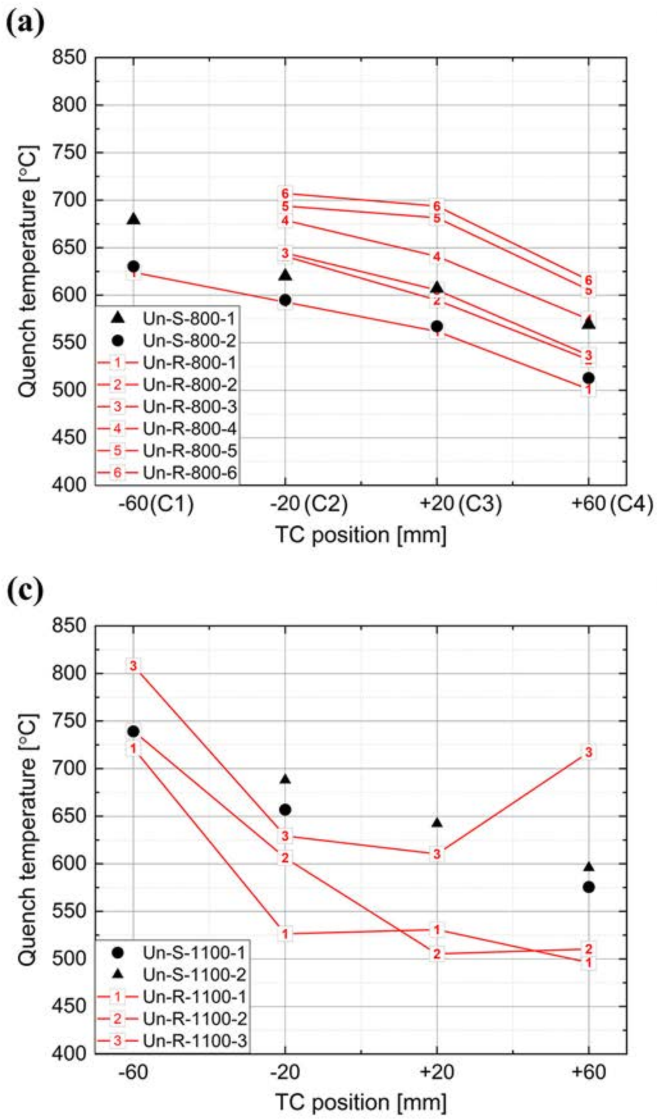



*Figure 2. Change in quench temperatures in repeated quench tests with different initial conditions and thermocouple positions using the uncoated and CS Cr-coated ZIRLO.*

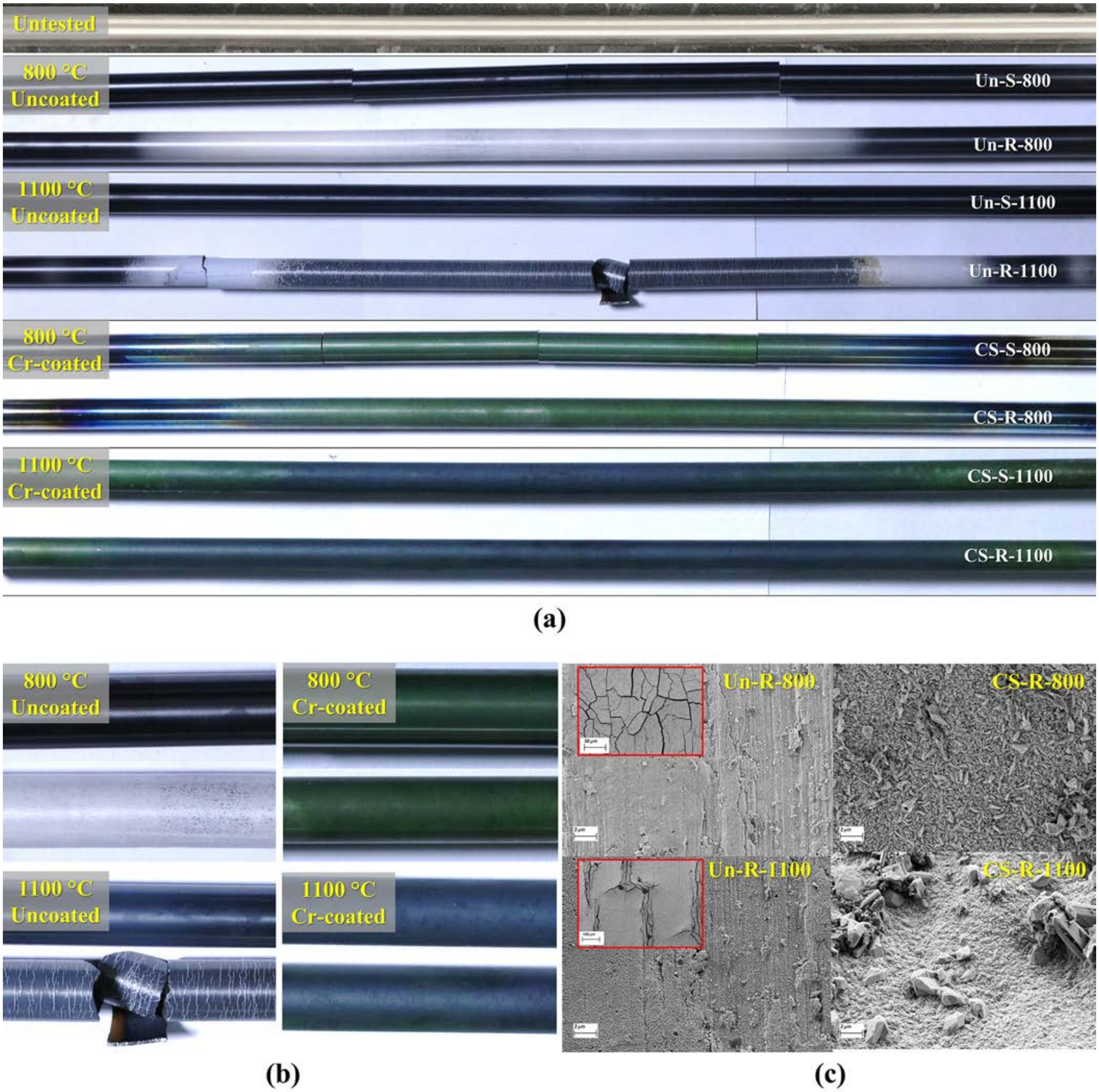



*Figure 3. Comparison of post-quench surface conditions after single quench and repeated quench tests: (a) photographs of post-test claddings, (b) magnified view at the center, and (c) plan-view scanning electron microscopy images of the repeated quench samples with the low-magnification image in the red box to observe the crack morphology.*

## Accomplishments

To achieve the objective of this research, five primary task areas have been identified: (1) construction of single rod reflood facility, (2) preparation of Cr-coated claddings, (3) quench tests for uncoated and Cr-coated ZIRLO, (4) ring compression tests for post-quenched samples, and (5) analysis of data and modeling. In the first stage of the project, the single rod reflood facility was constructed at the UWM (Figure 1) to evaluate the performance of Cr-coated cladding under the simulated high temperature LOCA condition up to 1200 °C. ZIRLO was used as the cladding substrate, which was provided by Westinghouse, and the Cr-coating was deposited on it through two different methods: CS and High Power Impulse Magnetron Sputtering method. The CS Cr-coated ZIRLO was fabricated at the UWM using the in-house cold spray deposition facility and PVD Cr-coated ZIRLO was fabricated by the commercial vendor.

A total of 22 of single quench tests and four sets of repeated quench tests were performed using three different types of surface conditions: uncoated, CS Cr-coated, and PVD Cr-coated. The initial cladding temperature for the quench tests was varied from 600 to 1200 °C under 20 K water subcooling and 4.5 cm/s reflood velocity. In the case of the repeated quench tests, it was tested only under 800 and 1100 °C initial cladding temperature conditions. The quench temperatures were obtained using the maximum curvature method, which was developed in this project to capture the quench temperatures even with the bottom reflood condition showing no minimum heat flux point.

Through extensive analysis of quench temperature results, the gap in predicted quench temperatures by Kim and Lee's correlation [1] and our experimental results were attributed to the thinner wall thickness of Zr-alloy cladding (0.57 mm) than the correlations applicable range leading to a lower probability of recovery of the local cold spot from the intermittent liquid-solid contact resulting in the earlier quenching (higher quench temperature). Based on experimental data, Kim and Lee's correlation has been modified for applicability in more prototypical LWR Zr-alloy cladding

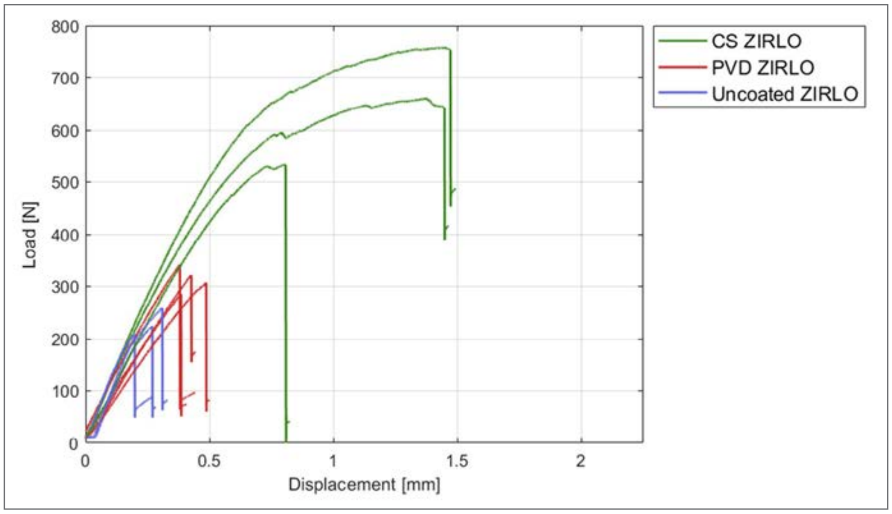



*Figure 4. Load-displacement curve of post-quench claddings at 1200 °C initial cladding temperature with three different coatings: uncoated, CS Cr-coated, and PVC Cr-coated ZIRLO.*

specifications. Repeated quench testing showed that the quench temperature of uncoated ZIRLO increased with the number of repeating quench cycles potentially due to the continued oxide layer development. However, the quench temperature of Cr-coated ZIRLO remained constant due to its high oxidation resistance, as shown in Figure 2.

The post-test analysis was conducted to investigate the oxide surface morphology and overall behavior. The post-quench uncoated ZIRLO shows severe damage in the surface after both six repeated quench at 800 °C or three repeated quench at 1100 °C compared to the single quench (Figure 3), but Cr-coated ZIRLO shows no discernable difference change in the surface condition even after all repeated quench tests.

Finally, ring compression tests were conducted to evaluate the post-quench ductility. Figure 4 shows the load-displacement curve of three types of cladding after a single quench at 1200 °C. The CS Cr-coated ZIRLO shows a significant improvement in post-quench ductility than the uncoated or even PVD Cr-coated ZIRLO. Further analysis is undergoing to investigate why Cr-coated claddings exhibit superior post-quench ductility.

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[1.] A.K. Kim, A.K. & Y. Lee (1979) A correlation of Rewetting Temperature, Lett. Heat Mass Transfer, Volume 6 (2), pp. 117-123.

Nuclear Energy University Project (NEUP) Award

# Fuel-to-Coolant Thermomechanical Behaviors Under Transient Conditions

*Principal Investigator: Nicholas R. Brown (University of Tennessee, Knoxville [UTK])*
*Team Members/Collaborators: Nicholas Meehan (UTK), Seok Bin Seo (UTK and Idaho National Laboratory [INL]), Trevor Howard (Oregon State University [OSU]), Charles Folsom (Idaho National Laboratory [INL])*

We have been working to enhance the prediction of thermo-mechanical fuel-to-coolant (F2C) heat transfer under transient conditions by using a coupled analysis and experiment approach. We are focused on thermo-mechanical and thermal hydraulic tools that will result in much more accurate prediction of F2C heat transfer during anticipated operational occurrences and design basis accidents of interest to the nuclear industry.

***Critical heat flux is a major factor in safety evaluation and analysis of light water reactors, so our efforts support both power uprate and enhanced accident tolerance of advanced nuclear fuel and high burnup fuel.***

## Project Description

Our project enhances the modeling of critical heat flux (CHF) in nuclear safety analysis tools by modifying the models to mechanistically account for important factors, including transient effects. This project is highly synergistic with integral fuel tests being conducted using the Transient Reactor Test (TREAT) facility to elucidate F2C heat transfer characteristics. We have used some of these TREAT experiments in Meehan et al. [1] to improve the modeling of reactivity accidents in nuclear safety codes. Our approach is crucial to validate and develop representative fuel systems models. We leverage past experience with thermo-mechanical (e.g., BISON) and thermal-hydraulic (e.g., RELAP5-3D) tools, but also use the features of RELAP5-3D related to tailoring of the boiling curve to assist with thermal hydraulic model calibration.

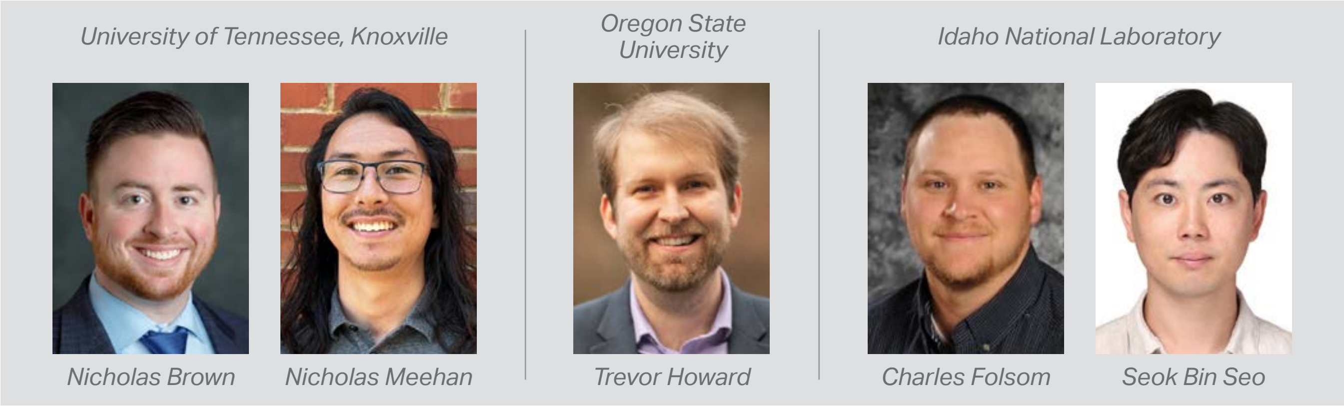



*Figure 1. Research team members from University of Tennessee, Idaho National Laboratory, and Oregon State University*

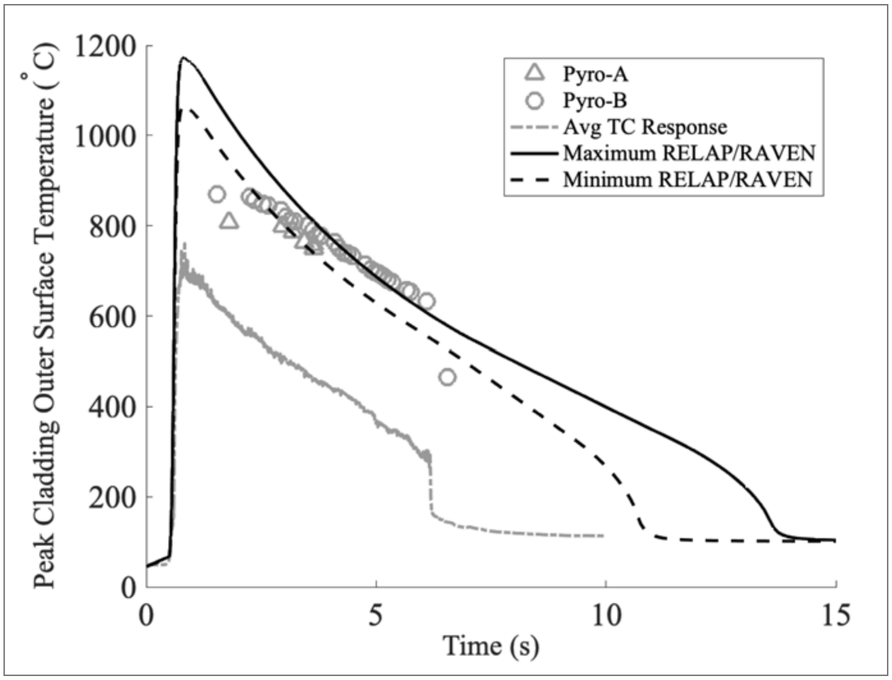
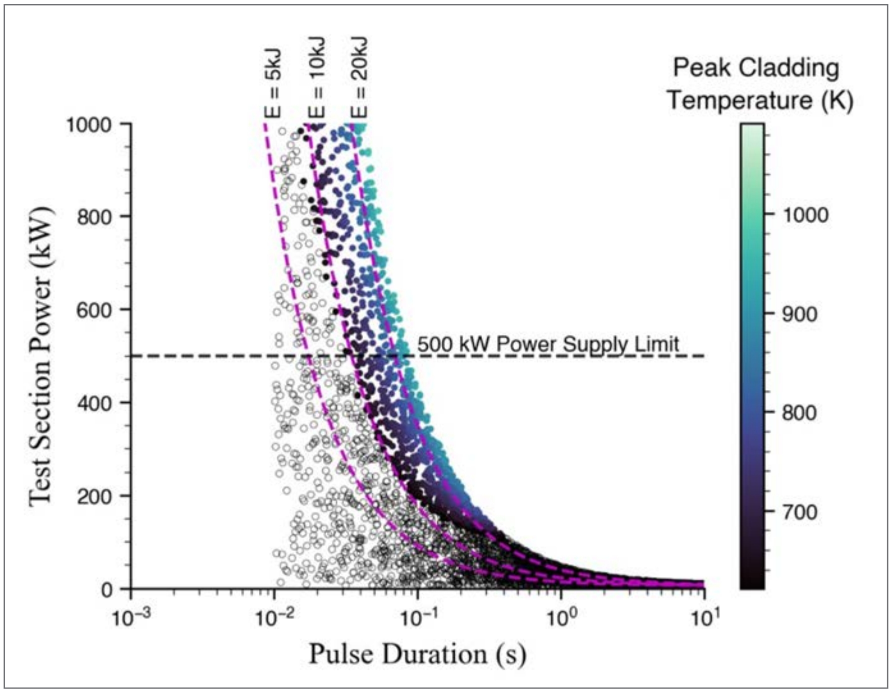



*Figure 2. Developed performance envelope model comparison to experimental results for CHF-SERTTA-B within TREAT with experimental data from Armstrong et al. [3] and Folsom et al. [4].*

Figure 3. Proposed CHF performance envelopes for TRTL power pulses.




## Accomplishments

The project team (Figure 1) compared several experimental pool and flow boiling critical heat flux experiments with corresponding computational models. We conducted several sensitivity analyses for the boiling heat transfer correlations, CHF prediction, and the thermal properties of the cladding for experiments in the Transient Reactor Test Loop (TRTL; [2]) and TREAT facilities [1].

The sensitivity analyses were consistent in determining that there is a large contribution from the critical heat flux which in this application mainly accounts for the effects of power transient critical heat flux. Comparison to experimental results of both the TREAT CHF- Static Environment Rodlet Transient Test Apparatus (SERTTA) experimental campaign (Figure 2) and a University of New Mexico pool boiling experimental campaign from the literature demonstrated both the under-prediction of transient CHF and transient nucleate boiling heat transfer. To address the gaps in knowledge regarding the transient heating effect for both boiling heat transfer and CHF prediction, we presented an experimental test matrix for the TRTL facility (Figure 3) that will isolate the power transient effect under relevant pressure and flow conditions. The test matrix was designed to incorporate several power pulses with a constant energy deposition and a timescale that spans over several orders of magnitude, including a steady state test with an exponential period of 10s.

### Reference


[1.] Meehan, N.A., Folsom, C.P. and N.R. Brown (2024). "A knowledge gap analysis for transient CHF prediction within RELAP5-3D." Nuclear Engineering and Design, 422, p.113171, https://doi.org/10.1016/j.nucengdes.2024.113171.

[2.] Meehan, N.A., Seo, S.B., Howard, T.K. and N.R. Brown (2023). "Sensitivity Analysis of Transient Critical Heat Flux by RIA Under High-Pressure Flow Boiling Conditions in TRTL." Nuclear Technology, 209, p. 1164, https://doi.org/10.1080/00295450.2023.2195355.

[3.] Armstrong, R. J. et al., 2021. "Results of the CHF-SERTTA In-Pile Transient Boiling Experiments at TREAT." TopFuel 2021, Santander, Spain.

[4.] Folsom, C. P. et al., 2022. Design of separate-effects In-Pile transient boiling experiments at the TREAT Facility." Nuclear Engineering and Design, 397, p. 111919.

Nuclear Energy University Project (NEUP) Award

# 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

*Principal Investigator: Matteo Bucci (Massachusetts Institute of Technology [MIT])*
*Team Members/Collaborators: Mark Anderson, Tiago Moreira (University of Wisconsin, Madison [UWM]); Charlie Folsom (Idaho National Laboratory [INL])*

Cladding-to-coolant heat transfer during transient conditions poses a significant uncertainty in nuclear reactor safety, particularly affecting accident progression and fuel performance predictions. In reactivity-initiated accidents (RIA), current thermal-hydraulic models result in large temperature variations (~600K), leading to conservative predictions and wide safety margins. Improved understanding of the transitions to film boiling could enhance safety margins in light water reactors (LWRs). However, current models fail to accurately predict behavior during rapid temperature increases. This proposal aims to close this knowledge gap through high-resolution, controlled experiments to develop accurate models and correlations for use in nuclear system design and safety analysis tools, and relevant to both the Transient Reactor Test Facility (TREAT) testing program and the existing pressurized water reactor (PWR) fleet.

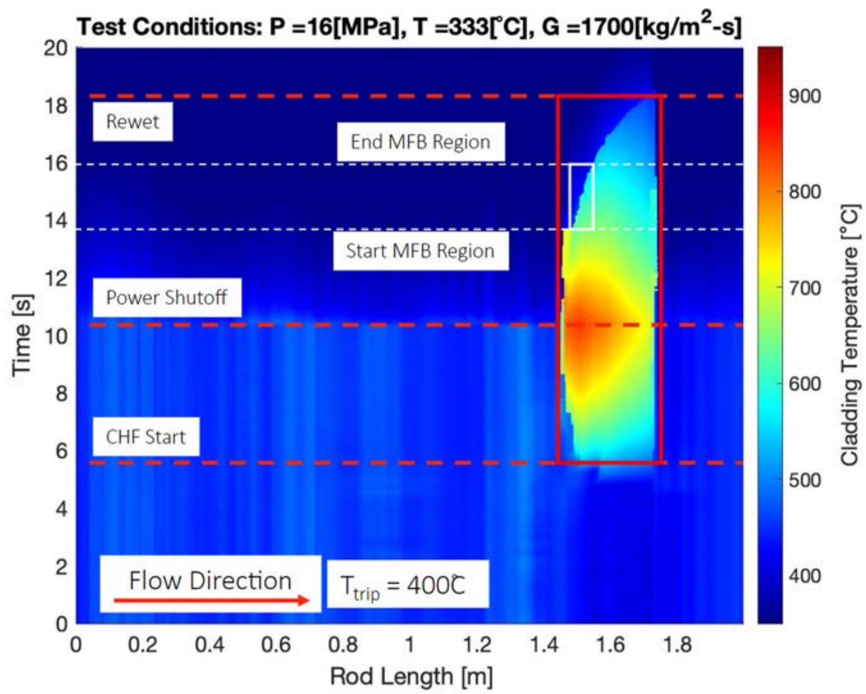



*Figure 1. Optical fiber temperature data during a DNB event under prototypical PWR conditions detailing regions used to get the rewet time, peak cladding temperature and minimum film boiling temperature.*

> ***This project elucidates the mechanisms governing transient heat transfer phenomena, which cannot be predicted using steady-state correlations and mechanistic models.***

## Project Description

The objective of this project is to:

1. Elucidate heat transfer phenomena between the fuel cladding and the coolant in transient conditions
    a. Focusing on critical heat flux (CHF) and post-CHF conditions
    b. Considering
        i. Exponential power escalations and rapid power pulses, representative of RIA, up to PWR pressures and at several flow rates and subcooling degrees,
        ii. Anticipated Operational Occurrence, such as BWR power pulses following a turbine trip,
        iii. Reflooding scenarios (e.g., Loss of Coolant Accident (LOCA) scenarios),
        iv. Post-CHF heat transfer at ambient pressure and PWR conditions.
    c. Considering conventional Zircaloy claddings and accident tolerant fuel concepts, with a focus on Cr-coated Zircaloy.
2. Generate a comprehensive database to critically evaluate existing models and correlations, and benchmark modeling tools developed by Nuclear Energy Advanced Modeling and Simulation and/or in use by the nuclear community, including thermal-hydraulics codes (e.g., CTF and RELAP5) and fuel performance codes (e.g., BISON).
3. Verify fundamental hypothesis and develop physical models and correlation to model these phenomena with the aforesaid computational tools.
4. Support the LWR fleet and the TREAT research programs with new understanding, targeted experimental results, and modeling with new and representative physical models and correlations.

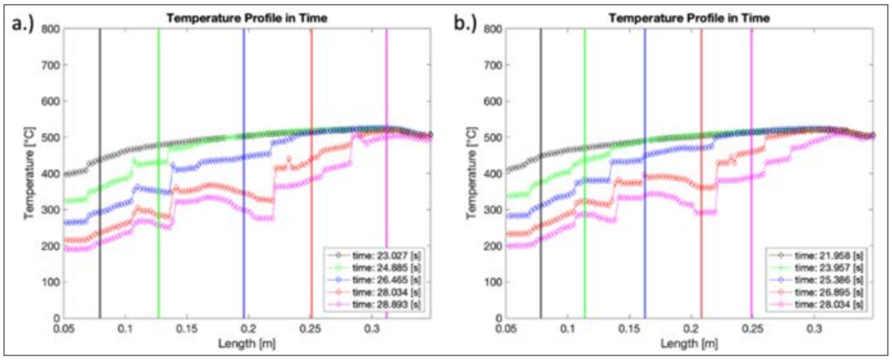



*Figure 2. Temperature profile for a.) Rod 1 (Ra=0.19 μm) and b.) Rod 2 (Ra=2.33 μm) at 500˚ C, dT/dt of 1˚ C/s and inlet temperature of 20˚ C. The vertical lines indicate the water level obtained from the coolant channel fiber at that time (matching colors).*

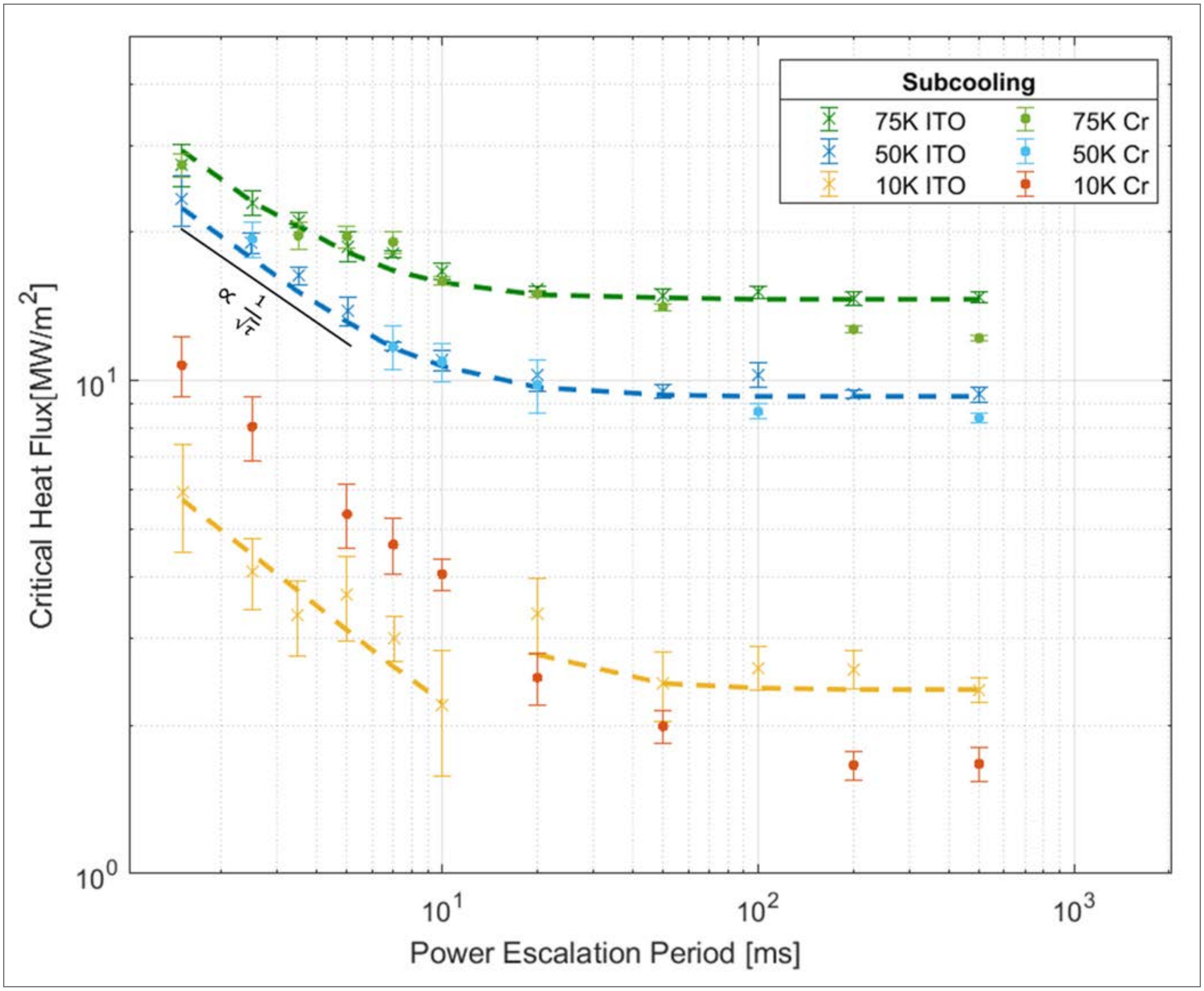



*Figure 3. Transient CHF values measured during exponential power escalations on smooth indium tin oxide surfaces (crosses) and rough chromium coated surfaces (dots) in flow boiling conditions at different subcooling (10, 50, and 75K).*

**Accomplishments**

During the last fiscal year, the team at the UWM was able to get the paper associated with the post-CHF heat transfer and rewet behavior under prototypical PWR conditions published at the journal Applied Thermal Engineering. In it, details about the post-CHF heat transfer and rewet phenomena were discussed based on state-of-the-art temperature measurements (frequency of 100 Hz and spatial resolution of 5 mm) performed using optical fibers. Temperatures up to 844°C were obtained in a Cr-coated zircaloy cladded simulated fuel pin, which showed great resistance to oxidation under departure from nucleate boiling (DNB) conditions for up to 14.1 consecutive seconds dry. Minimum film boiling temperatures were also obtained and provided with this data, which example is shown in Figure 1. Experiments were also performed under atmospheric pressure conditions of a prototypical LOCA scenario, where decay heat effects were included, and the effect of surface roughness was assessed on the quenching speed for cladding temperatures up to 750°C. Optical fibers were placed on the coolant and wall sides, allowing for high timely and spatially resolved measurements (frequency of 100 Hz and spatial resolution of 5 mm). Details of the rewet phenomenon were obtained, indicating the presence of transition boiling and that the reflood water front (at prototypical speeds) had a higher velocity than the quenching one. It was also noticed that the quenching velocity decreased with increasing roughness. Figure 2 shows the temperature profile inside the rod at different times during the quench together with the water level.

The team at MIT has conducted tests of transient CHF in exponential power escalations on a smooth surface and surface mimicking the finish of commercial chromium coated zircaloy claddings, under different flow rate and subcooling conditions (see Figure 3). The results confirm that the CHF limits depend on the period of the exponential power escalation. Notably, CHF increase for rapid transients (i.e., exponential power escalations with a shorter period). Importantly, the value of the CHF in transient conditions is not correlated with the CHF limit in steady-state conditions. The phenomena triggering the boiling crisis in transient conditions seem to be different from the phenomena triggering the boiling crisis under steady heat inputs. The CHF limit in "transient" conditions depends heavily on the flow subcooling and only partially on the surface properties. Precisely, surface conditions affect the CHF limits for low subcooling, when is the first generation of bubbles nucleating on the surface that triggers the boiling crisis. Conversely, in highly subcooled flows, the transient CHF limit does not seem to be affected by the surface conditions, but it is a mere function of subcooling.

Nuclear Energy University Project (NEUP) Award

# Fragmentation and Thermal Energy Transport of Chromia-doped Fuels under Transient Conditions

*Principal Investigator: Heng Ban (University of Pittsburgh [PITT])*
*Team Members/Collaborators: Jie Lian (Rensselaer Polytechnic Institute [RPI]); Yun Long, Liping Cao (Westinghouse Electric Company)*

This project focuses on multiple aspects of experimental testing and engineering-scale modeling in understanding thermal energy transport from high burnup, fractured/ fragmented accident tolerant fuels (ATF), establishing a strong scientific basis to fill a critical knowledge data gap for modeling and simulation of transient fuel performance and safety, such as loss of coolant accident, for future integral testing and fuel licensing.

## Project Description

The objective of this project is to provide essential data, analysis and modeling to fill a major knowledge gap in thermal energy transport relevant to transient fuel performance for high burnup ATFs. Heat transfer from fuel to coolant is governed by: (1) local heat generation rate in high burnup fuels (higher rim area heat generation), (2) thermal conductivity of fuel pellets after significant microstructure changes and fracture/fragmentation (significant radial variation), (3) heat conduction through the fuel-cladding interface and cladding itself, and (4) cladding-to-coolant heat transfer that involves transient boiling. Among these steps, the most critical ones are (2) heat conduction in fractured/fragmented fuel and (4) transient boiling for high burnup fuels because they are most rate limiting and least understood.

In addressing these technical gaps, the project targets a unique aspect of thermal conductivity measurement correlated with high burnup microstructure and fracture characteristics, which can provide a better understanding and prediction of the temperature and fragmentation induced by power transients. The fracture/fragmented fuels with controlled microstructure will serve as a model system to develop thermal property experiments of the bulk and across cracks. The developed protocols can be applied for a wider range of fuels to tackle the challenging and coupled multiple physics of fuel behavior under transient conditions. This project will also provide research methodology including experimental and analysis procedures and tools for future reactor testing, such as the Transient REActor Test Facility (TREAT), of irradiated ATF fuels.

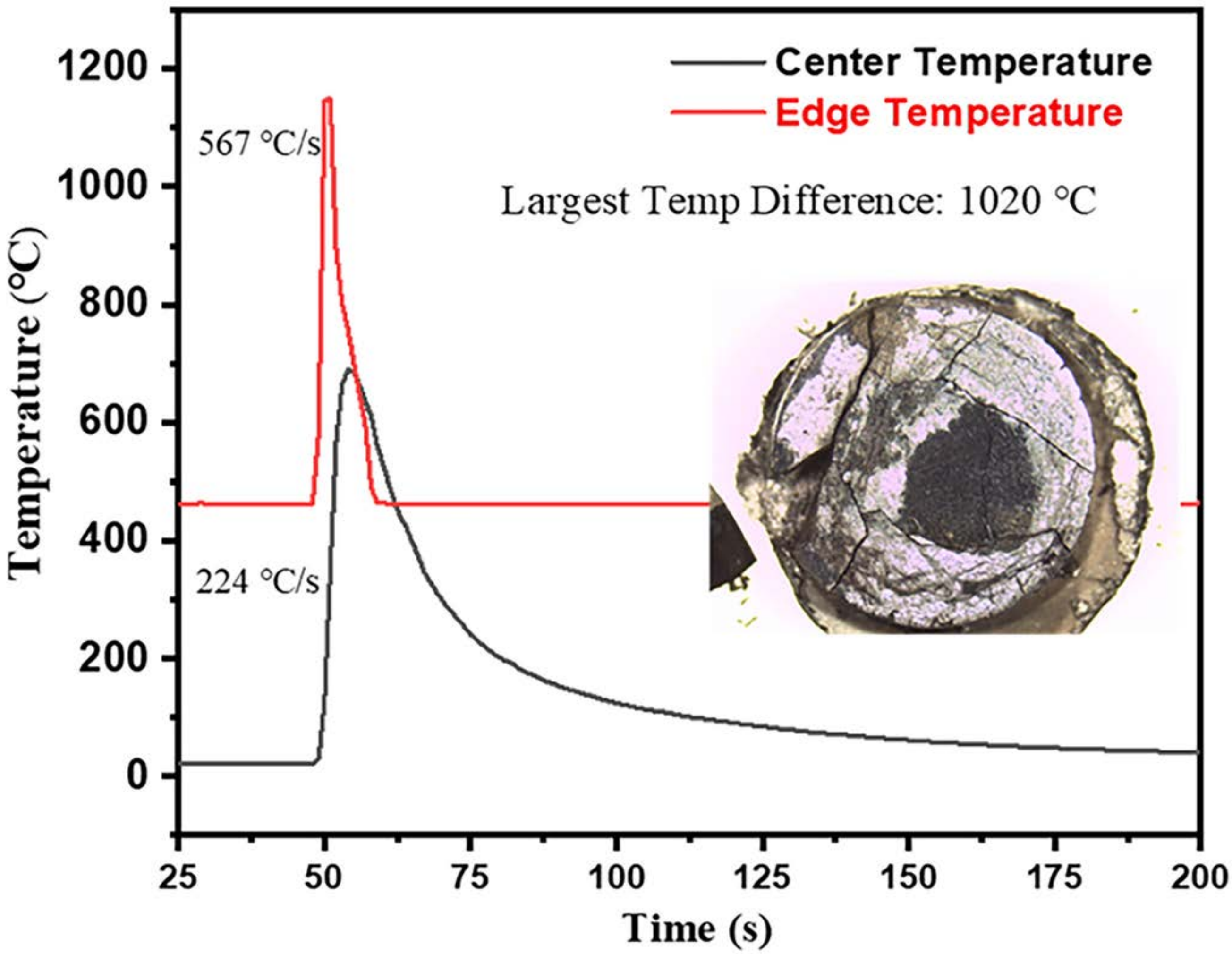



*Figure 1. In-situ temperature profile and fragmentation of $UO_2$ sample under RIA thermal shock test.*

## Accomplishments

Synthesis of high burnup structure $UO_2$ with submicron grain size, micron pore size and controlled porosity were demonstrated through an innovative volume control sintering process by Spark Plasma Sintering (SPS). Chromia doping was also incorporated into the engineered fuels mimicking high burnup structure. The thermomechanical properties of the engineered fuels were characterized and correlated with microstructure features such as grain structure, porosity, pore size, etc., establishing scientific understanding of the microstructure-dependent fuel properties. Porosity has a significant influence on its hardness, fracture toughness and thermal conductivity, and $Cr_2O_3$ doping positively influenced the Young's modulus and fracture toughness.

A unique capability of using SPS has been developed under the support of this project for ultrafast thermal shock testing of nuclear fuels to evaluate their transient behaviors. The ramping rate for transient testing could be controlled from 5-10 °C/s to 700 °C/s to mimic temperature ramping of the loss-of-coolant-accident (LOCA) and reactivity-induced-accident (RIA) events (Figure 1). The unique capability with different experimental designs and controlled temperature

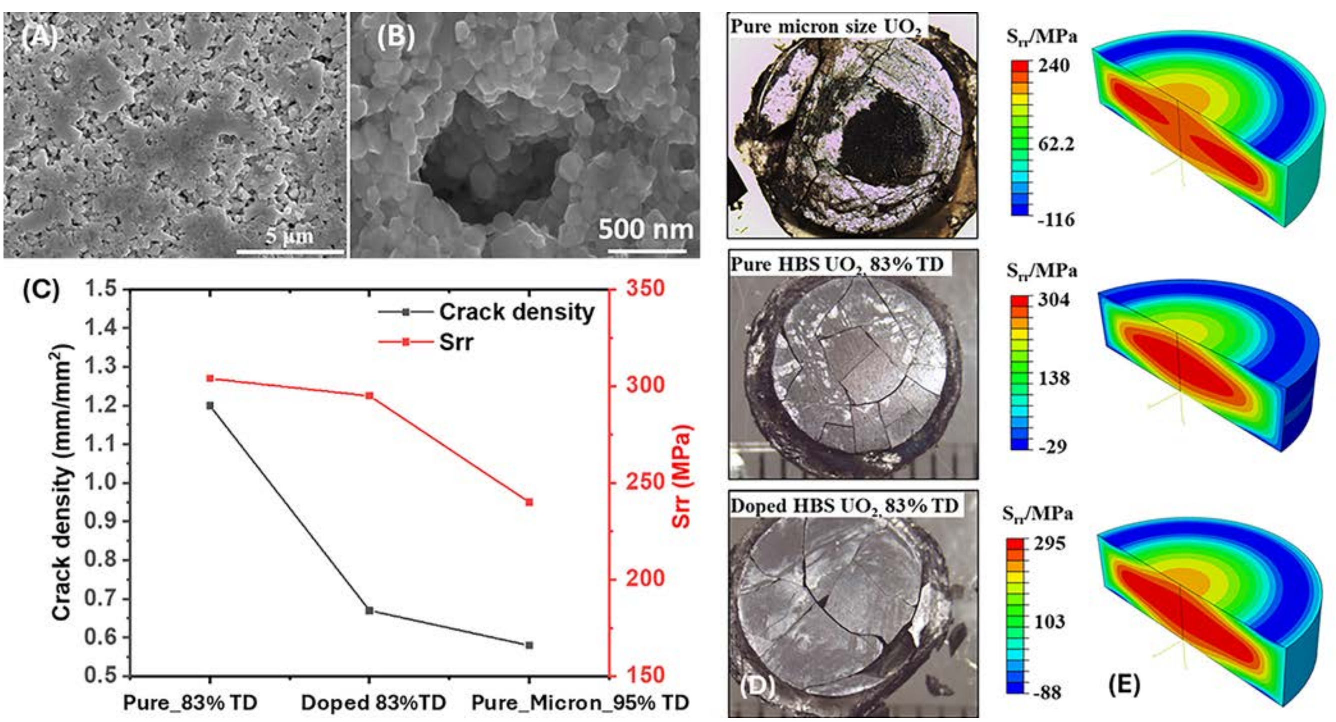



*Figure 2. Fragmentation of $UO_2$ with controlled grain size and porosity under RIA transient testing (ramping rate: 500 °C/s and terminal temperature: 1300 °C): (A-B) porous nano-sized $UO_2$ fuels mimicking HBS; (C) Correlations of the crack density as measured from microstructure evolution post-transient testing (D) with tensile stress in radial direction of the fuel pellets as calculated by finite element analysis (E).*

***Integrated research synergizing innovative transient testing of nuclear fuels, microstructure control and doping, meso-scale finite element analysis and engineering fuel performance modeling, enabling a science-based understanding of their transient behaviors under simulated LOCA events, essential for quantifying fuel forms for future nuclear energy systems.***

ramping enables a cost-effective approach for rapid screening and high throughput evaluation of the transient behavior of nuclear fuels, in complementary to more advanced testing using TREAT reactors for future development of nuclear fuels.

The crack and fragmentation behavior of $UO_2$ pellets with controlled grain structure and $Cr_2O_3$ doping were tested with rapid power ramping (5 to 15 ˚C/s up to 500 ˚C/s) mimicking prototypical LOCA and RIA heating profiles (Figure 2). Dense micron-sized $UO_2$ pellets display well-maintained integrity without cracking with the ramping up to 8 ˚C/s to 1500 ˚C. Dense nano-sized $UO_2$ pellets fracture in both fresh undoped and Cr2O3-doped pellets. The $Cr_2O_3$ doped oxide fuel with a larger grain size (~22.2 μm) displays the best transient performance under LOCA testing due to its better thermal conductivity under high temperature.

Combined with in-situ temperature profile of samples during the thermal shock tests, finite element analysis (FEA) has been used to probe temperature profile and residual stress distribution to provide an insightful view of the transient behavior of $UO_2$. FEA calculations suggest a temperature gradient across the fuel pellet during transient testing, resulting in residual stress and thus pellet cracking, which can be correlated with their thermal-mechanical properties.

Porosity and grain size significantly impact pulverization and fragmentation behavior of $UO_2$. Furthermore, the $Cr_2O_3$ doped high burnup structure $UO_2$ sample shows better resistance against fracture and fragmentation than pure sample as evidenced by lower peak temperature and crack density under power transient tests (Figure 2).

Nuclear Energy University Project (NEUP) Award

# Estimation of Low Temperature Cladding Failures During a Reactivity Initiated Accident Transient

*Principal Investigator: Arthur Motta (Pennsylvania State University [PSU])*
*Team Members/Collaborators: Luiz Aldeia Machado, Katheren Bosson Nantes, Elia Merzari, Mia Jin (All PSU); Lise Charlot, (Idaho National Laboratory [INL])*

***This project has created the capabilities for the fuel performance code BISON to assess the effect of hydride localization in promoting cladding failure during an RIA.***

The Reactivity Initiated Accident (RIA) in light water reactors is a postulated design basis accident which can occur when control rods are ejected or dropped from the core leading to a local increase in fission rate. Two mechanisms can cause fuel failure – pellet cladding mechanical interaction (PCMI) or clad ballooning. If the cladding temperature rises fast, the cladding fails at high temperature by ballooning. We study low temperature PCMI failures which can cause fuel ejection and channel blockage failures. PCMI failure is controlled by several parameters, including total energy deposition and the cladding state after reactor exposure. Although the hydrogen in the cladding can redistribute under concentration, temperature and stress gradients present during operation, current models to evaluate cladding failure during an RIA in the fuel performance code BISON are based on the average hydride concentration rather than the local concentration. We aim to provide this capability in BISON.

### Project Description

Utilities operating nuclear power plants must show that such a potential RIA accident would not entail severe consequences and radioactivity release. This project aims to evaluate the likelihood of low-temperature failures during an RIA and the key parameters that may lead to such failures for near-term accident tolerant fuel concepts, including coated cladding, high burnup fuel, and higher enrichment pellets. We perform this evaluation using coupled multiphysics simulations (materials, thermal hydraulics and reactor physics). This includes developing a new criterion for localized failure induced by hydrides, establishing a clear demarcation

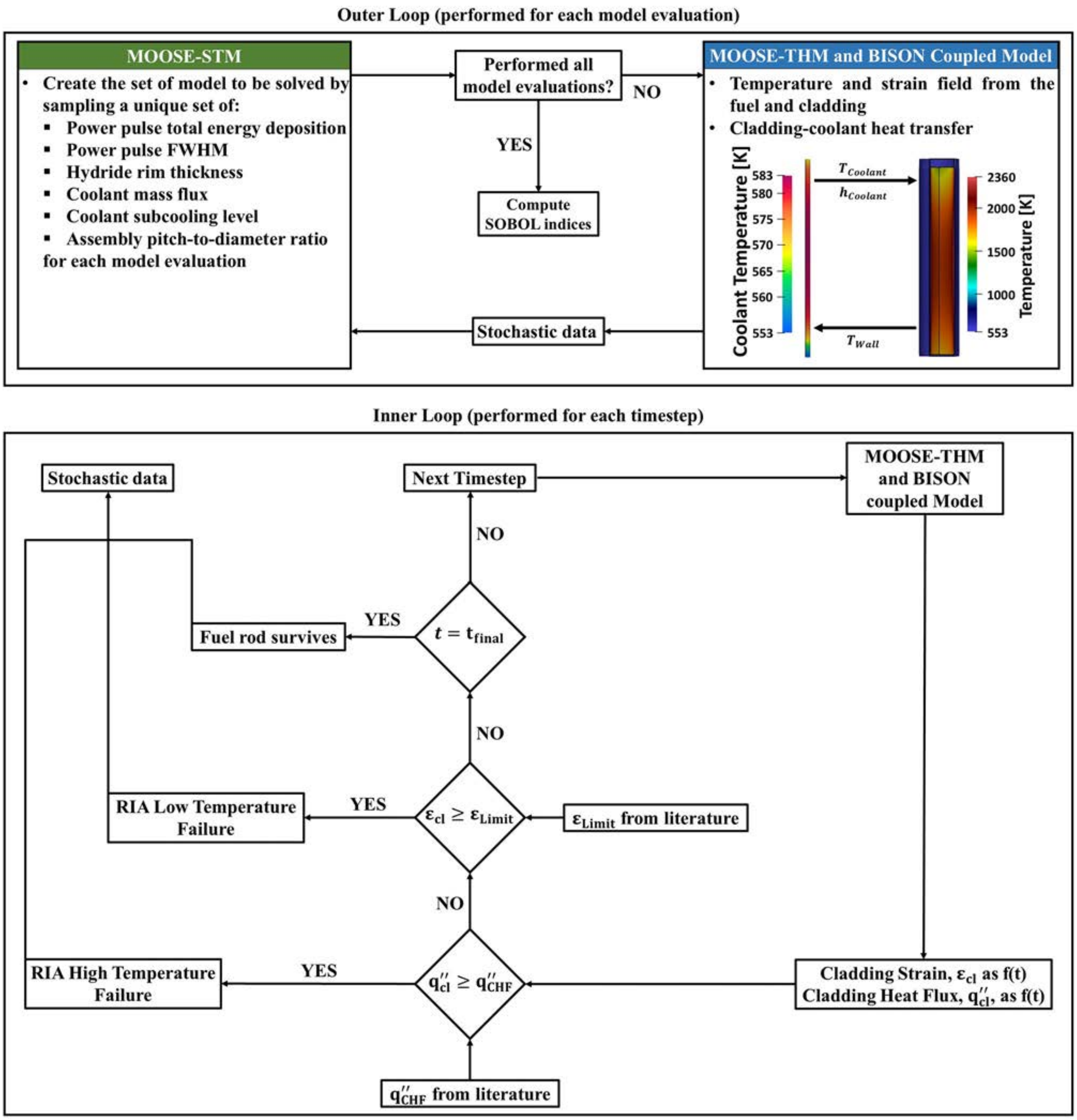



*Figure 1. STM-THM-BISON coupled model workflow.*

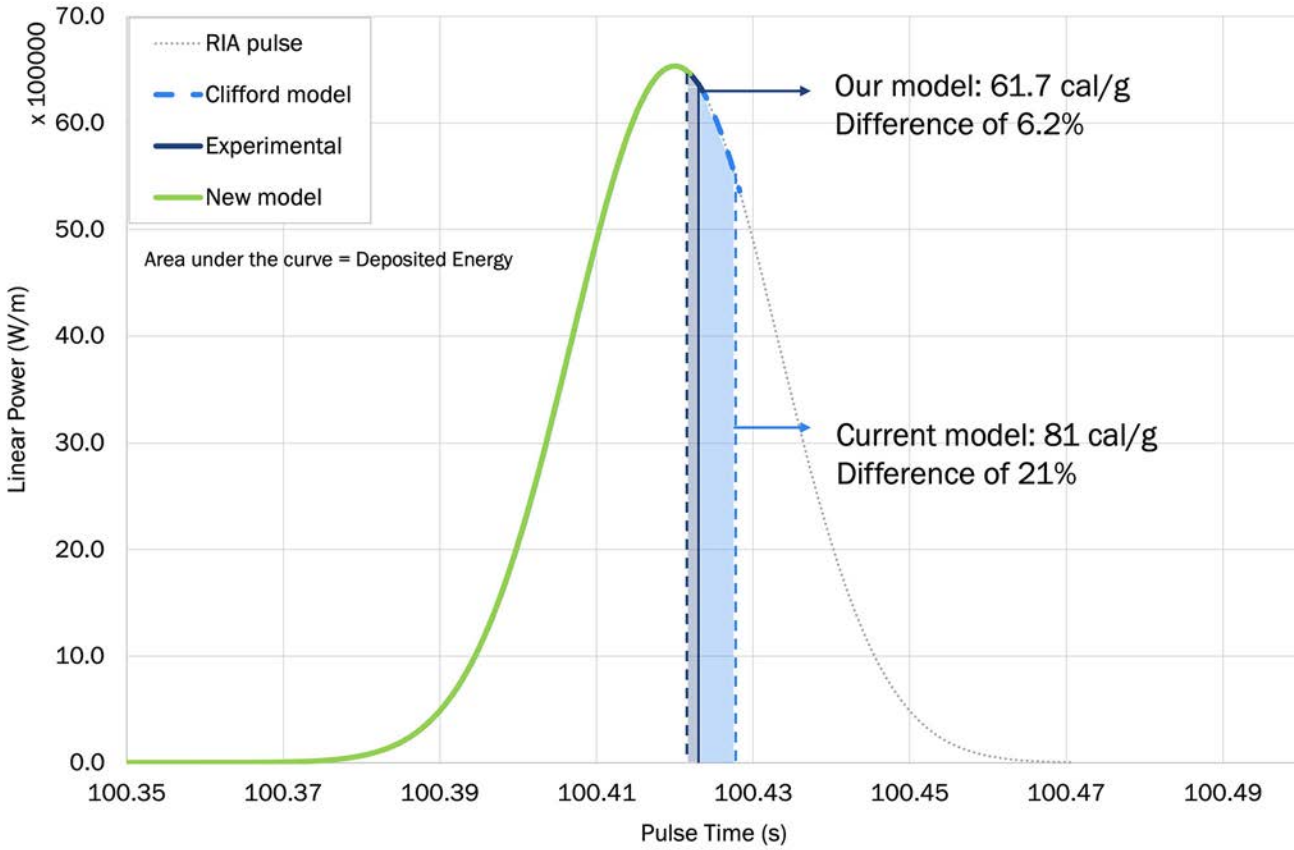



*Figure 2. Comparison of each model with experimental reported deposited energy at power peak node for CABRI REP-Na10 test modeling.*

of the limit between low and high temperature failure using the Thermal Hydraulics Module in BISON (THM) and benchmarking through a full BISON simulation of the CABRI reactor Rep-Na experiments. The achievements of the project are:

- Increasing the capabilities of the fuel performance code BISON to more precisely assess the probability of low temperature PCMI failures based on inhomogeneous hydride distribution. This includes the development of a new failure criterion to be inserted into BISON, based on experiments conducted at temperature and with the correct stress state of plane strain tension.
- Simulating benchmark experiments both during normal operation and during an RIA transient by performing coupled thermal hydraulics calculations using THM, reactor physics and materials behavior to organically discern the limit between low temperature and high temperature failures.
- Benchmarking these efforts against the known experimental information from the integral tests performed in the CABRI program in France and the ongoing High-burnup Experiments in Reactivity Initiated Accidents experiments at the Transient REActor Test Facility (TREAT) at INL.

**Accomplishments**

To date we have modeled the effect of the development of a hydride rim in the sample during operation on the PCMI failure probability, the effect of a spalled oxide layer and the concentration of hydrogen in the interpellet region. Figure 1 shows the method by which the calculation is performed by coupling the MOOSE-Based Scalar Field Transport Modeling (STM) with THM and BISON. Figure 2 shows the degree of accumulation of hydrogen in the interpellet region during service, because of the lower temperature in that region. Hydrogen is quite mobile in Zr at reactor temperatures and can concentrate quickly on the cold zones in response to temperature gradients present in the cladding. Those local concentrations will cause hydride accumulations that will fragilize the cladding. . The concentration of hydrogen in typical cases and its effect on ductility was evaluated using a new failure model based on local hydrogen concentration to predict PCMI failure. This model is being applied to the experiments conducted at the CABRI reactor. Figure 3 shows the results of the simulation of the RIA in the experiment Rep-Na3, where the time to failure calculated by this model is quite a bit closer than the previous criterion proposed by Clifford[1]. Figure 4 shows the result

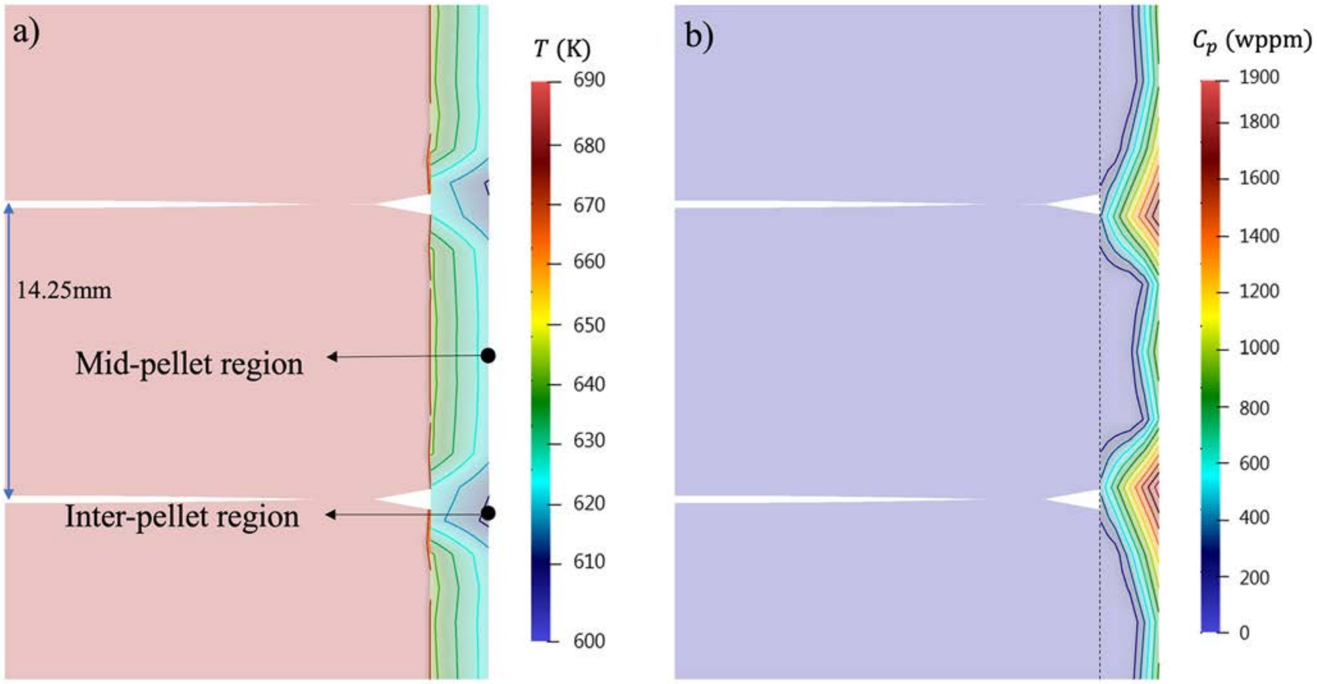



*Figure 3. Contour plots for (a) temperature distribution and (b) hydride concentration at the rodlet center height, at 0.29 m from the bottom of the pin. Fuel average burnup of 54 GWd/tU and average concentration of hydrogen in hydrides 474 wppm. (Figures scaled 5x)*

of a sensitivity study of the various parameters of the RIA on a low temperature failure of high temperature failure when the critical heat flux (CHF) occurs. It is clear from the study that the most important parameters are the hydride rim thickness, energy deposition and the pulse width, whereas the other parameters (inlet temperature, mass flow rate, and channel geometry) are comparatively less important.

PCMI failures however can only happen if the cladding temperature rise is slow enough so that the hydrides formed during operation do not dissolve durring the transient. Clearly, when the CHF is reached during the RIA the cladding will be ductile enough to resist PCMI failure, so this is a clear limit between the two failure modes.

Through the collaboration between PSU and INL, this project also contributed to the MOOSE-framework development, especially by the implementation of new thermal-hydraulic correlations into MOOSE-THM, which are already available to other users, and the future implementation of a new PCMI failure criteria into BISON.

### References

[1.] Clifford P. (2020) Pressurized-water reactor control rod ejection and boiling-water reactor control rod drop accidents, Tech. Rep. RG 1.236, United States Nuclear Regulatory Commission.

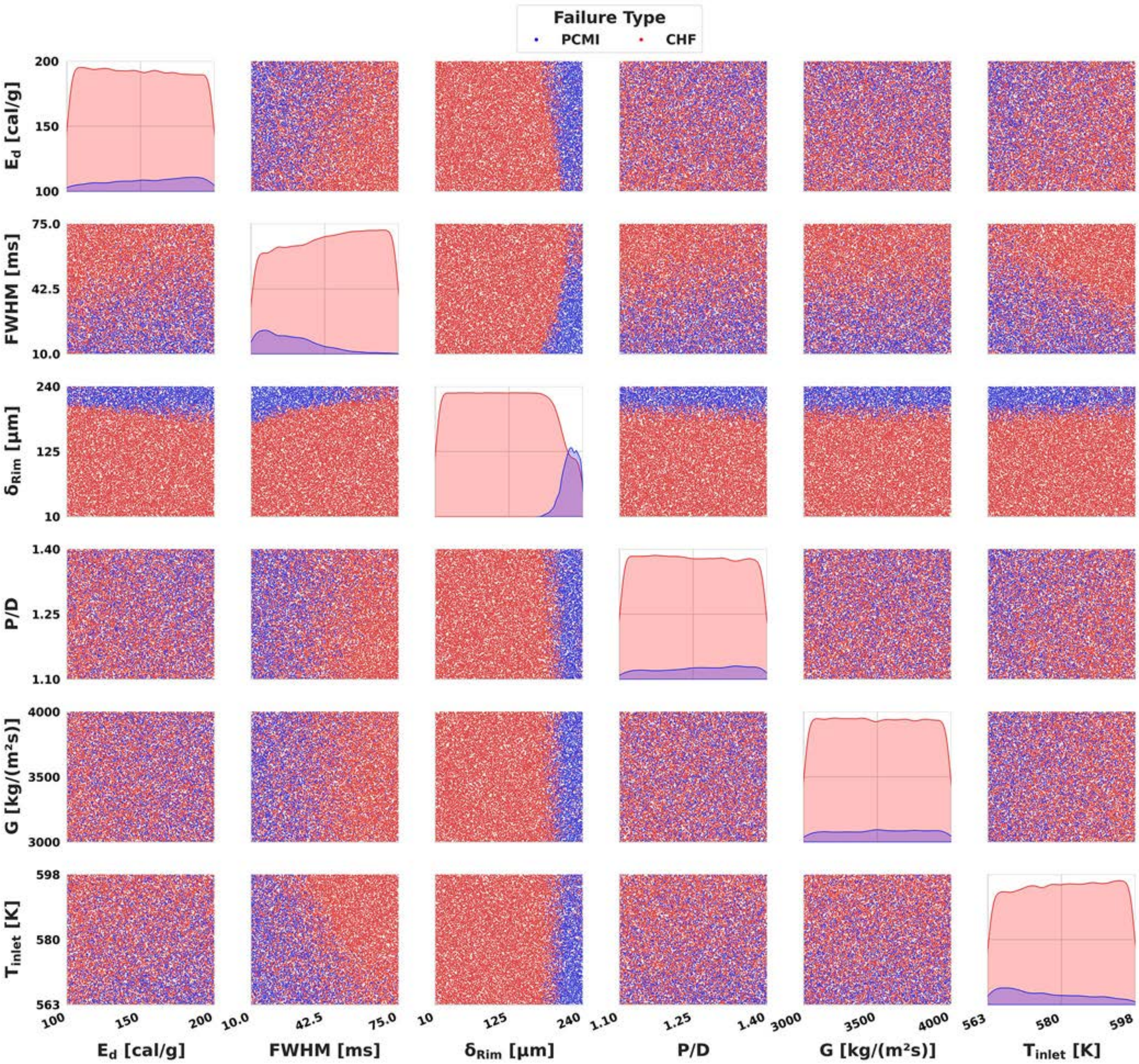



*Figure 4. Pairwise Scatter Plot showing the uncertainties value distribution, and the failure type associated with these values.*

Nuclear Energy University Project (NEUP) Award

# Safety Implications of High Burnup Fuel for a 2-Year Pressurized Water Reactors Fuel Cycle

*Principal Investigator: Nicholas R. Brown (University of Tennessee, Knoxville [UTK])*
*Team Members/Collaborators: Isabelle O. Lindsay, Mason Fox (All UTK); Koroush Shirvan, Assil Halimi (Massachusetts Institute of Technology [MIT])*

Pressurized Water Reactors (PWRs) currently achieve burnup levels of about 50-60 GWd/MTU. However, it would be economically advantageous for utility providers to be able to extend the life of the fuel to longer cycles and higher burnup via increased enrichment. The objective of this project was to perform safety analysis of high burnup fuel for a Westinghouse PWR. The safety analysis encompassed normal operation and selected anticipated operation occurrences (AOOs) and design basis accidents (DBAs). Our team (see Figure 1) leveraged significant experience in both fuel performance modeling and core design modeling. We identified potential opportunities and gaps for high burnup fuel by utilizing both well-established and modern methodologies to model fuel performance, reactor physics, thermal-hydraulics, and plant system-level response.

## Project Description

A notable challenge of increasing burnup levels is the resulting additional excess reactivity encountered during the early stages of fuel life. To accommodate, burnable absorbers beyond soluble boron are introduced into the fuel system. In high burnup fuels, the possibilities of cladding lift-off and fuel melting increase due, in part, to increased rod internal pressures and limited fuel thermal conductivity, respectively. This research program focused on answering key questions for high burnup traditional and near-term Accident Tolerant Fuel (ATF) fuel focused on normal operation, AOOs, and DBAs, including the following examples:

- What is the impact of high burnup fuel and near-term ATF on AOOs in a Vogtle-like plant?
- How do high burnup (>60 GWd/t rod average burnup) and near-term ATF core designs of interest to Southern Nuclear and Westinghouse impact fuel rod integrity and possible failure limits?
- For AOO (control rod withdrawal (CRW)) and selected DBA (control rod ejection) of interest to Southern Nuclear, how would high burnup fuel and near-term ATF impact the potential for pellet cladding mechanical interaction failure?


*Figure 1. Research team members from University of Tennessee and Massachusetts Institute of Technology.*

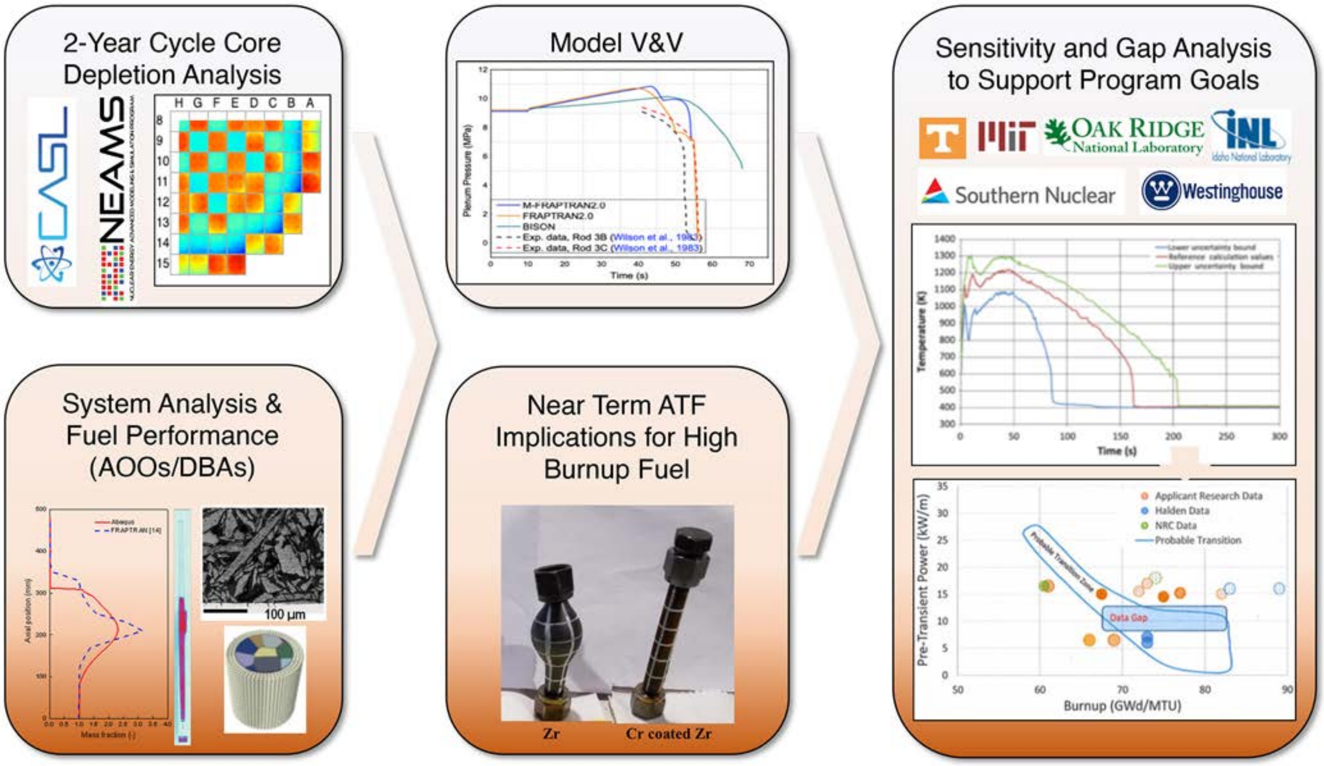



*Figure 2. Overarching project objectives and outcomes.*

***The project results indicated that the proposed fuel design with homogenously blended gadolinium operates with greater safety margins regarding plenum pressure and hoop strain limits versus the Integral Fuel Burnable Absorber core design.***

The focus of the project was on high burnup core designs of interest to Southern Nuclear and Westinghouse (~70 GWd/t rod average burnup) which also utilize higher enrichment (up to ~7%). The over-arching vision for the project is shown in Figure 2.

## Accomplishments

The analysis performed in the project [1, 2] compared two core designs, one by MIT [3] and one by Southern Nuclear. The results demonstrate the MIT annular fuel design with homogenously blended gadolinium (Gd) as a burnable absorber operates with greater safety margins during normal operation, allowing for additional operational flexibility. During normal operation, the Southern Nuclear core design utilizing Integral Fuel Burnable Absorber pins contained fuel pins which reached plenum pressures above 15.5 MPa by the end of the first fuel cycle and fuel pins experienced cladding hoop strains above 1%. In the Gd core design by MIT, only two analyzed pins experienced plenum pressures above 15.5 MPa and no pins exceeded 1% cladding hoop strain. During the control rod withdrawal scenario, plenum pressures for pins in both designs marginally exceeded system pressure, however neither experienced excessive hoop strain. The Gd core design from MIT experienced a maximum fuel temperature of 2418 K, which is significantly higher than the Integral Fuel Burnable Absorber design from Southern Nuclear at 2157 K, but still within regulatory guidance. We predicted that the fuel in both could return to service after the CRW event. We also predicted that cladding would not fail during the Control Rod Ejection in either core design. Generally, the Integral Fuel Burnable Absorber core design from Southern Nuclear performed with greater safety margin with regards to temperature during normal operation and the transient events. However, the Gd core design from MIT performed with greater safety margin regarding plenum pressure and hoop strain limits during normal operation and both transient events.

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Nuclear Energy University Project (NEUP) Award

# Modeling High-Burnup Light Water Reactor Fuel Behavior Under Normal Operating and Transient Conditions

*Principal Investigator: Brian D. Wirth (University of Tennessee, Knoxville [UTK])*
*Team Members/Collaborators: Oliver Baldwin, W. Cade Brinkley, Charles Lieou (All UTK); Suresh Yagnik (Electric Power Research Institute [EPRI]); Nathan Capps (Oak Ridge National Laboratory [ORNL])*

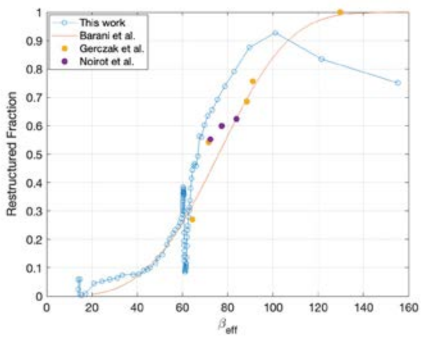



*Figure 1. Comparison of predicted HBS fraction from our microstructurally informed model (this work in blue) with a literature model from Barani and Pastore (red line) and experimental data obtained from Gerczak et al. and Noirot et al. (data points).*

***Improved understanding and predictive ability of fuel fragmentation during loss-of-coolant accidents will enable definition of the maximum fuel burnup in the commercial nuclear industry, paving the way for extended cycle lengths.***

This project aims to develop a high-burnup light water reactor nuclear fuel modeling capability to implement in the US Department of Energy fuel performance code BISON, with a high impact in terms of informing the safety case and supporting the extension of burnup limits currently pursued by the US nuclear industry. The developed capability will enable the accurate simulation and improved understanding of high-burnup fuel rod behavior during normal reactor operation and accidental transients, including estimation of the quantity of fuel susceptible to fragmentation and dispersal, fission gas release (FGR) and rod inner pressure, to ultimately identify the rod life-limiting factors and define the boundaries of safe operation.

## Project Description

In broad terms, two main facets of fuel physical behavior at high burnup form the basis to achieve a fuel behavior representation during normal operating conditions and transients, i.e., (a) fuel fine fragmentation and (b) the evolution of the high burnup structure (HBS). Accordingly, the logical path that we envision for the proposed work starts with the development of two main capabilities to model fragmentation of unrestructured fuel and the evolution of the HBS. The fragmentation model will represent a first component to model transient behavior and will be integrated with a capability for transient FGR; the HBS model will cover high-burnup fuel behavior under normal reactor operation, including potential fuel rod life-limiting factors such as gaseous fuel swelling due to HBS gas pores, and constitute the basis for modeling transient fragmentation and FGR in the HBS, which will be tackled immediately afterwards. The result will be a comprehensive model for high-burnup fuel behavior under normal operating and transient conditions, covering both unrestructured and HBS fuel. The developed capability will be implemented in BISON and applied in the simulation of integral high-burnup fuel rod experiments for validation.

In more detail, the modeling strategy that we envision for the proposed project is as follows:

- We will develop a model for HBS formation that is microstructurally informed and predicts porosity, grain size and grain boundary character evolution as a function of fuel burnup and temperature.
- To model fuel fragmentation driven by fission gas bubble pressure at grain boundaries, we will develop a modeling criterion that incorporates fission gas bubble density and pressure on grain-boundary rupture.

## Accomplishments

UTK has successfully developed and validated a model for the high burnup restructured fraction of nuclear fuel, which is informed by microstructural data, as shown in Figure 1. The blue circles and blue line in Figure 1 are the results of our new high burnup fraction data and plot the predicted restructured fuel fraction as a function of effective burnup, $b_{eff}$. Our prediction generally matches the sigmoidal shape of a model developed by Barani and Pastore, except at very large effective burnup where limited data on microstructural evolution of the fuel is available to inform the model. Further, the use of an effective burnup parameter has resulted in the collapse of the central restructured region of the fuel into a sharp peak at 58 gigawatt-day per ton of Uranium (GWD/tU), followed by an equally steep drop at 61 GWD/ tU. Between 60 and 80 GWD/tU, our model prediction is in good agreement with the overall trend of the experimental data.

As well, UTK, in partnership with our collaborators at Idaho National Laboratory, Los Alamos National Laboratory and ORNL have developed an improved representation of microcracking in ceramic nuclear fuels and corresponding transient fission gas release (FGR). Figure 2 shows the results of predicted temperature and stress in the fuel (Figure 2a) as well as the predicted grain size, microcrack and areal fraction (Figure 2b) from a newly developed model that provides an improved description of microcracking and fission gas bubbles in ceramic nuclear fuels, using internal state variables that represent the statistical densities of microcracks, GBs, and grain-face bubbles. Figure 2 shows fuel behavior following a base irradiation (red lines) and after a 48 hour transient test associated with a test fuel rod segment from the fuel test RISO-AN3. Our new model predicts comparable to improved fission gas release compared to the original simplified fission gas release (SIFGRS) model in BISON and reveals that FGR is dominated by restructuring and grain refinement throughout the fuel pellet. Our analysis of the results raises fundamental questions about the nature and extent of fuel restructuring in the pellet interior, as well as the possibility and prevalence of intra-granular cracking.

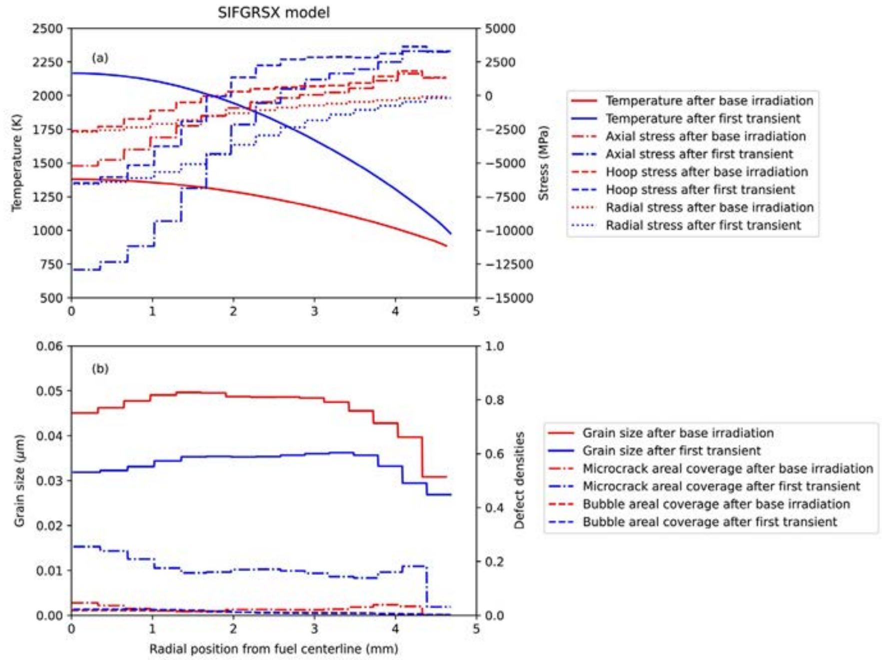



*Figure 2. Model results predicting the variation of a) temperature and axial, hoop and radial stress, and b) grain size, microcrack density and bubble areal coverage after base irradiation (red) and immediately after a temperature increasing transient (blue).*

**Nuclear Energy University Project (NEUP) Award**

# Characterizing Fuel Response and Quantifying Coolable Geometry of High-Burnup Fuel

*Principal Investigator: Wade Marcum (Oregon State University [OSU])*
*Team Members/Collaborators: Trevor Howard, Guillaume Mignot, Sade Campos (All OSU);, Shanbin Shi, Avinash Moharana (All Rensselaer Polytechnic Institute [RPI]); Douglas Crawford, Ben Spencer, Kyle Gamble (All Idaho National Laboratory [INL])*

As high burn-up fuel is examined as a means of improving the economics of a reactor, evaluating the safety implications of these fuels normal and accident conditions, such as Loss of Coolant Accidents (LOCAs), is essential. Of utmost importance is the characterization of fuel fragmentation, relocation and dispersion – primarily dispersion – and the impact on long term core coolability. Using experimental and numerical methods, the project assesses means of simulating the dispersion event and experimentally quantifying dispersion while aligning with the Department of Energy's (DOE) Advanced Fuels Campaign (AFC) and other related programs.

## Project Description

Ultimately, the main objective of this research is to evaluate the safety implications of high-burnup fuel in light water reactors (LWRs) with an emphasis on pressurized water reactors (PWRs) and applicability to boiling water reactors (BWRs) as appropriate. In doing so, the project provides insights into whether high-burnup fuel compromises coolable geometry and long-term cooling

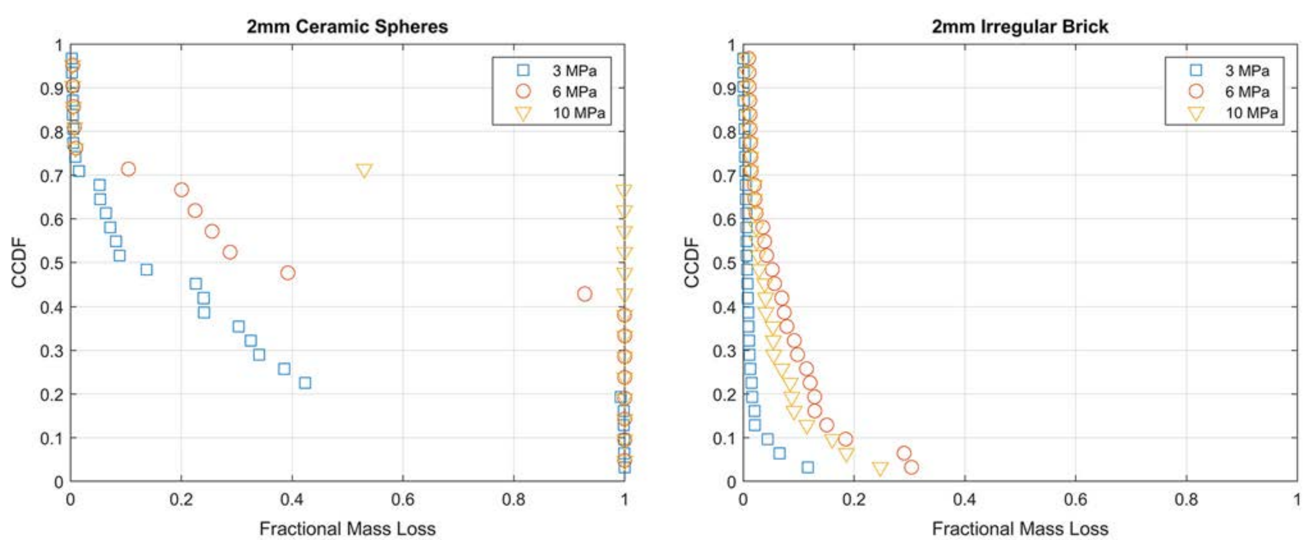



*Figure 1. Fractional mass loss compared to the continuous cumulative distribution function for the representative geometry for similar density materials with spherical (left) and irregular (right).*

during normal operations and accident conditions, such as LOCAs. The study provides empirical data to support the verification, validation, and development of simulation using state-of-the-art tools to support the feasibility of extending fuel burnup while ensuring compliance with regulatory safety criteria. Upon completion, this research will have significantly advanced the state of knowledge by providing a comprehensive understanding of dispersion behavior at varying burnup levels (i.e., particle size distributions). It will offer objective assessments of fuel dispersion impacts, contributing to more accurate safety analyses. This in turn improves economics as well as enhanced safety due to the removal of assumptions made around the accident event. Doing so enables the nuclear industry to optimize fuel usage, improve reactor economics, and extend the operational life of the current reactor fleet without compromising safety.

This research aligns with DOE's objectives by enhancing the safe, reliable, and economic operation of the nation's current and next-generation reactors. By leveraging advanced experimental infrastructure and computational tools, the project supports DOE's AFC and the Light Water Reactor Sustainability Program. The development of a validated solids transport model provides predictive capabilities for fuel behavior under accident conditions, informing safety cases and regulatory decisions. This research will help the DOE achieve its goals of maintaining the safety and reliability of the existing reactor fleet while facilitating the transition to advanced reactor technologies. It also contributes to the broader objective of ensuring a sustainable and economically viable nuclear energy future.

## Accomplishments

The technical goals of the research project titled "Characterizing Fuel Response and Quantifying Coolable Geometry of High-Burnup Fuel" are to evaluate the safety implications of high-burnup fuel in LWRs and determine whether high-burnup fuel compromises coolable geometry and long-term cooling during

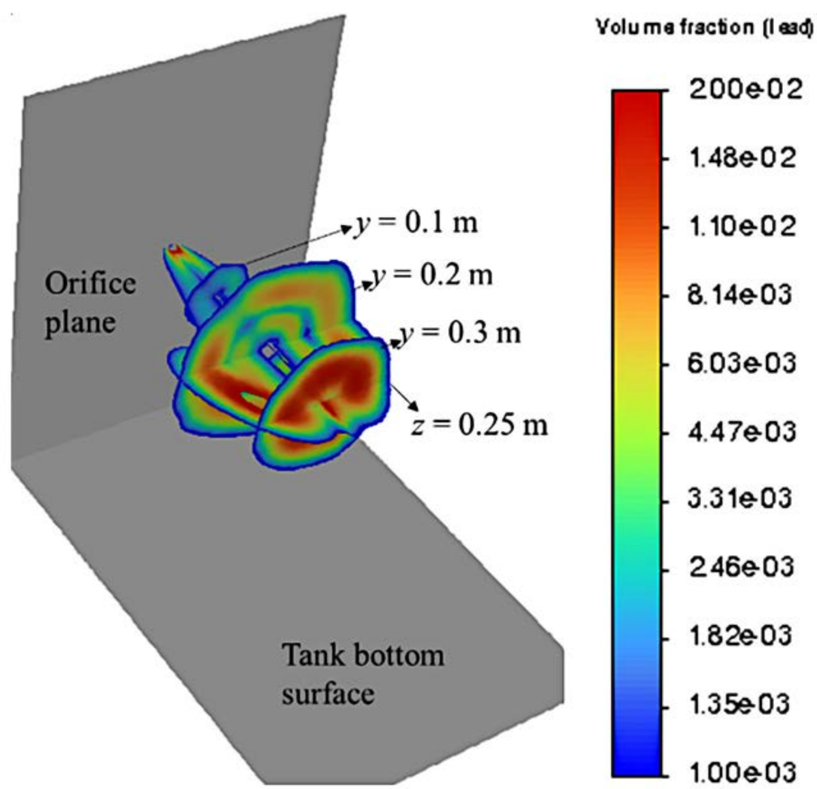



*Figure 2. Volume fraction of solid particles being dispersed under a simulated case.*

*This project aims to enhance the safety, reliability, and economic efficiency of nuclear reactors by extending the operational life of high-burnup fuel while ensuring compliance with regulatory safety criteria and highlights the need for improved understanding and interpretation of dispersion results.*

normal operations and accident conditions, such as LOCAs. The project provides experimental data, empirical relationships, and numerical data to support the feasibility of extending fuel burnup while ensuring compliance with regulatory safety criteria. As of September 1, 2024, significant progress has been made toward these technical goals. The project is organized into three main tasks: experimental testing, modeling development, and benchmarking and analysis.

### Task 1: Experimental Testing

The experimental campaign has successfully observed and characterized critical impacts of fuel dispersion during fuel failure on coolable geometry and long-term cooling. The initial proposed study presumed a known distribution of particles that would be distributed upon a grid and that such a distribution would be reasonably represented by a normal distribution indicating that dispersion into a fluid medium was of the utmost importance. Upon initial review of previous studies, and a collection of more data, it became apparent that a notable locking of the particles would result in little to no fuel loss even under ideal conditions. The consequences of this were that fuel locking, and a lack of understanding thereof would have two effects: (1) without appropriate knowledge of the particulate distribution (i.e., a non-normal distribution), conclusions drawn from other experimental results using a surrogate mater may be inappropriate without understanding and (2) locking may create a physical system in which the dispersion quantities are minimal and thus in-consequential. Using this knowledge, more than 1000 tests were conducted in air to better characterize the expected dispersion. A subset of results is presented below.

The test results produced the following major conclusions:

1. Particle diameter relative to the geometric area is the most important parameter in determining the amount of dispersed fuel.
2. The coefficient of restitution (CoR) drives the particles' ability to spread apart during the dispersion event which limits the locking. The CoR increases with yield strength, decreases with velocity, density and modulus of elasticity, and increases with Poisson's ratio. This has several significant impacts:
   - spherical shapes are highly conservative as irregular shapes result in a CoR of near 0.
   - metals (with some exceptions, e.g., lead) have higher CoR's, and this effect is likely heavily mitigated by irregular particle shape.
   - Rod internal pressure at the inception of fuel failure causing dispersion is somewhat self-regulating as increases pressure increases the energy of the system and separation of the particles, causes an increase in particle velocity.

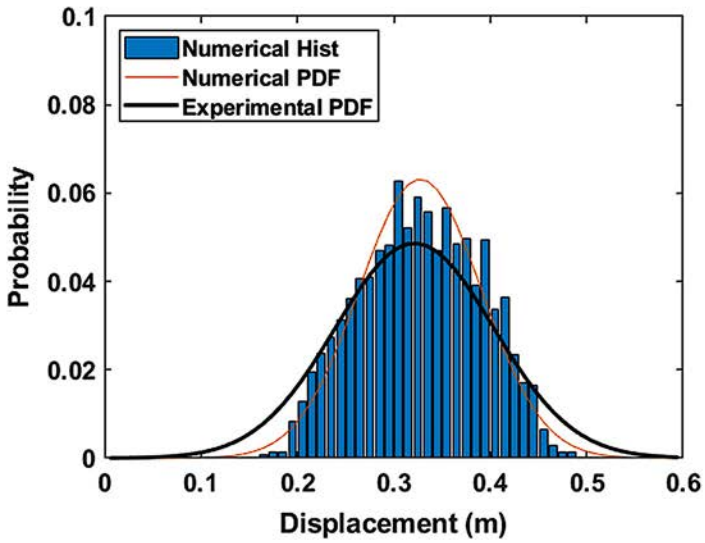



*Figure 3. Comparison of numerical and experimental probability distribution functions during a test dispersion event at 8 MPa plenum pressure (right) and experiment images (left) using lead pellet surrogate in water.*

3. The statistical distribution of the particles tended to be exponential when locking dominated, normal when locking wasn't present, and was reasonably represented by a combination of both distributions.

### Task 2: Modeling Development

Under the model development, a framework was developed to address the continuum treatment of fragmented particles under a three fluid medium. The framework leverages a continuous distribution of the particles – the simulation can be resolved entirely within this modeling approach. This framework was then used to run a suite of simulation cases, which are continuing to be run. A sample set of results is shown in the next figure.

### Task 3: Benchmarking and Analysis

In addition to the results presented under Tasks 1 and 2, additional experiments and simulations were conducted for the purposes of benchmarking and analysis of the simulation framework. Overall, the results are showing good agreement between both the simulation and the experimental values. An important outcome of this comparison is that the probabilistic distribution of the mass loss has very limited impact on the probabilistic distribution of the particles.

## The Use of Dual Dopants as Boiling Water Reactor Burnable Absorber

*Principal Investigator: Arantxa Cuadra (Brookhaven National Laboratory [BNL])*
*Team Members/Collaborators: Cihang Lu (BNL)*

The Boiling Water Reactor (BWR) is a key contributor to low-carbon electricity generation worldwide. Burnable absorbers (BAs) are critical for optimizing reactor performance and safety by reducing excess reactivity at the beginning of cycle through neutron absorption. This process prevents rapid power excursions and minimizes thermal stress on the reactor core. Additionally, BAs promote uniform power distribution, reducing the risk of localized overheating and material degradation. They also enhance fuel utilization by maintaining steady reactivity, enabling higher burnup (HBU) and extending fuel cycles, which lowers operational costs. While single BA materials like gadolinium oxide ($Gd_2O_3$) and boron carbide ($B_4C$) have been well studied, the combined use of dual BAs potentially offers additional advantages. This study explores the impact of dual BA materials as fuel dopants on the performance and safety of BWRs.

### Project Description

The primary technical objective of this research is to evaluate the impact of dual BA materials, used as fuel dopants, on the performance and safety of BWRs, specifically those utilizing HBU / high enrichment fuels. This includes analyzing how these dual BAs influence excess reactivity and fuel utilization throughout the reactor's fuel cycle. By exploring the synergistic effects of combining two BA materials, the research aims to uncover potential advantages over the use of single BAs, such as lower excess reactivities (enhanced controllability), reactivity coefficients (inherent safety), and extended fuel burnup (BU) capabilities. If successful, this research will advance the state of knowledge by providing a deeper understanding of dual BA strategies, potentially leading to more efficient and safer reactor designs.

***This project is important because it seeks to optimize burnable absorber strategies in BWRs with HBU fuels, potentially enhancing reactor safety, efficiency, and cost-effectiveness, which are crucial for advancing both current and future nuclear power technologies.***

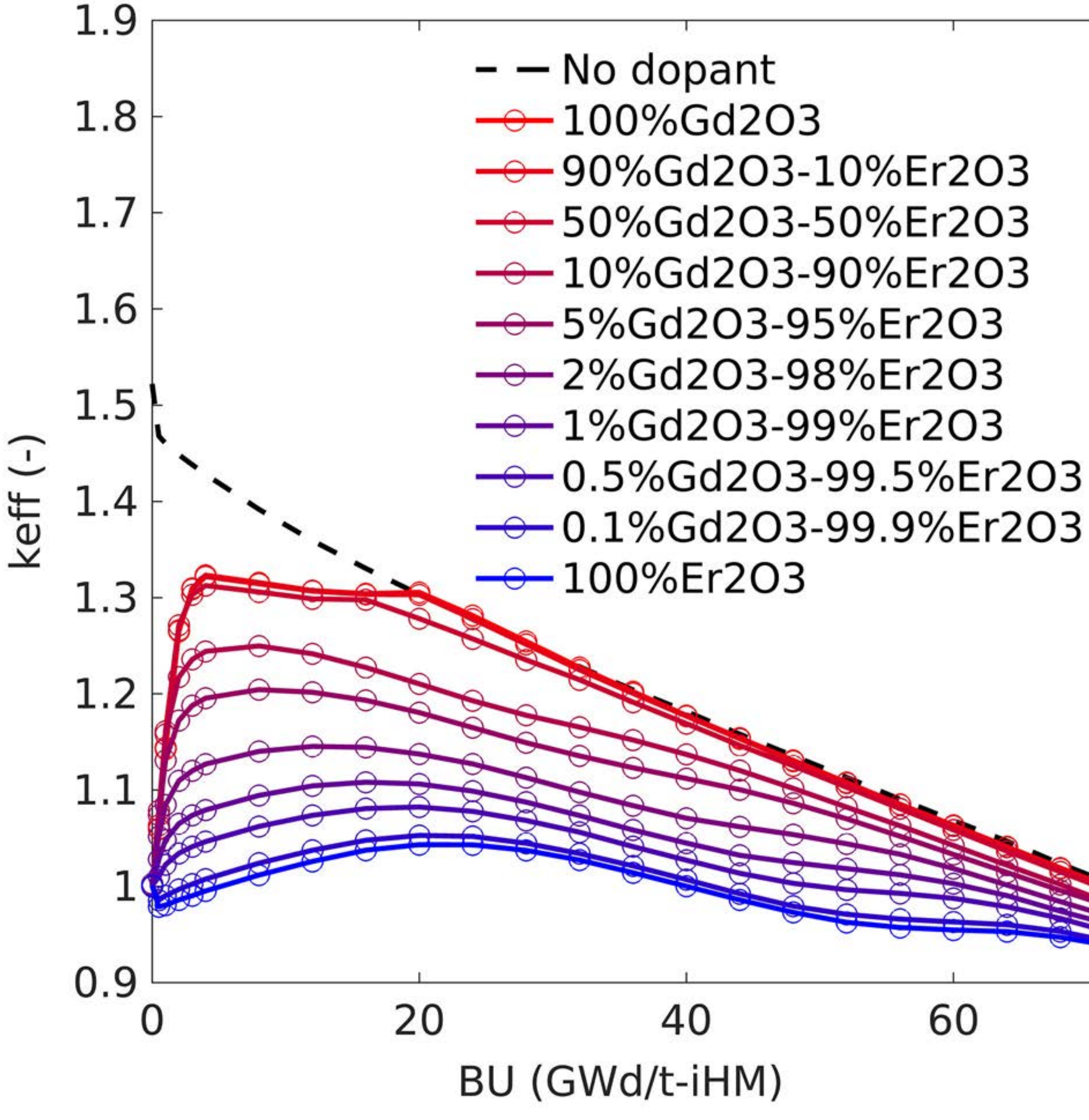



*Figure 1. $k_{eff}$ as a function of the burn up for different $Gd_2O_3$– $Er_2O_3$ mixing ratios.*

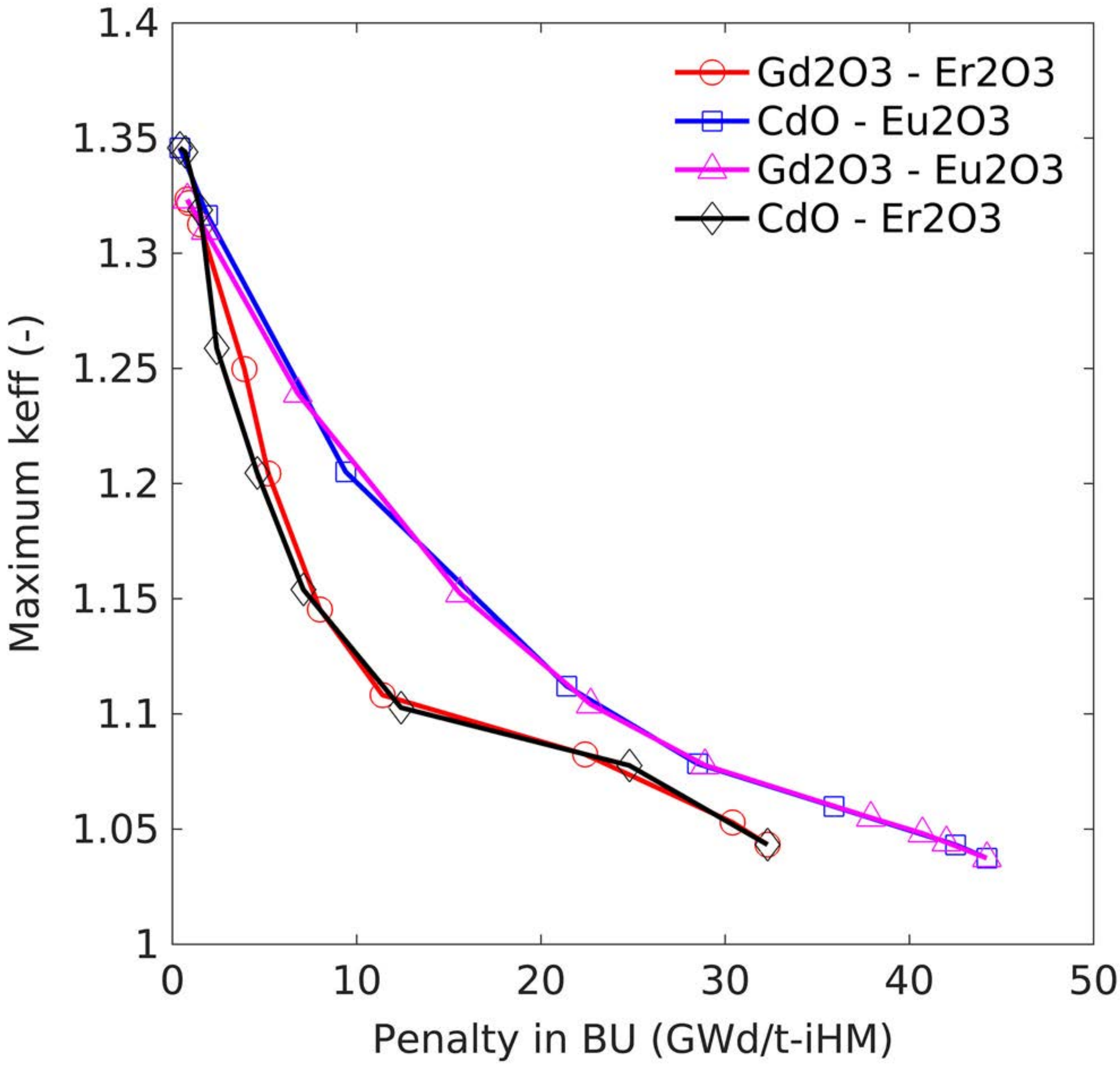



*Figure 2. Maximum keff as a function of the penalty in discharge BU for different dual-dopant options.*

This research supports the Department of Energy's (DOE) objectives by contributing to the safe, reliable, and economic operation of both the nation's current reactor fleet and next-generation reactors. Enhanced reactivity control, achieved through dual BA strategies, directly address safety concerns by reducing the risk of rapid power excursions. The improved fuel utilization translates into longer fuel cycles, reducing operational costs and increasing the economic viability of reactor operations. Additionally, the findings could inform the design and optimization of next-generation reactors, ensuring that they meet DOE's goals of enhanced safety, reliability, and cost-effectiveness.

### Accomplishments

The viability of combining two dopants in BWR HBU $UO_2$ fuel was studied to investigate if the advantages of both single dopants can be combined. The dual-dopant options considered included:

- Mixture of $Gd_2O_3$-doped $UO_2$ and $Er_2O_3$-doped $UO_2$.
- Mixture of $Gd_2O_3$-doped $UO_2$ and $Eu_2O_3$-doped $UO_2$.
- Mixture of CdO-doped $UO_2$ and $Er_2O_3$-doped $UO_2$.
  - Mixture of CdO-doped $UO_2$ and $Eu_2O_3$-doped $UO_2$.

The mixing ratio of each dual-dopant option was varied from 0% to 100%, and the impacts on core characteristics were examined. Figure 1 compares the $k_{eff}$ curves with different $Gd_2O_3 – Er_2O_3$ mixing ratios as an example, which suggests that the maximum keff and discharge (BU) of dual dopants were bounded by those of the individual single dopants. Also, intuitively, the higher the mixing ratio of a single dopant, the more the $k_{eff}$ curve resembles that of the single dopant.

Figure 2 shows the maximum keff during the fuel cycle as a function of the penalty in discharge BU for different dual-dopant options. Aiming at the same maximum keff during the fuel cycle, using $Gd_2O_3$- $Er_2O_3$ or CdO-$Er_2O_3$ caused less penalties on discharge BU (or cycle length) than CdO-$Eu_2O_3$ and $Gd_2O_3$-$Eu_2O_3$. This emphasized the importance of the appropriate selection of the BA constituents with slowly burning isotopes. Also, The maximum $k_{eff}$ – penalty in discharge BU curves were not linear, which suggested the need of optimization studies for reactor designs to meet maximum $k_{eff}$ and discharge BU requirements (simple linear interpolation cannot provide reasonable estimates).

The beginning of cycle fuel reactivity temperature coefficients and moderator void coefficient of the four dual-dopant options were compared. Because the reactivity coefficients of the dual-dopant options were bounded by those of the single-dopant options, using dual dopants did not have negative impacts on the reactor core safety.

# Scaling of Fission Gas Release

*Principal Investigator: Ian Ferguson (Oregon State University [OSU])*
*Team Members/Collaborators: Tianyi Chen (OSU); Ramon Yoshiura, Daniel Wachs (All Idaho National Laboratory [INL])*

***Accelerated fuel experiment design complemented by scaling analysis results in data that is verifiably applicable to a full-scale system.***

This work examines the impact of scaling $UO_2$ fuel rodlet radii and their initial grain size on fission gas release behavior similarity. INL's fuel performance code, Bison is used to model a $UO_2$ rodlet irradiated at a constant average linear heat generation rate of 30 kW/m until a burnup of 9% FIMA is achieved [1]. The prototypical (full scale) and model (scaled) fuel rodlet radii are 5 mm and 2.5 mm, respectively. Additionally, the initial grain radii are 50 µm and 25 µm. The Dynamic System Scaling methodology is applied to investigate the similarity of the fission gas release mechanistic evolution throughout the steady power irradiation for the rodlets [2].

## Project Description

Fuel testing and qualification is a costly and time-consuming process, taking anywhere between 20-25 years [3] for a full-scale integral test. These lengthy tests effectively bottleneck the deployment of new fuel designs. Smaller scale separate effects tests may be used to reduce the financial and temporal demands of qualifying nuclear fuel. Idaho National Laboratory's (INL) Fission Accelerated Steady-State Tests and Oak Ridge National Laboratory's Minifuel tests are both examples of this kind of scaled fuel testing technique [4][5]. One popular technique to accelerate this process is scaling the fuel rod radius to increase the burnup rate. While these burnup-accelerated, scaled fuel tests are promising tools to accelerate the fuel qualification process, it is essential to demonstrate that the data from the scaled system is physically representative of the full-scale system's behavior. As materials and fuels testing continues to move towards lower length scale and accelerated separate effects test, a scaling framework becomes increasingly more necessary to validate that the results from these tests are applicable. Providing this kind of data validation is essential to deploying safe, reliable fuel designs at a more rapid pace.

## Accomplishments

Altering the physical dimensions of the fuel, the neutron flux, and the timescale of the experiment may induce dissimilarity in the underlying mechanisms that drive fission gas release. Mechanisms are often enhanced or suppressed due to scaling and characterizing the impact of this on scaling is necessary for determining how physically representative the scaled data is of the original system. Dynamic System Scaling enables a visualization of these changes to the underlying fission gas release mechanisms through a mathematical treatment of the fission gas release data to create the dimensionless three-coordinate system called the phase space.

This work will examine one facet of the phase space for the scaled rodlets. The coordinates of the facet are ($\beta,\tilde{\Omega}$). $\beta$ is the normalized concentration of fission gas atoms released, where it is normalized against the concentration of fission gas atoms released at the end of the irradiation. $\tilde{\Omega}$ is the effect metric and is the normalized time rate of change of the fission gas released. Changes in the magnitude of the effect metric indicate the fission gas release mechanisms are accelerated or decelerated. Points of inflection (features) in the ($\beta,\tilde{\Omega}$ ) phase space may indicate the activation of a new mechanism or the influence of two competing mechanisms. Normalized concentration of fission gas released is plotted against time in Figure 1 below. Applying the Dynamic System Scaling methodology yields the phase space for the 50 μm initial grain size rodlets shown in Figure 2.

The mechanistic evolution of the underlying fission gas release mechanisms is considered perfectly similar if the curves for the prototypical and model systems overlap one another throughout the entire process. The features (points of inflection) and effect metric magnitude appear to be suppressed in the half radius rodlet. After analyzing the fuel temperature and microstructural data, it appears the features present in both phase spaces are created, in part, by the competition between three rate processes: thermal diffusion, grain growth and refinement.. However, the features created by these rate processes appear to be less discernable for the scaled half radius rodlets.

This is due to the temporal acceleration of the burnup supprressing the effect of thermal rate processes like diffusion and grain growth for the half radius rodlet. The burnup rate scales with the inverse square of the rodlet scaling ratio, meaning that the scaled rodlet acheives 9% FIMA in 1/4 the amount of time it takes for the full size rodlet. Additonally, the microstructure is not similar between the full and half size

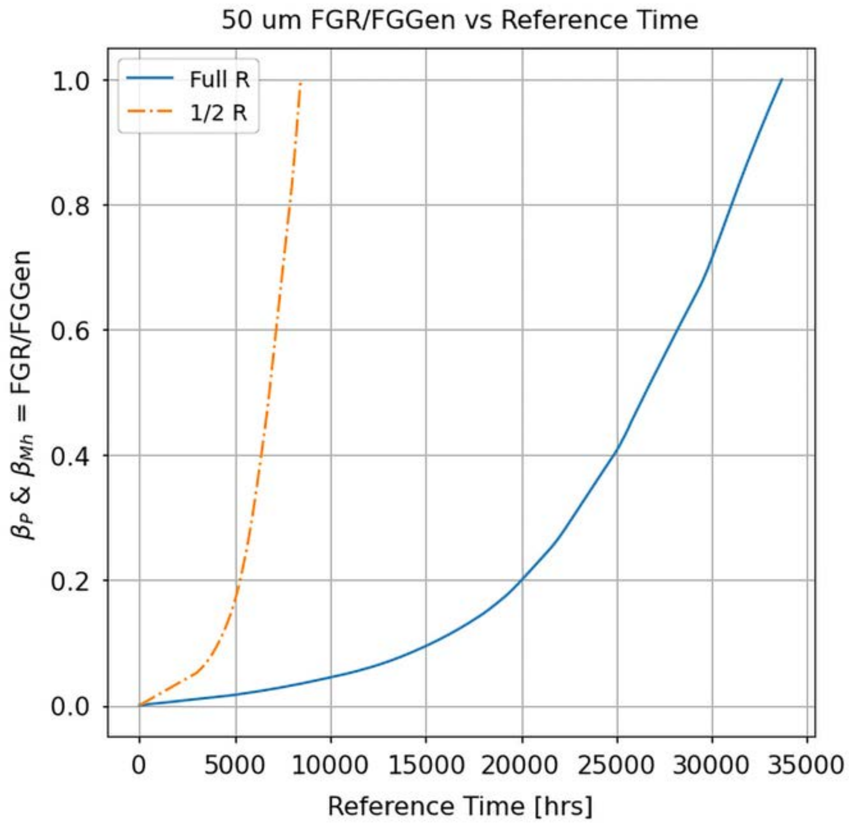
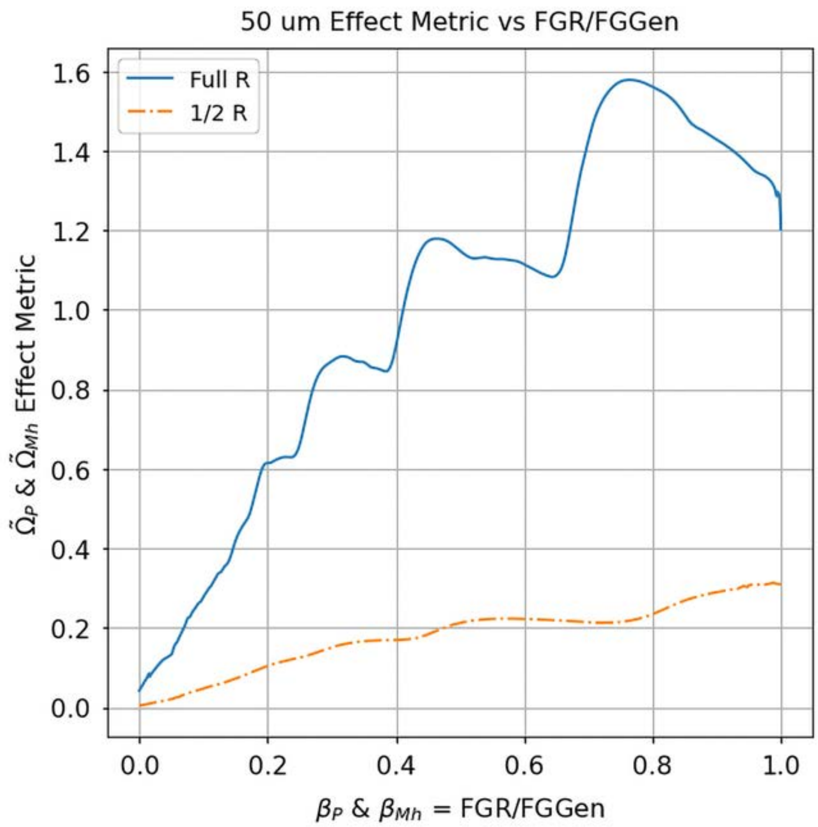



*Figure 1. Normalized concentration of fission gas released plotted against time for rodlets with an initial grain size of 50 μm.*

Figure 2. 50 µm initial grain size phase space.




rodlets – which contributes to the dissimilarity in fission gas release behavior as the number of grains and grain boundaries per unit area are not preserved when the initial grain size is not scaled proportionally with the fuel rodlet radius. To investigate the importance of scaling the initial grain size with the fuel radius to improve the physical similarity of fission gas release, Dynamic System Scaling is applied to the data from the full radius 50 µm initial grain radius rodlet and the half radius 25 µm initial grain radius rodlet. The phase space is shown below in Figure 3.

Preserving the microstructure improves the physical similarity of the fission gas release mechanistic evolution. Although the features in the half radius rodlet phase space are still suppressed, the data is more representative of the full radius rodlet fission gas release behavior. This scaling relationship between the initial grain size and fission gas release similarity should be considered when designing future burnup accelerated fission gas release tests. It may also be advantageous to investigate the impact of temporal acceleration on other key fuel performance phenomena that are influenced by thermal rate processes.

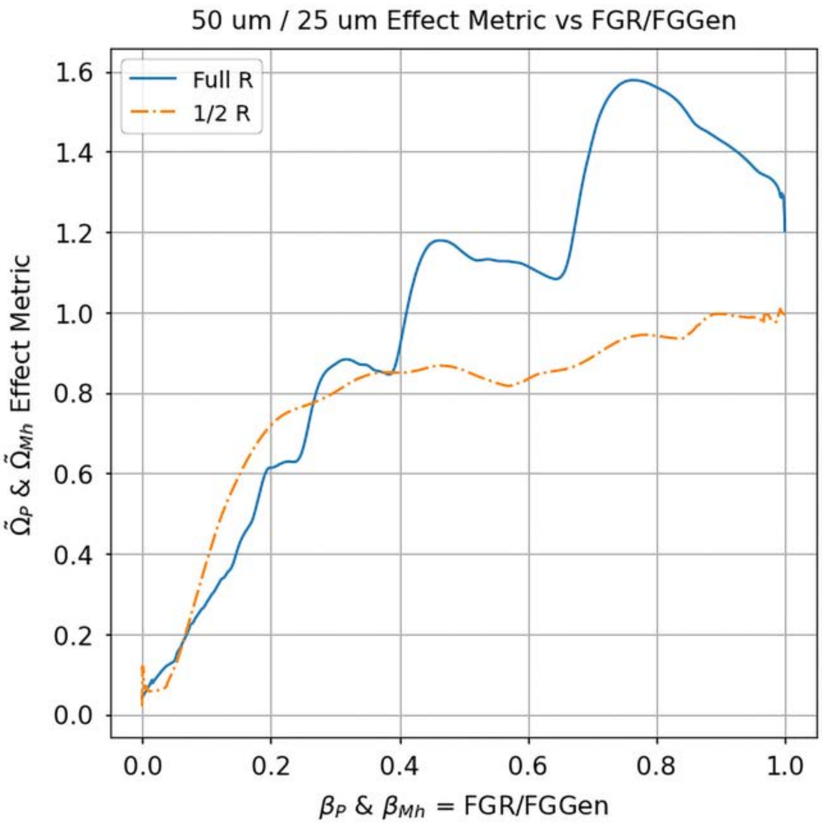



*Figure 3. Microstructure-preserved scaled rodlet fission gas release phase space.*

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# 2.7 ACCIDENT TOLERANT FUEL INDUSTRY ADVISORY COMMITTEE

## Industry Advisory Committee Summary

*William Gassman, Constellation*

The Advanced Light Water Reactor Fuel Advisory Committee was established in 2012 to advise the Advanced Fuel Campaign (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 within their respective institutions.

During the past year the committee provided important industry input to the Department of Energy (DOE) regarding utility and fuel vendor perspectives. This included input on the potential benefits of near-term ATF concepts, extending the burnup limit and licensed enrichment of current fuels and continued efforts in testing, and 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 $UO_2$ and coated clad concepts. Steady state/ramp testing/transient testing infrastructure needs and gaps remains a priority, in particular Advanced Test Reactor (ATR), Transient Reactor Test Facility and Severe Accident Test Station at Idaho National Laboratory and

Oak Ridge National Laboratory. Further focus is aimed on test reactor and post-irradiation examination (PIE) capability, and the forthcoming development of the ATR I-loop. Along these lines industry focus also centers on the maintenance of the Loss of Coolant Accident (LOCA) Test Plan and national laboratory long term PIE schedule. Used fuel transportation, packaging, handling and receipt issues have been a concern; along with future plans for investigation of the longer-term ATF concepts such as silicon carbide and advanced steel cladding and next generation tri-structural isotropic and metallic fuels testing and qualification. Another key issue is continued coordination between DOE and Industry groups such as EPRI and NEI (particularly in the areas of LOCA, accident source term, and high burnup and time at temperature phenomena) and with industry programs such as Framework for Irradiation Experiments, High Burnup Experiments in Reactivity initiated Accidents, LWRS and Nuclear Energy Advanced Modeling and Simulation. The committee also asked that DOE stand by to provide assistance as necessary for uranium supply concerns and 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.

## Westinghouse Electric Company FY24 Accomplishments

*Principal Investigator: Edward J. Lahoda (Westinghouse Electric Company [WEC])*
*Team Members/Collaborators: WEC; General Atomics (GA); Bangor University (BU); Idaho National Laboratory (INL); Los Alamos National Laboratory (LANL); Oak Ridge National Laboratory (ORNL); Trinity College, Oxford University (TC); 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); Southern Nuclear Company (SNC); Air Liquide (AL); Royal Institute of Technology, Sweden (KTH); Karlsruhe Institute of Technology (KIT)*

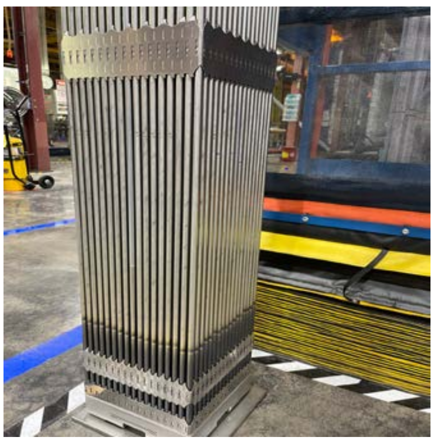

*Figure 1. Assembly with LTRs of Cr coated cladding for Plant Vogtle-2 insertion in 2025*

*Figure 2. Low Enriched Uranium+ ADOPT pellets for loading into the Plant Vogtle-2 LTRs.*

Westinghouse is working to commercialize accident tolerant **EnCore®*** fuel (ATF) designs. These include advanced Cr coatings using nitrogen cold spray on zirconium alloy cladding and **SiGA®*** silicon carbide (SiC) cladding with the capability of >5% $^{235}U$ enriched **ADOPT**™* fuel ($Cr_2O_3$+$Al_2O_3$ doped $UO_2$) and $U_{15}N$ 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 into Vogtle Unit 2 in early 2025. These will contain rods of Cr coated cladding with ADOPT greater than 5% enriched $^{235}U$ 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% $^{235}U$ fuel provides 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**) with liquid metal bonding between the fuel pellets and **SiGA** cladding 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 test at Westinghouse Churchill. Westinghouse is also pursuing alternative approaches to use SiC with Zr cladding as well as for advanced integral fuel burnable absorber materials. To accelerate licensing of these new products, Westinghouse is pursuing implementation of atomic scale modeling (ASM) for property evaluation of topical reports and testing and use of in-rod sensors in lead test rods (LTR) to produce data in real time to validate the atomic scale modeling predictions.

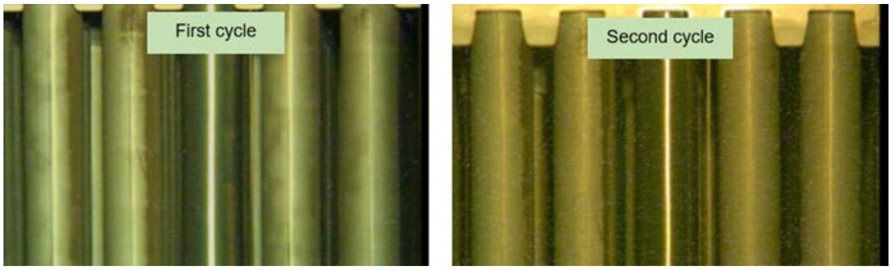


*Figure 3. Comparison of first and second cycles at Doel-4 show little crud buildup on Cr coated rods.*

## Accomplishments

Licensing of the **EnCore** fuel has rapidly advanced during fiscal year 2024. Westinghouse has received the final Safety Evaluation Report (SER) for **ADOPT** pellets and for burnup extensions to 68 MWd/kgU. The Higher Enrichment topical report was sent to the Nuclear Regulatory Commission (NRC) on June 28, 2023, and Requests for Additional Information were received in July 2024. A draft SER is expected by February 2025. SNC received NRC authorization on August 1, 2023, to receive and operate nuclear fuel with enrichment exceeding 5 w/o $^{235}$U, a first in the United States for a commercial reactor. Constellation Energy Corporation received NRC approval on their License Amendment Request to reload **ADOPT** fuel in Cr coated fuel rods in Byron-2 to achieve burnups of 75 MWd/ kgU. These will be shipped to Idaho National Laboratory (INL) in 2025.

The manufacture of Cr coated tubes for the Vogtle-2 LTRs was completed (Figure 1) as was the manufacture of ~6% $^{235}$U **ADOPT** pellets (Figure 2) for insertion in 2025.

Successful post irradiation examinations (PIE) on the Byron-2 and Doel-4 rods (Figure 3) were carried out pool side allowing the Doel-4 rods to be inserted for a 3rd cycle. The remaining rods were re-inserted for third cycle exposure in the Byron-2 reactors to reach burnups of 75 MWd/kgU. PIE on the Byron-2 at ORNL and on rodlets from the Advanced Test Reactor (ATR) are continuing. Out-of-pile loss-of-coolant-accident (LOCA) testing using the severe accident testing station at ORNL has been performed on 2nd cycle non-coated sister rods in order to establish and confirm appropriate testing parameters, with the chromium coated cladding (CCC) test specimens undergoing preparation for testing during the next test campaign.

Twenty-five rods total including five ATF rods (Figure 4) were shipped from Byron-2 to INL for PIE characterization of the rods following the second cycle of irradiation as well as ramp tests in the ATR and reactivity insertion accident (RIA) and integral LOCA tests in the Transient REActor Test Facility (TREAT) reactor at INL. Characterization including non-destructive examination (NDE) and destructive examination (DE) is underway, with preliminary results indicating excellent performance of the CCC upon discharge from its second cycle, at approximately 55MWd/kgU rod average burnup. The use of NDE will continue

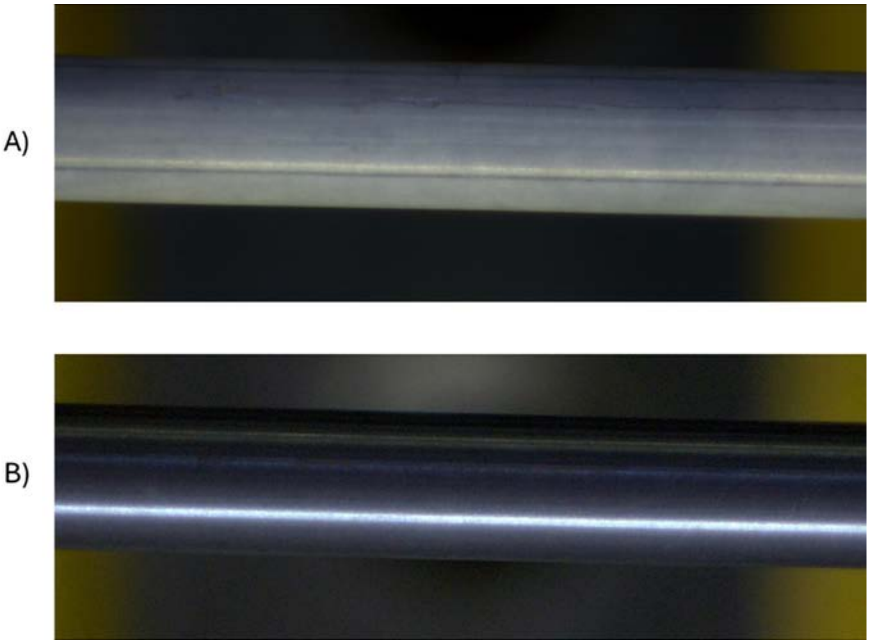

Figure 4. Visual appearance of (a) uncoated zirconium alloy and (b) chromium coated zirconium alloy after discharge from 2nd cycle. Note the pristine appearance of CCC at discharge compared to the coherent oxide.

throughout 2024 and early 2025, with DE and advanced characterization to follow.

Planning for shipment of rods from Kernkraftwerk Leibstadt containing high burnup **ADOPT** pellets to Studsvik is continuing. These pellets will be shipped to Paul Sherer Institute in Switzerland (planned in 2025) for initial PIE and disassembly to allow shipment to Studsvik where fuel fragmentation, relocation and dispersal (FFRD) tests will be carried out. Westinghouse is participating in the Framework for International IrraDiation ExperimentS program of the Nuclear Energy Agency to support RIA testing in TREAT and Cr coated cladding fatigue testing at the Czech Technical University.

Lower length ASM is being used to develop **ADOPT** fuel properties to support future extensions to the **ADOPT** fuel topical report. ASM is being used to develop UN fuel properties to support a future fuel topical report. This work is being pursued by BU in the United Kingdom and at the USC. Development of in-rod sensors for temperature, pressure and fuel swelling is continuing with testing completed at High Flux Isotope Reactor and planned for the Pennsylvania State University Breazeale Reactor.

Basic knowledge on how cracks form and propagate in Cr coated claddings continued using the Berkeley National Laboratory beam line in conjunction with the UBr and TC. To validate the accident tolerant benefit of Cr coated cladding, a second bundle test will be carried out by KIT on Westinghouse supplied Cr coated cladding. The 2024 test (also with Westinghouse supplied grids and coated rods) will model a station blackout scenario with a temperature rise of ~0.5°C/ second and a maximum temperature of ~1600°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 along with probability safety analyses which take advantage of FLEX capabilities will be used as the basis for economic benefits which require that the NRC allow downgrading of some equipment classifications to less than safety grade. Westinghouse is supplying tubes and grids for a third and final test at KIT that will be Cr coated using physical vapor deposition (PVD) at the Westinghouse Springfields facility by National Nuclear Laboratory (NNL) in 2024. This test will be run in 2025.

In support of $U^{15}N$ utilization, oxidation resistance using additives has been abandoned and coatings are being pursued at LANL. LANL has defined a scalable production method for UN powder fabrication using $UF_4$ as a starting point. Both LANL and NNL in a complementary program funded by the United Kingdom are exploring a plasma approach using $UF_6$ as the feed material.

LANL completed UN minidiscs and is waiting on shipment to BR2. BR2 testing is not expected to start until late 2025 at the earliest. LANL completed **ADOPT** pellets achieving ~94.5% density which is considered acceptable. The plan is to ship both UN and **ADOPT** discs together. The testing in the BR2 is being paid for by the Electric Power Research Institute as part of the Fuel Innovation Research Discovery program.

Six unfueled **SiGA** rodlets from GA completed two 60-day irradiation cycles within ATR Cycle 171 under the ATF 2C program. The six unfueled **SiGA** rodlets are currently at Hot Fuel Examination Facility with non-destructive PIE results following. In parallel, GA is also collaborating with Westinghouse autoclave support to improve the corrosion performance in support of upcoming ATR irradiations in ATR Cycle 175.

**_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._**

Westinghouse initiated work to seek alternate means of employing SiC to achieve high temperature performance goals such as ballooning and burst resistance up to at least 1600°C to address FFRD, but at a cost that is low enough to be competitive with current cladding. Free Form Fibers (FFF) is growing SiC fibers directly on the Zr alloy tube to allow the tube to reach temperatures of ~1600°C without suffering ballooning and bursting, thus avoiding FFRD issues. Work at the UW to develop a SiC wrapped Zr alloy cladding using a PVD applied Cr layer to hold the SiC layer in place. The objective of this work is the same as for FFF.

# Framatome Developments on Enhanced-Accident Tolerant Fuel During Government Fiscal Year 2024

*Principal Investigator: Matthieu Aumand (Framatome)*
*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 is swiftly implementing major upgrades to production facilities which will enable industrial scale production of Cr-coated $M5_{Framatome}$ cladding and LEU+ fuel, which along with a fast-paced licensing plan will bring these key technologies to the market in record time.***

Framatome's Enhanced-Accident Tolerant Fuel (E-ATF) strategy relies on a two-phased approach to balance benefits with the anticipated timeline for full-core deployment. PROtect Cr-Cr is Framatome's evolutionary solution which brings incremental benefits compared to standard Zr-$UO_2$ system. The goal is to deploy this product in commercial 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 Silicon carbide (SiC)-based solution with an initial goal to demonstrate proof-of-concept in test reactors by mid-2020s.

In response to the Department of Energy's (DOE) direction, Framatome further expanded the 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 outage.

### 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 four focus areas: (i) Coatings for pressurized water reactor (PWR) and boiling water reactor (BWR) claddings, (ii) Chromia-doped and Chrome-variant $UO_2$ fuel pellets, and (iii) SiC composite materials, and (iv) Advanced Fuel Management to secure the economic health and reduce spent fuel inventories for the current reactor fleet by increasing uranium enrichment and fuel burnup.

A dense Cr-coating on a zirconium-based cladding substrate has the potential for improved high temperature steam oxidation resistance and high temperature creep performance, as well as improved wear behavior. Manufacturing and testing activities are being carried out on Cr-coated $M5_{Framatome}$ cladding in support of batch implementation. Building on the knowledge and experience from the PWR Cr-coated cladding development, a coating material suitable for BWR application is under development.

Chromia-doped $UO_2$ pellets can improve pellet wash-out behavior after cladding breach and reduce fission gas release. Chromia-doped fuel topical report for PWR application was approved by the Nuclear Regulatory Commission (NRC) this


*Figure 1. Visual inspection of the GOCHROM LTRs after 3 and 4 cycles.*

fiscal year. In addition, Framatome developed variants of Cr-doped $UO_2$ fuel pellets for which out-of-pile testing showed a significant increase in thermal conductivity compared to $UO_2$. Framatome fabricated and shipped rodlets containing chromium variant pellets for irradiation in the accident tolerant fuel (ATF)-1 experiment, which started in 2023.

Framatome is developing a composite cladding comprising a silicon carbide fiber in a SiC matrix (SiCf/SiC) for revolutionary performance improvements. The objective is to develop a system not subject to the rapid oxidation kinetics of zirconium while having attractive operating features such as reduced neutron absorption cross-section and higher mechanical strength at accident temperatures. Framatome fabricated test rodlets made of SiC-based composite for irradiation in the Massachusetts Institute of Technology and the Advanced Test Reactor (ATR) test reactors which started in 2022 and 2024, respectively. 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

At INL, the destructive analysis of fueled rodlets irradiated in the ATR trough 30GWd/mtU was completed, concluding with the fracture analysis of mechanical

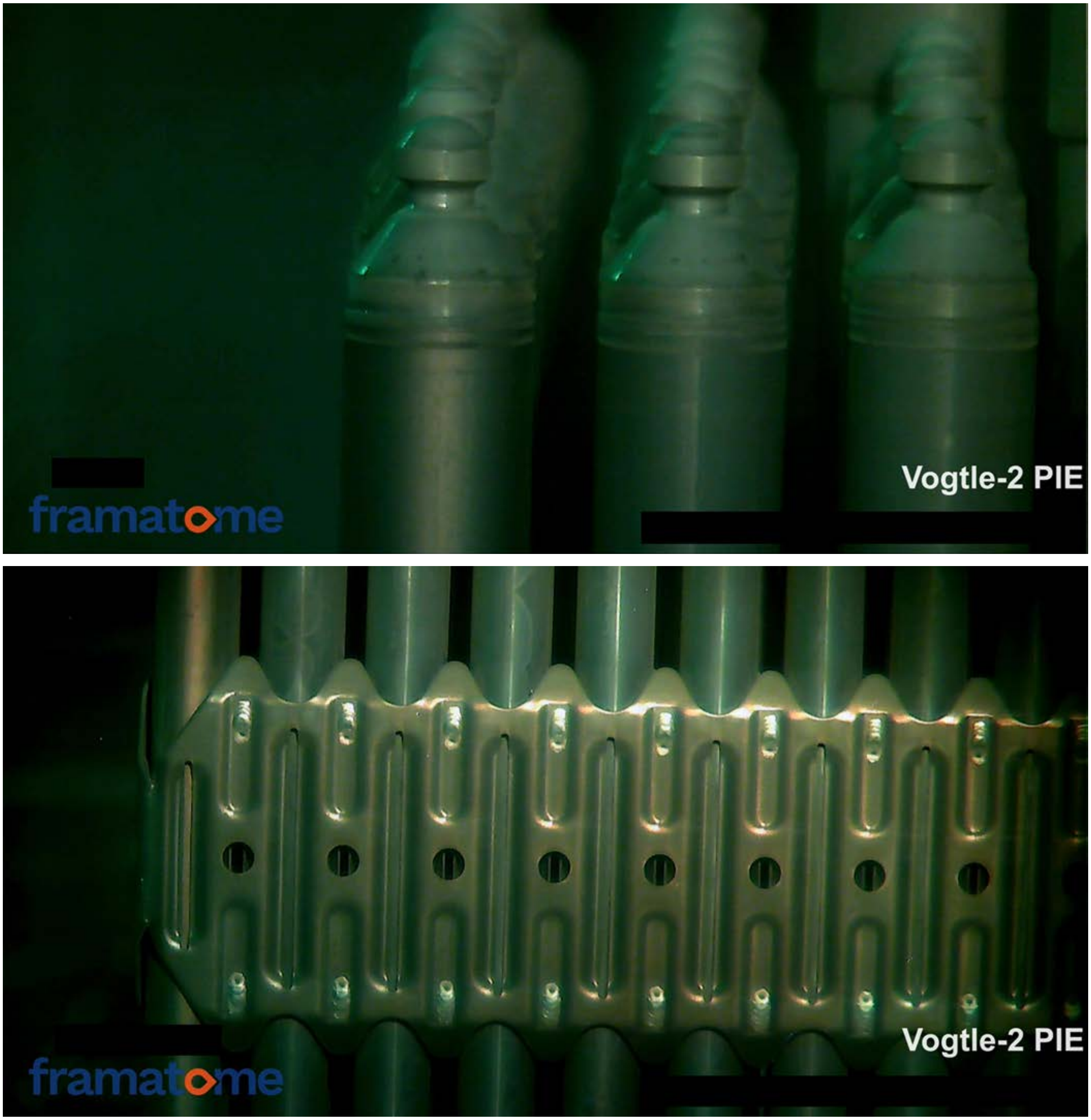


*Figure 2. Visual inspection of the Vogtle LTRs (corner fuel rod) after 3 cycles.*

test specimens. At ORNL, the last specimens irradiated in the High Flux Isotope Reactor, 4in. long fatigue cladding tubes, were disassembled from their capsules which marked a successful first of a kind with this capsule design. Axial tensile testing activities were completed for various irradiation doses. The ORNL team pursued qualifications for plane strain test and fatigue testing and started hot cell testing. The GOCHROM fuel rod retrieved after one 12-month's cycle of irradiation at the Gösgen Nuclear Power Plant (NPP) were defueled at the CEA hot cells.

The lead test rods (LTR) inserted in the Vogtle NPP in 2019 were inspected on site by Framatome, following the completion of their 3rd irradiation cycle (54 months total). The results demonstrate the excellent performance of the coating on the 12 LTRs loaded with Chromia-doped $UO_2$ (Figure 1). The GOCHROM LTRs completed a 4th cycle of irradiation (48 months total) in the Gösgen NPP in Switzerland (Figure 2). On site post irradiation examination (PIE) was completed, demonstrating the excellent performance of the coating. The LTRs were re-inserted for irradiation through 6 cycles total. Burning trough their end of life, the LTRs exhibit a distinctive purple hue from Cr corrosion. All the data collected poolside directly feeds the fuel performance codes. A new irradiation program kicked off, as a total of 36 LTRs started irradiation in an EDF plant. After each cycle (12 months), poolside PIE will occur and LTRs will be harvested for hot cell PIE. This program establishes Framatome's long term plans for building operating experience with the Cr-coated $M5_{Framatome}$ cladding design. At INL, four SiC-based rodlets were inserted in the ATR for irradiation in the ATF-2 water loop, with hot cell PIE planned to start in 2025.

Several hundred LTRs have been produced for the different irradiation programs, providing Framatome with extensive manufacturing feedback. This experience was leveraged to design an industrial Cr-coating pilot. Construction for the warehouse designed to host this equipment has initiated on the Framatome site of Paimbœuf in France, while the construction of the industrial pilot has started at the supplier. It will be commissioned in 2026 and will produce up to 100 000 Cr-coated tubes per year to serve the emerging E-ATF market, providing Framatome with invaluable industrial manufacturing experience for future upscaling. Alongside, Framatome is developing new non-destructive testing and cladding marking technologies to provide future customer with a well-controlled, fully industrialized product in 2026.

The Richland fuel fabrication plant is undergoing major upgrades to enable low enriched uranium (LEU) + production in the near term. Significant progress was made during the fiscal year. Most notably, the complete upgrade of the first of 3 uranium dry conversion lines at the site was completed and work has been initiated on the second line. In June 2024 the NRC approved Framatome's BWR transport container submittal allowing shipments of fresh fuel assemblies enriched up to 8 wt% U235.

# General Electric Progress in Developing Accident Tolerant Fuels

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***GE is actively working on enhancing fuel materials for LWRs, aiming to improve their durability for both near-term and longer-term commercial deployment.***

General Electric has been actively engaged in the research and development of Accident Tolerant Fuels (ATF) 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, 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.

## 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. Collaboration with utility partners such as Southern Nuclear Company (SNC) and Constellation, as well as national laboratories like INL, LANL, ORNL, and PNNL is ongoing. Importantly, the insights gained from studying materials like

IronClad (or FeCrAl) and ARMOR coated cladding can also be applied to future reactor generations (i.e., Advanced Small Modular Reactors), as these claddings 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:

1. During the fiscal year of 2024, ARMOR research and development activities accelerated. A large number of tests representing a full spectrum of reactor conditions were conducted on multiple ARMOR concepts to screen concepts and advance the understanding of coating characteristics and performance mechanisms that contribute to a stable and effective coating. Several distinct classes of coating concepts, each with promising performance attributes, were identified and selected for optimization over this period of work. Both prototyping and testing capabilities at GEV-ARC were greatly expanded. Prototyping improvements allowing expansion of in-house production of novel materials, and coating application for subsequent testing. Testing capability expansion includes additional autoclaves for a greater number of specimens under a wider variety of exposure conditions, and several improved mechanical evaluation capabilities. In addition to equipment, GE scientists and engineers have outlined improved protocols for several tests. Coating screening test methodologies were advanced at both the GEV-ARC and GNF sites. Additionally, GE has expanded partnerships with universities in the areas of advanced characterization. Specimens were delivered to MIT and INL for test reactor insertions, and significant planning and manufacturing progress was made towards commercial Lead Test Assembly (LTA) programs. GE is planning to capitalize on future irradiation opportunities and is working with INL to insert materials in the initial I-loop irradiations when available.
2. The IronClad (FeCrAl) Development work at GE was paused and focus through the end of phase 2C is to continue the low-cost irradiation programs both at the MIT reactor and ORNL's High Flux Isotope Reactor. Data collected from these irradiation campaigns will be utilized to understand post-irradiation mechanical response of current generation IronClad materials. However, small advancements in the characterization of IronClad, mostly partnering with universities (to continue supporting students), continued during the 2024 fiscal year resulting in more than 10 published articles.

3. Hotcell examinations of wrought $C_{26}M$ (IronClad) rodlets irradiated in ATR's center flux trap (ATF-2) continued in 2024 at INL. Hotcell examinations of a wrought $C_{26}M$ LTA segmented rod irradiated in SNC Hatch Unit 1 have also continued in 2024 at ORNL. The hotcell examinations have focused on mechanical testing and microstructural characterization of the earliest generations of the wrought $C_{26}M$ alloy class to establish the extent and cause of irradiation embrittlement. Notably, results from both INL and ORNL collectively indicate that wrought $C_{26}M$ 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. Poolside inspection was conducted on IronClad LTAs irradiated at Constellation's Clinton Power Station in 2023, which indicated that at least one segmented rod had broken at the segment connector; further handling and rod extraction was halted. Irradiations of C26M in Hatch 1 and Clinton are now complete. Two traditional fuel rods containing both $UO_2$ and $(U,Gd)O_2$ fuel pellets within Zircaloy-2 cladding were transported from Clinton to ORNL in order to obtain material performance data on modern GNF fuel designs. Six fuel rods with a variety of Zircalloy cladding materials and doped pellets were transported from Gundremmingen C to Studsvik in order to obtain ultra-high burnup (~ 78 GWd/MTU bundle average, peak pellet > 90 GWd/MTU) material performance data on advanced GNF fuel materials. Fuel rods operated under varying power histories – including rods from a high burnup (HBU) lead use assembly (LUA) – were inspected via GNF's poolside inspection capabilities and harvested for shipment to PNNL by the end of 2024.
4. Interactions with the U.S. Nuclear Regulatory Commission (NRC) continue to progress towards licensing of GNF's ATF products. GNF has received approval for the SNM-1097 License Amendment Request (LAR) to enable the GNF-A fuel fabrication facility to handle LEU+. The significant set of calculations to support the LANCR downstream implementation LTR which was submitted late in 2022 are near completion. Once the calculations are completed in early 2025, the NRC will audit this work to support completing the final Safety Evaluation (SE). GNF has nearly completed all licensing work needed to support reloads of LEU+ using modern codes and methods. GNF plans to submit the PANAC12 (core simulator) LTR in addition to the PRIME (thermal-mechanical) LTR for increased enrichment and burnup extension in

2025. The PRIME LTR is the last engineering code required for full implementation of LEU+, followed by reload readiness once the manufacturing facility modifications are complete. The New Powder Container (NPC) LAR is scheduled to be submitted here shortly with an expected final SER by end of 2025. GNF updated both the NRC (August 2024) and the ACRS (September 2024) with the overall LEU+/HBU licensing plan and program trajectory.

5. GNF continues to progress on updating thermal-mechanical (PRIME) models and methodologies to enable efficient deployment of LEU+ and HBU. GNF is progressing towards submitting an updated PRIME LTR – in addition to supplementing multiple fuel-related topical reports – with an expect submission date in 3Q 2025. The on-going and completed LTA and LUA programs – and the associated poolside/hotcell PIEs – is resulting in high quality data that is being used to supplement GNF's existing database with modern fuel product data. GNF is also working to prepare a separate topical report to cover all other methods needed for HBU licensing, including addressing the issues raised in NRC RIL-2021-13.

# General Atomics Electromagnetic Systems – Silicon Carbide Cladding Development Fiscal Year 2024 Accomplishments

*Principal Investigator: Sean Gonderman (General Atomics [GA])*
*Team Members/Collaborators: Idaho National Laboratory (INL), Oak Ridge National Lab (ORNL), Constellation Energy Generation, Westinghouse Electric Company LLC (WEC), University of South Carolina*

*Figure 1. Visual inspections of a SiGA® rod showing no degradation after irradiation in ATR.*

General Atomic Electromagnetic Systems (GA-EMS) is developing SiGA® cladding as a drop-in replacement for current cladding fuel rods to provide improved safety and economic benefits to the existing light water reactor (LWR) fleet. SiGA® cladding is an engineered ceramic matrix composite (CMC) that uses high purity silicon carbide (SiC) fiber to reinforce a SiC matrix enabling valuable safety and performance benefits like higher fuel burnup and increased power. These benefits are achieved through SiC's excellent high temperature material properties, irradiation stability, and composite nature. GA-EMS is pursuing a progressive irradiation strategy guided by an Accelerated Fuel Qualification (AFQ) framework to efficiently bring this technology to the LWR market by the mid-2030s. In the current program, the goal is (i) Irradiation benchmarking and (ii) fabrication scaling in preparation for readying SiGA® cladding for lead test rods.

## Project Description

Irradiation campaigns at INL, ORNL, Massachusetts Institute of Technology Research Reactor (MITR) are underway to secure the needed performance data. In a separate program, this testing is being supplemented with commercial reactor testing of reduced length SiGA® rods and coupons as part of a collaboration between GA-EMS and Constellation. The combination of test reactor and commercial reactor data will be used to build the case for commercial testing of full-length SiGA® cladding.

In addition to securing irradiation performance data, the scaling of SiC CMC fabrication methods to commercial lengths is a critical objective for the program. GA-EMS is adapting its lab scale fabrication processes to meet the dimensional and material property requirements of full-length SiGA® cladding. This is leveraging high temperature chemical vapor deposition process equipment at GA-EMS that is capable of densifying composite cladding preforms up to 14-ft in length. Downstream processes to control the wall thickness, apply endcap seals, and load fuel are also being scaled to accommodate


*Figure 2. Innovative test vehicles have been developed and built by GA-EMS to support commercial testing of advanced in-core materials. (Top) Cutaway thimble plug irradiation capsule with SiGA® test samples prepared for commercial PWR irradiation in guide tubes. (Bottom) D capsule with SiGA® test samples for commercial BWR core periphery.*

full-length rods. All processes are being scaled with long term manufacturing in mind to ensure the final SiGA® cladding product provides economic value to the utilities.

GA-EMS SiC cladding development is being guided by an AFQ Framework, where physics informed multi-scale modeling is being used to drive both fabrication process improvements and experimental testing. The development of high fidelity SiC composite modeling tools enables accurate prediction of how SiGA® cladding will perform in operation. This is critical for the determining material property specifications and what tests are needed to establish the operation limits of the technology.

### Accomplishments

In fiscal year 2024, first-of-a-find irradiation data has been obtained which confirms joint and cladding wall integrity of SiGA® cladding is maintained following irradiation in pressurized water reactor (PWR) conditions. Unfueled rods irradiated at both MITR and Advanced Test Reactor (ATR) have started post irradiation examination and are providing critical information on the SiC cladding response in representative PWR environments. The MITR test demonstrated that unfueled SiGA® cladding rods maintained geometry and seal integrity over the course of 120 days of irradiation in-core. During this time the rods reached the irradiation saturation point for SiC, a point beyond which further swelling and changes to the material are not expected.

***The irradiation test data being generated by this program, combined with the fabrication scaling to full-commercial lengths is advancing SiC cladding towards lead rod testing in commercial reactors. SiGA® cladding maintains significant strength up to 1900 °C and is stable under irradiation providing substantial improvement to safety and enabling key economic benefits to utilities, like higher burnup and power uprates (Figure 4).***

*Figure 3. Commercial length SiGA® cladding fabrication underway at GA-EMS.*


*Figure 4. SIGA® brings has excellent high temperature strength for nuclear cladding applications.*

A similar unfueled test in ATR has completed with initial results showing all 6 rods irradiated in ATR remained intact after two 60-day cycles. Figure 1 is an image of an unfueled SiGA® being handled in the Hot Fuel Examination Facility at INL. Initial mass data indicated no water ingress and good performance of the cladding seals. Further investigation on the post irradiation thermal and mechanical performance will be completed in early 2025. Successes in these early unfueled tests are providing the critical data needed to finalize the cladding design for the first ATR irradiation of fueled SiGA® cladding in Spring 2025.

To support the test reactor irradiations, irradiation capsule design, sample fabrication, and final capsule assembly has been completed for a first-of-a-kind commercial irradiation in the Limerick Boiling Water Reactor (BWR) in the Spring of 2025. The irradiation will provide SiC performance data under commercial reactor coolant and radiation conditions. GA-EMS has partnered with Constellation to develop, manufacture, and license an irradiation of GA-EMS' SiC material through two different concepts (see Figure 2):

- A PWR irradiation utilizing a modified guide tube thimble plug assembly that is inserted into the top of the reactor core.
- A boiling water reactor (BWR) irradiation utilizing a surveillance capsule design that is positioned in a holder in the downcomer region on the reactor vessel wall.

Currently, an insertion of SiGA® material is planned for the Limerick BWR in spring 2025. This insertion will provide first-of-a-kind irradiation data for SiC material in a commercial reactor environment paving the way for future SiGA® LTRs.

In parallel with the collection of irradiation data, commercial length SiGA® cladding fabrication demonstrations have commenced. Figure 3 is an image of 12-ft long SiGA® cladding rods being handled as part of the composite densification process. Initial evaluation showed uniform mechanical properties over the full cladding length while maintaining critical dimensional targets. Additionally, GA-EMS has demonstrated scaling of the critical rod sealing processes required to ensure the cladding remains gas tight in-core. This year, the endcap sealing method used for SiGA® cladding was successfully demonstrated on a 3-ft length of cladding with the final endcap seal meeting the target gas-tightness requirement. Importantly, the sealing process at 3-ft can be further scaled for full-length cladding applications. A fully sealed, commercial length SiGA® cladding rod is expected in 2025.

# ADVANCED REACTOR FUELS

## Report on Metal Fuel Fabrication Technology Gaps

*Principal Investigator: Randall Fielding (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Alejandro Dayag (INL)*

***The investigation provided in this report will be used to guide future metallic fuel fabrication development.***

The fuel fabrication process used at the Experimental Breeder Reactor-II has become the reference fabrication process for metal-fueled, sodium-cooled fast reactors with over three decades of proven production performance using counter-gravity injection casting. It is noted that some of the challenges encountered during the injection casting process such as charge utilization and waste generation affected production throughput and ultimate economics (metal fuel slug yield). Charge utilization was affected by the large amount of material that was recovered as directly recycled material (casting heels and slug trimmings). An important disadvantage of injection casting remaining today is that the silica quartz molds used. The molds must be broken to remove the metal fuel slug and are not reusable, creating an undesirable radioactive waste stream. Several possible fuel fabrication technologies, as shown in Figure 1, were identified including continuous casting, centrifugal casting, gravity casting, microwave casting, extrusion, and additive manufacturing (3D printing). These techniques were evaluated and prioritized for future metallic- fuel fabrication alternatives development and investment based on economic conditions that improve the gaps identified in the reference process. Continuous casting and extrusion development were judged to be the highest priority for further development, followed by gravity casting. Microwave, centrifugal, or additive manufacturing were seen as low priority due to lack of unique advantages or questionable applicability.

### Project Description

As advanced reactor vendors and designers begin to investigate the economics of the metallic fuel cycle, the cost and impact of the fabrication approaches has become critical. This work provides an analysis of areas within the current reference fabrication process that should be improved to enhance the metallic fuel cycle. Additionally, this report examines several alternative fabrication processes that can fill the technological gaps of the reference process and describes advantages and disadvantages of each. As the high priority processes are further developed and fully documented, fuel fabricators will be able to produce fuel in a more economic and efficient manner, further enhancing the attractiveness of advanced reactor and fuel designs.

## Accomplishments

The current reference process of counter gravity injection casting is a mature technology for fuel slug fabrication, proven to be suitable for remote operation and is capable of mass production. To improve the economics of the metal fuel cycle, the metal throughput yield must be improved, and the related charge utilization increased, and the amount of waste products decreased. It is also important to note that through clearly defining fuel acceptance criteria and related inspection techniques, costly re-work can be avoided and in some cases inspection equipment be less complicated.

Continuous casting should be a high priority for further development. This process could provide a time and cost savings that will eliminate waste, such as quartz molds and greatly increase charge utilization by not requiring a large heel be left over for recycle, resulting in higher yield recoveries, while providing fuel slugs that are dimensional consistent. For metal fuel fabrication, a scaled design must accommodate limited amounts of alloy, generally tens of kilograms, due to criticality concerns and produce fuel slugs on the order a several millimeters in diameter Although continuous casting for fuel fabrication is still immature, the potential for greatly enhanced product yield, waste minimization, and efficiency is enticing. The technique must be fully explored to determine the most reliable and simple processing approach to metal fuel fabrication.

The extrusion process is also a high priority for further development. Extrusion is an industrially common manufacturing process that could be used to produce uranium fuel slugs from a starting uranium alloy ingot. It has recently been used to produce U-10Zr rods with dimensions similar to fast reactor fuel. Despite these successes, further improvements are needed especially to show commercial viability, required secondary processing and to accurately compare fabrication costs to other fabrication methods, such as casting to net shape.

Induction gravity casting has been shown to be feasible on a lab scale using re-usable graphite molds. Because the feasibility has been shown, further development is a medium priority. The next step in development would be to design and test casting systems closer to

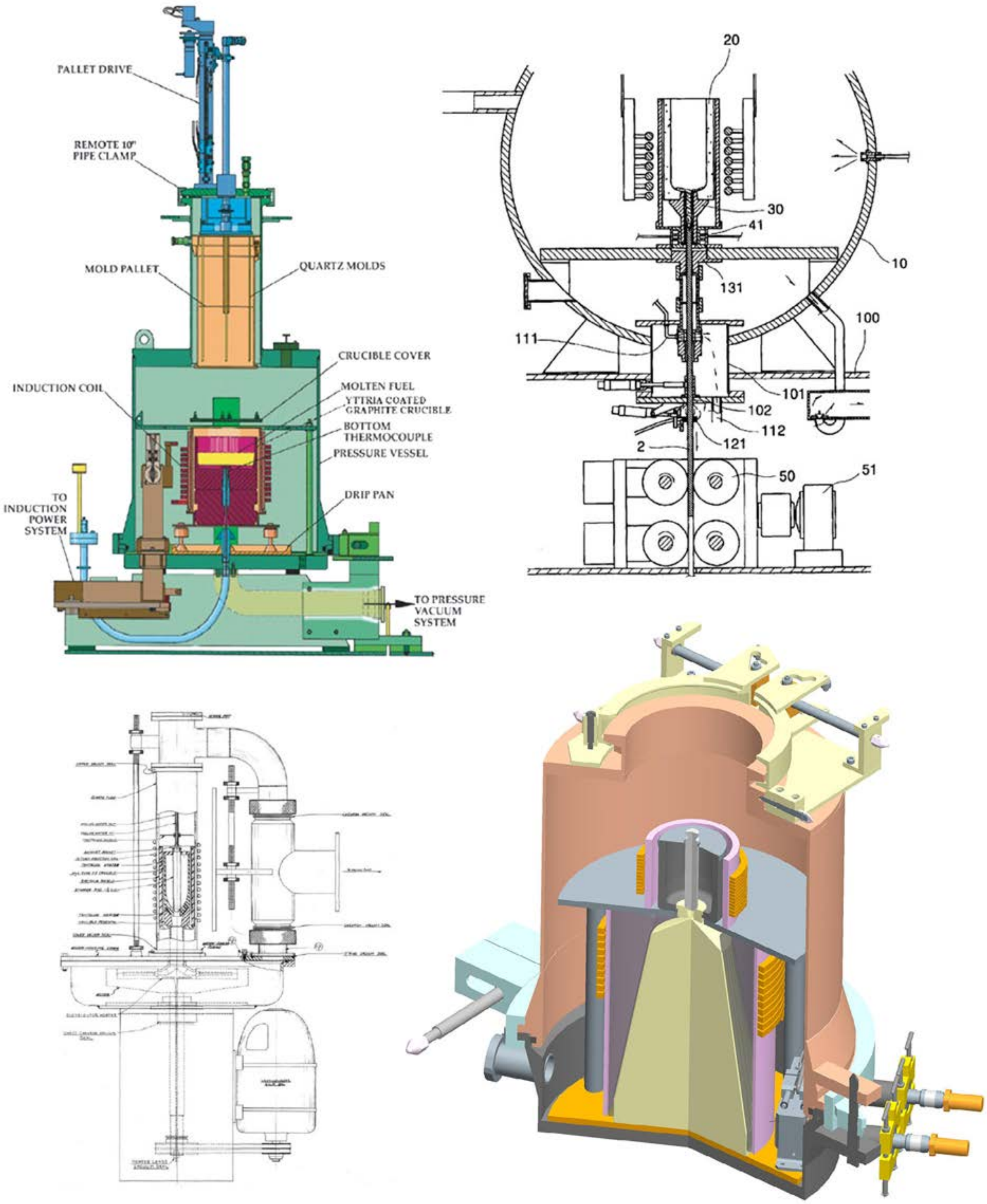


*Figure 1. Possible fuel fabrication technologies.*

engineering or full scale. Although microwave casting is a subset of gravity casting, development of this technique is a low priority. The advantage of a simple furnace chambers used in microwave casting do not adequately set the technology apart from induction casting.

Extrusion and castings will benefit from process simulation as future systems can be tested for feasibility and existing systems further optimized using simulations rather than only experimentation. This emphasizes the importance of determining critical material properties and measuring these accurately to produce applicable simulation results.

Although additive manufacturing is now being used in other industries like automotive, aerospace, and nuclear power plants to make metal parts and assemblies, additive manufacturing to produce "standard" sodium cooled fast reactor metallic fuel slugs is a very low priority and requires additional advantage identification. For standard or simple fuel geometries additive manufacturing is likely less efficient than traditional forming fabrication techniques and the production of and subsequent handling of the pyrophoric uranium alloy powder is also non-trivial and would require substantial research and development.

## AR Cladding Roadmap Program Plan

*Principal Investigator: Benjamin Eftink (Los Alamos National Laboratory [LANL])*
*Team Members/Collaborators: Caleb Massey (Oak Ridge National Laboratory [ORNL]), Stuart Maloy (Pacific Northwest National Laboratory [PNNL]), Tarik Saleh (LANL)*

***The ultimate goal of this planning initiative is to simultaneously (1) improve methodologies to bridge gaps in legacy and modern experimental datasets and to (2) improve the technology readiness level for near-term and next-generation fuel cladding materials as an enabling technology to accelerate the qualification and deployment of advanced fuel technologies.***

This work provides the framework for the core materials advanced reactor strategic plan spanning the next decade, with a specific focus on prioritized workscope necessary to support the Metal Fuels Research and Development (R&D) Plan. This plan, in milestone M2FT-24LA020302051, summarizes key research areas necessary to close gaps related to near-term and long-term core materials challenges, with identified capabilities, expertise, and tools needed for subsequent experiments and solutions. The ultimate goals of this initiative are to simultaneously (1) improve methodologies to bridge gaps in legacy and modern experimental datasets and (2) improve the technology readiness level for near-term and next-generation fuel cladding materials as an enabling technology to accelerate the qualification and deployment of advanced fuel technologies.

### Project Description

The purpose of the milestone is to serve as a ten-year R&D plan to achieve the goals identified to support enabling advanced reactor deployment related to metallic fuel cladding, starting in Fiscal Year 2025. The specific objectives of the document are to i) align program R&D work across technical areas and national laboratories, as well as with stakeholder interests, and ii) aid yearly and outyear scope and budgetary planning activities.

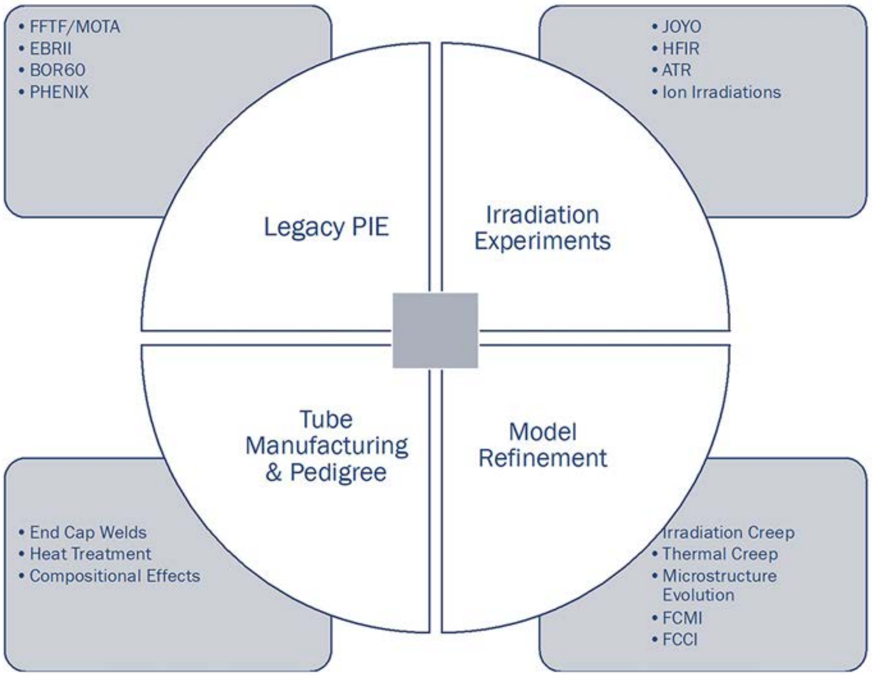


*Figure 1. Core materials research areas to improve cladding property prediction capabilities.*

The successful implementation of this roadmap will directly result in the following notable outcomes:

1. An understanding of HT9 performance necessary to provide a basis for cladding qualification for relevant sodium cooled fast reactor designs, enabled by:
   - Separate-effects data from research reactor irradiations (creep/ fuel-cladding chemical interaction (FCCI)/etc.) incorporated into relevant topical reports for regulatory review.
   - A complete handbook of time-independent and time-dependent high-temperature mechanical properties of HT9 as a function of irradiation temperature and dose.
   - Quantified heat-to-heat variations in irradiation hardening and swelling data for HT9 informed through high-throughput ion irradiations interpreted through prior legacy post irradiation examination (PIE) data.
   - Optimized end cap weld parameters for ferritic/martensitic (F/M) steels with recommendations as to the need for post-weld heat treatments informed from systematic ion/neutron irradiation campaigns.

- Versatile material models, enabled through the Nuclear Energy Advanced Modeling and Simulation toolbox, that can accurately predict HT9 performance as a function of processing methodology and thermal/irradiation history (including simultaneous FCCI and fuel-cladding mechanical interaction phenomena).
- Additional fast reactor (JOYO) irradiated material data (up to and beyond 250 dpa)

2. An increased technical readiness levels (TRL) for prioritized long-term advanced cladding for fast reactors operating at steady-state temperatures at or exceeding 650°C, enabled by:
   - Production of large-batch commercial heats of advanced (F/M and/or austenitic) steels and oxide dispersion strengthened alloys for future campaign testing.
   - Model development for pilger processing of high-strength alloys to reduce tube failures during fabrication.
   - Irradiation effects data on varied microstructures of advanced alloys, as a function of grain size, prior cold work, and in prototypic tube geometries.
   - Additional irradiation performance data for well-known advanced alloys (14YWT) up to 80–100 dpa through legacy PIE (PHENIX/BOR60) and new reactor irradiations (High Flux Isotope Reactor (HFIR)/JOYO).

### Accomplishments

A four-pronged approach was developed and will be implemented over the next 5 years (for HT9) and 10 years (for D9 and advanced cladding concepts). The four research areas are shown in Figure 1, and examples of relevant reactors or generated data are summarized for each area. The areas are legacy PIE, irradiation experiments, tube manufacturing and pedigree, and model refinement. In the report, each of these is described in detail.

Additionally, near term and longer-term activities are planned and shown in Figure 2. In alignment with the Metallic Fuel Research Plan, activities associated with near-term HT9 data collection and qualification are prioritized to ensure that data relevant to technical reports can be made available in time for regulatory review. Relevant data in this area include: 1) Source data relevant to the quality of mechanical testing data used as model inputs 2) Validation data to qualify new modeling approaches, specifically those relevant to reduced-order models integrated into the MOOSE/BISON framework. 3) High-dose PIE data on BOR60 and Fast Flux Test Facility/ Materials Open Test Assembly irradiated specimens 4) PIE data on HFIR-irradiated end-cap weldments and 5) FCCI-relevant models and experimental data (Experimental Breeder Reactor, Fission Accelerated Steady-state Testing, and MiniFCCI).

Longer term development of more advanced cladding materials will involve strongly coordinating with other Department of Energy Nuclear Energy programs such as the Innovative Nuclear Materials and Advanced Materials and Manufacturing Technology programs as well as programs for advanced reactors such as the molten salt reactor or high temperature gas reactor programs. As these novel alloys are developed at lower TRLs, if they show promise in core materials programs and reach the TRL 4-5, they could be investigated through further testing and development to advance the TRL for cladding applications for sodium fast reactors or other advanced reactors.

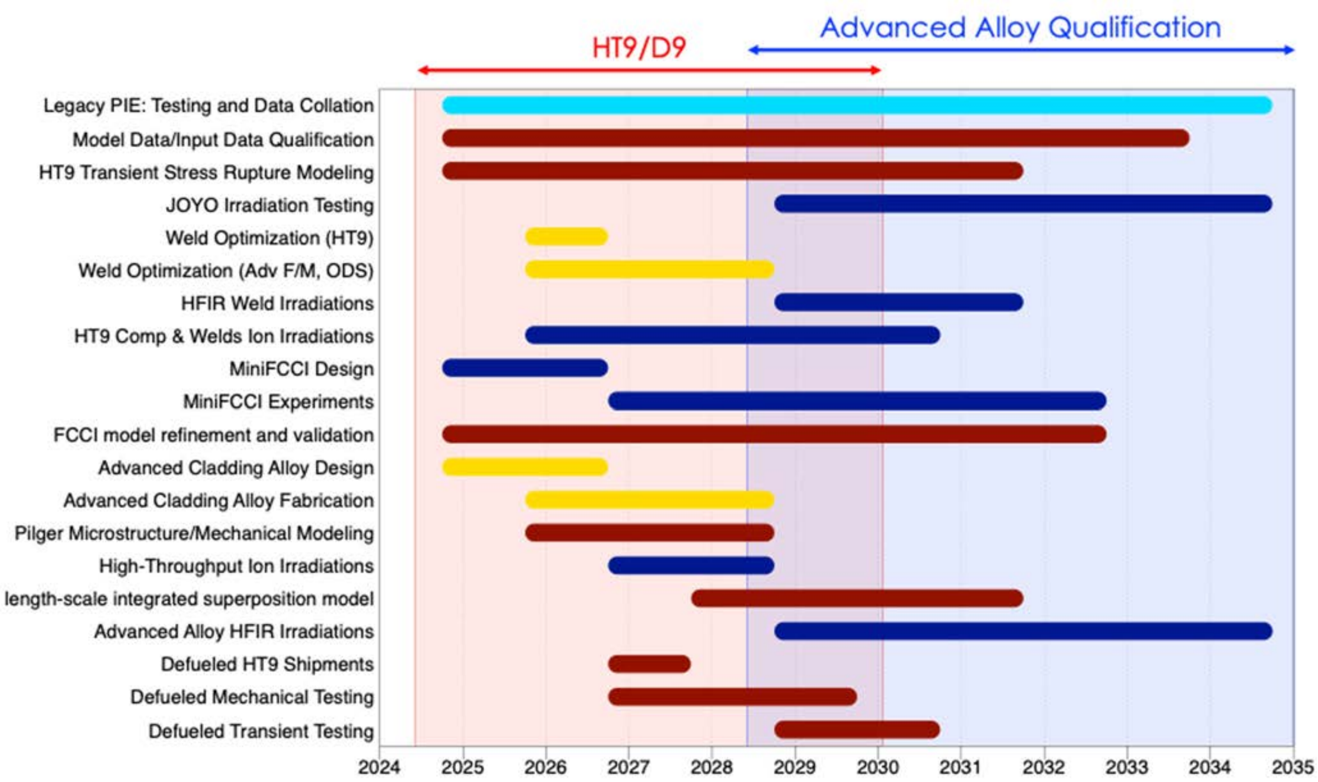


*Figure 2. Projected timeline for Advanced Fuel Campaign-led experiments to increase the TRL of HT9 and longer-term alloys. Initiatives are color-coded with respect to relevant research areas: Legacy PIE (cyan), Irradiation Experiments (blue), Tube Manufacturing and Pedigree (yellow), and Modeling (red).*

# Fast Flux Test Facility HT9 Cladding Microstructure Characterization

*Principal Investigator: Yachun Wang (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Douglas L. Porter, Liang Zhao, Bao-Phong H. Nguyen, Joshua E. Rittenhouse, Tiankai Yao, Tanner J. Mauseth (All INL)*

The sodium-cooled fast reactor (SFR) is a promising candidate for next generation nuclear reactors, operating at extreme conditions which include high temperatures (>500°C core outlet temperature) and significant neutron damage. High-Cr martensitic HT9 steel is an excellent candidate for SFR cladding and duct material due to its compatibility with liquid sodium, good thermal conductivity, resistance to void swelling, and strong creep rupture strength [1-4].However, the harsh in-core environment of SFRs can cause complex microstructural changes and mechanical property degradation in HT-9. Ensuring the safe use of HT9 cladding for metallic fuel requires both a thorough understanding of its mechanical response to microstructure evolution as well as reliable microstructure-sensitive modeling predictions. Microstructure-sensitive modeling of high temperature creep behavior in HT9 cladding for SFR applications currently lack experimental data to model the phenomena accurately. To fill this need, methods to perform microstructural characterization have been developed and performed on HT9.

## Project Description

The development of microstructure-sensitive models to predict high temperature creep behavior in HT9 cladding requires a large, curated microstructure dataset of in-reactor irradiated HT9 cladding. Therefore, the primary objective of this work was to harvest HT9 cladding microstructure data by integrating state-of-the-art transmission electron microscopy (TEM)-based characterization techniques and deep learning (DL)-based methods for microstructure quantification analysis at INL. Ten HT9 clad/U-10Zr samples extracted out of four FFTF (Fast Flux Test Facility) irradiated U-10Zr fuel pins (Table 1) were selected for this project. Microstructure features of interest were dislocations, precipitates, and martensitic boundaries. Dislocation quantification was emphasized in this work because dislocation microstructure and density are most prone to evolution upon neutron irradiation at elevated temperatures. Such evolution is directly linked to plastic deformation behavior and bulk mechanical properties in HT9 cladding. TEM can directly reveal dislocations at nanoscale; however, deriving accurate dislocation statistics remains a challenge.

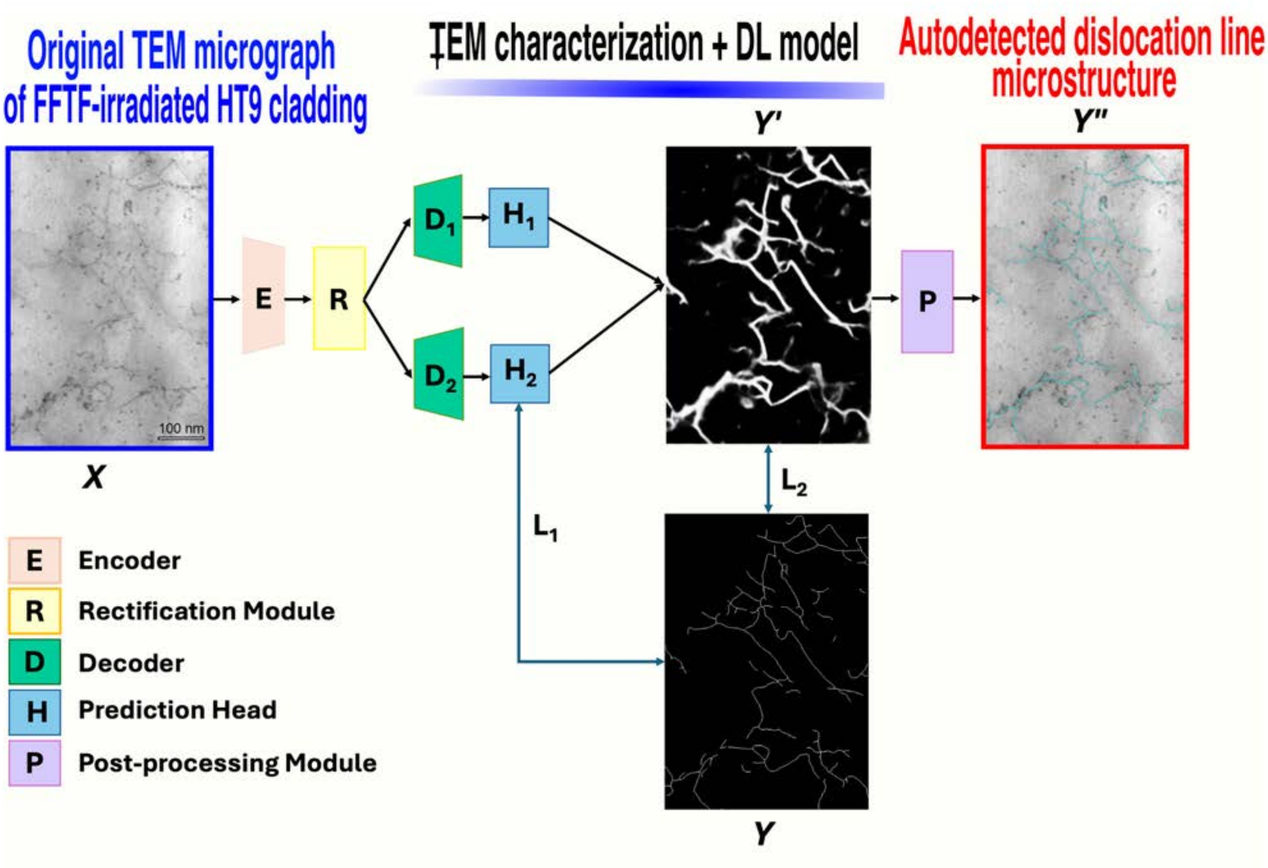


*Figure 1. Overview of the DL-based model architecture for automatic detection and measurement of dislocation line microstructure in TEM micrographs. An input STEM micrograph X is fed into the encoder-decoder based framework. Two new modules, named rectification module and post-processing module, are designed and integrated into the framework for dislocation line detection and length quantification.*

To overcome this challenge, this work developed a DL-based model to automatically detect and measure dislocation line microstructure in as-collected HT9 TEM micrographs. The model proved its high efficacy by accurately identifying dislocations in TEM micrographs containing high density dislocation networks this work established standardized procedures for data collection and data analysis to ensure consistency between datasets. The primary goal of this work was to benefit the nuclear material research community. Microstructural data obtained from this work will feed in microstructure-sensitive creep model development and validation led by Los Alamos National Laboratory. Overall, this effort is expected to improve high-temperature creep modeling of HT9 cladding in SFR environments, crucial for minimizing the risks of cladding failure and maximizing economic viability in the SFR energy sector.

### Accomplishments

Table 1 summarizes the microstructure characterization matrix for all selected samples. For the selected samples, the time-averaged peak inner cladding temperature ranged from 540°C to 635°C. Estimated

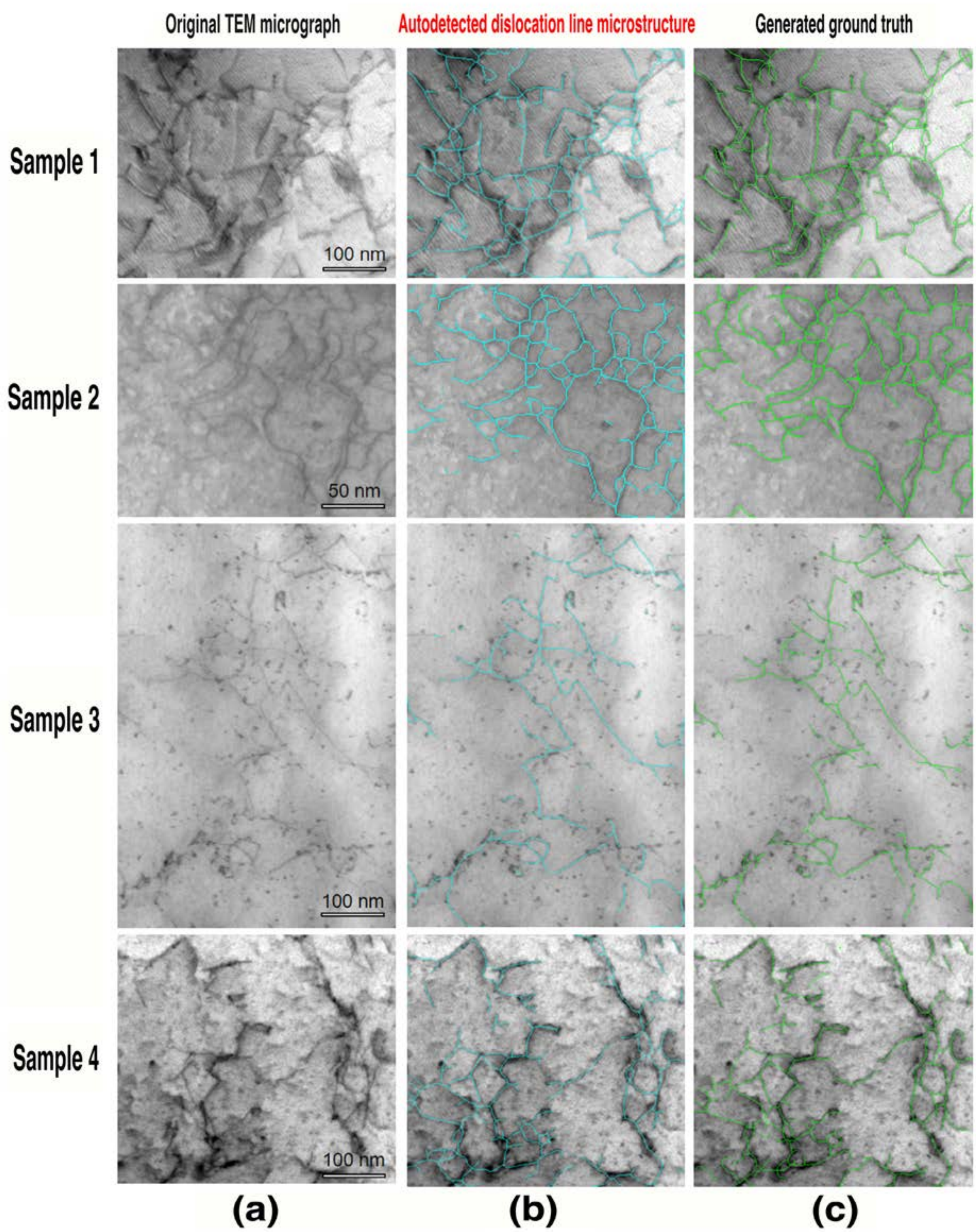


*Figure 2. Dislocation microstructure detection results: (a) Input TEM micrographs; (b) Dislocation reconstruction (post-processing) of initially detected dislocation with rectification module; (c) Generated GT by manual labeling.*

irradiation damage levels ranged from 25 to 83 displacement per atom (dpa). Work completed in FY2024 is highlighted with red check symbol. The entire work scope is planned to complete by FY2025. The accomplishment in FY2024 is summarized as follows: (1) dislocation microstructure characterization and quantification; (2) martensitic boundary characterization through 4DSTEM (four-dimensional scanning electron microscopy).

Dislocation microstructure characterization and quantification: Seven out of the eleven prepared TEM lamella (Table 1) were characterized using a Thermo Scientific Titan Themis scanning transmission electron microscope (STEM) at the Irradiated Material Characterization Laboratory. Hundreds of STEM micrographs of dislocation microstructure were collected and required statistical dislocation analysis. To accelerate data analysis while attempting to prevent human bias in dislocation identification, this work developed an end-to-end deep learning-based method for automatic dislocation line detection and length measurement from TEM micrographs (Figure 1). By treating dislocations as edges rather than cracks or lines, two modules were designed to adapt the network from general edge detection to dislocation detection. Specifically, a rectification module was developed and integrated into the basic framework to enhance feature segmentation for continuous and non-cycle detection of TEM dislocations. Furthermore,

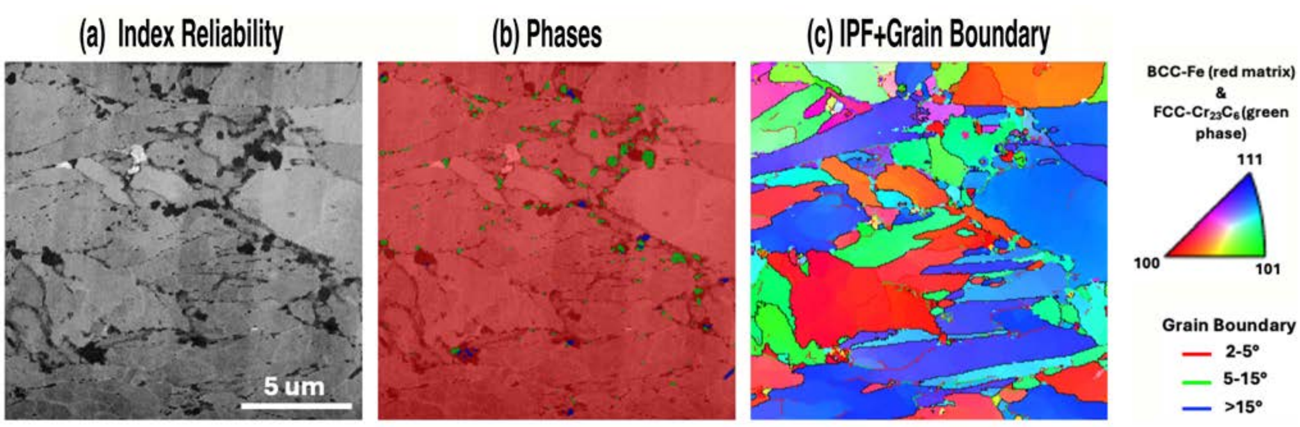


*Figure 3. 4DSTEM characterization of phases, crystalline orientation, and martensitic boundary for a TEM lamella prepared for MNT88T sample listed in Table 1.*

| Fuel pin ID | Sample ID | Time averaged PICT (°C) | Neutron fluence (*10^22 n/cm^2) | Estimated dpa in the HT-9 cladding | Planned TEM lamella preparation using FIB OD=outer diameter ID=Inner diameter | TEM lamella preparation using FIB | TEM characterization microstructure | Data analysis |
|---|---|---|---|---|---|---|---|---|
| Fresh HT9 | -- | -- | -- | -- | 1 | ✓ | ✓ | ✓ |
| 195011 (MFF5) 92239 Heat | MNT86T | 556 | 14.24 | 63 | 1 (OD) +1 (ID) | ✓ | ✓ | ✓ |
| | MNT87T | 612 | 11.97 | 53 | 1 (OD) + 1 (ID) | ✓ | × | × |
| | MNT88T | 635 | 5.57 | 25 | 1 (OD) + 1 (ID) | ✓ | ✓ | ✓ |
| 193045 (MFF3) 92235 Heat | MNT83T | 615 | 6.01 | 26 | 1 (OD) + 1 (ID) | ✓ | ✓ | ✓ |
| 193114 (MFF3) 92235 Heat | MNT54C | 540 | 18.92 | 83 | 1 (OD) + 1 (ID) | ✓ | × | × |
| | MNT46Z | 580 | 17.94 | 79 | 1 (OD) + 1 (ID) | × | × | × |
| | MNT34Z | 630 | 9.12 | 40 | 1 (OD) + 1 (ID) | × | × | × |
| 193019 (MFF3) 92235 Heat | MNT62E | 542 | 17.98 | 79 | 1 (OD) | × | × | × |
| | MNT63E | 578 | 15.11 | 67 | 1 (OD) + 1 (ID) | × | × | × |
| | MNT64E | 605 | 7.68 | 34 | 1 (OD) + 1 (ID) | × | × | × |

*Table 1. FFTF-irradiated HT9 cladding microstructure characterization matrix to support microstructure-sensitive HT9 creep model development. Red check mark indicates tasks completed in FY2024. The black cross symbol means to be completed in FY2025.*

a post-processing module was embedded into the framework to filter out redundant and overlapping dislocations. Following filtration, the method was trained using publicly accessible datasets [5, 6]. Finally, the trained method was used to automatically generate the location and length of dislocations for multiple in-house collected TEM micrographs (Figure 2a).

The full process of generating dislocation length results (Figure 2b) for each tested micrograph (Figure 2a) took only seconds, proving the method's high efficiency. Additionally, the generated dislocation length results showed high accuracy (>96%, Table 2) when compared to micrographs analyzed manually by INL researchers (Figure 2c). Those results highlight the potential impact of the DL-based method developed in this work on saving human power and time for analyzing labor-intense data. This method can be applied to large datasets of TEM micrographs containing high densities of dislocations (on the order of 1014m-2). The manuscript highlighting the end-to-end deep learning-based method for dislocation analysis will be submitted in beginning of FY2025. By the end of FY2025, a large dataset of dislocation statistics for FFTF irradiated HT9 cladding will be generated, documented, and published.

Martensitic boundary characterization through 4DSTEM: In this work, 4DSTEM was applied to map phases and crystalline orientation as well as boundary characterization. Figure 3 shows preliminary 4DSTEM data for one of the MNT88T TEM lamella. The phase map in Figure 3b clearly revealed nanoscale M23C6 precipitates. Figure 3c shows the inverse pole figure along with grain boundaries, where some preserved lath structure is clearly revealed. Figure 3c also reveals the onset of subgrain formation based on the subgrain boundaries (red lines). Comprehensive 4DSTEM data analysis will continue in FY2025.

| Micrograph in Figure 2 | Sample ID listed in Table 1 | Micrograph dimension (pixel) | | DL-model generated dislocation length (pixel) | | Ground truth (GT) (pixel) | Accuracy (%) | Dislocation density ($10^{14}$ /$m^2$) |
|---|---|---|---|---|---|---|---|---|
| | | Height | Width | Figure 2(c) | Figure 2(d) | | | |
| 1 | MNT83T | 326 | 399 | 4508 | 4959 | 4929 | 99.39 | 7.4028 |
| 2 | MNT88T | 231 | 280 | 2208 | 2884 | 2973 | 97.01 | 8.6580 |
| 3 | Fresh HT9 | 653 | 450 | 3564 | 3642 | 3782 | 96.30 | 2.4066 |
| 4 | Fresh HT9 | 458 | 513 | 4792 | 4870 | 4927 | 98.84 | 4.0248 |

*Table 2. DL-based method generated dislocation statistics for HT9 TEM micrographs showed in Figure 2(a). The accuracy (>96%) is calculated by dividing the DL-model generated results in Figure 2(b) with the ground truth (GT; Figure 2c), demonstrating the high accuracy of the model to quantify high dislocation line density (on the order of 1014 /m2).*


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***This work highlights a high-efficiency and accuracy, end-to-end deep learning-based method for automatic dislocation line detection and length measurement from TEM micrographs.***

# Production of Best Practice Heat of Nanostructured Ferritic Alloy 14YWT for Campaign Testing

*Principal Investigator: Caleb P. Massey (Oak Ridge National Laboratory [ORNL])*
*Team Members/Collaborators: David Hoelzer (ORNL)*

The development and deployment of advanced cladding materials in advanced reactors requires the collection of relevant performance data for each cladding concept. Novel cladding concepts typically undergo significant innovation and iteration over the course of their development, and intention should be given to providing a snapshot of present performance. Thus, a modern heat of the nanostructured ferritic alloy (NFA) 14YWT is produced using state-of-the-art procedures for impurity minimization combined with thermomechanical processing methods designed to optimize fracture toughness, tensile strength, and ductility. This new heat of 14YWT, NFA2, will provide the Advanced Fuels Campaign with adequate material for upcoming irradiation and mechanical testing campaigns.

## Project Description

Over the past two decades of development, 19 different batches (i.e., heats) of 14YWT have been produced, and significant batch-to-batch variation has been noted between each heat as a function of varying extrusion ratios, milling parameters, impurity contents, and consolidation temperatures. Unfortunately, the differences in processing and impurity contents in these various heats of this important oxide dispersion strengthened (ODS) advanced reactor cladding candidate have led researchers to believe that ODS materials intrinsically have heterogeneous microstructures with little capacity to balance impurity contents, particle dispersion characteristics, grain size, and consequently the final cladding performance metrics needed for fast reactor environments. In fact, much of the variation in properties of the historical 14YWT materials have been either purposefully pursued (as in the high-strength, nanoscale grain-sized SM10 heat of 14YWT), or have been a result of the evolution of the material over the course of its development.

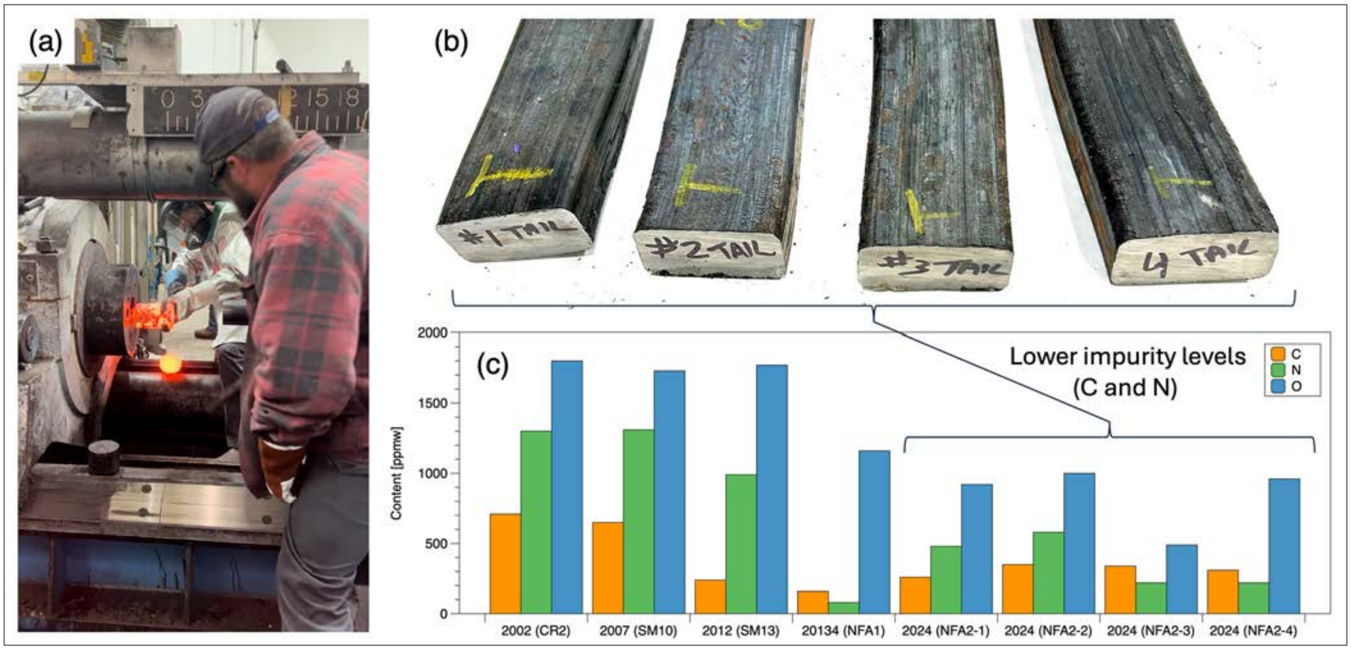



*Figure 1. Summary of produced rectangular bars of 14YWT with reduced impurity content. In (a), the extrusion process is shown for cans to produce (b) the as-extruded 14YWT bar stock. In (c) the impurity contents (C, N, and O) are shown in comparison to historical heats of 14YWT.*

In fiscal year 2024, a new 16kg best-practice heat of 14YWT was produced via mechanical alloying and extrusion, producing four rectangular bars and four cylindrical bars of material that can be used for irradiation testing and tube manufacture, respectively. Following extrusion, the four rectangular bars were sectioned, analyzed for C, N, and O impurity content, and microstructurally characterized prior to mechanical testing.

## Accomplishments

Images of the four new bars, and a snapshot of the extrusion process by which they were consolidated, are shown in Figure 1. These new bars have sufficient oxygen content to produce the high number density of Y-Ti-O rich nanoprecipitates that provide the enhanced irradiation resistance, thermal stability, and mechanical performance expected for 14YWT, while limiting C and N content to below 500 ppmw, elements that in excess can deteriorate key performance metrics such as fracture toughness. Although not shown here, the high-density of

***The new large-batch heat of 14YWT produced in this project will enable the generation of relevant irradiation performance data in test reactors around the world to accelerate the maturity of this ODS fuel cladding concept.***

(1-3 nm diameter) nanoprecipitates within these bars of 14YWT was confirmed and measured to be between 2x1023-4x1023/$m^3$, which is competitive with prior heats.

The mechanical properties of the four rectangular bars show consistent material response between each, even though each bar is comprised of powders from two distinct mechanical alloying runs. This shows that the reproducibility for ODS mechanical properties is improving. This decrease in impurity content, coupled with optimized annealing and extrusion conditions, has a drastic effect on strength and ductility, as summarized in the comparative analysis of tensile data in Figure 2. At temperatures below 600°C, the NFA-2 alloys have higher uniform elongation than the previous SM10 heat, whereas total elongation is comparable, even at these elevated temperatures. The SM10 heat, designed specifically for refined grain size and high strength, also suffered from a lack of ductility at all test temperatures, which has been alleviated with these new variants. In addition, although the strength values are lower than those of SM10, these new, cleaner NFA-2 heats still have yield and ultimate tensile strengths above 1 GPa at room temperature, and they retain yield strength and ultimate tensile strength values in the 200–250 MPa range at temperatures up to 800°C. More importantly, these tensile properties are for the as-extruded condition and may be improved with future thermomechanical processing methods. Specimens from the new NFA-2 heats will be used for future irradiations, such as the upcoming United States/Japanese Atomic Energy Agency joint Joyo irradiation scheduled tentatively to start in 2027.

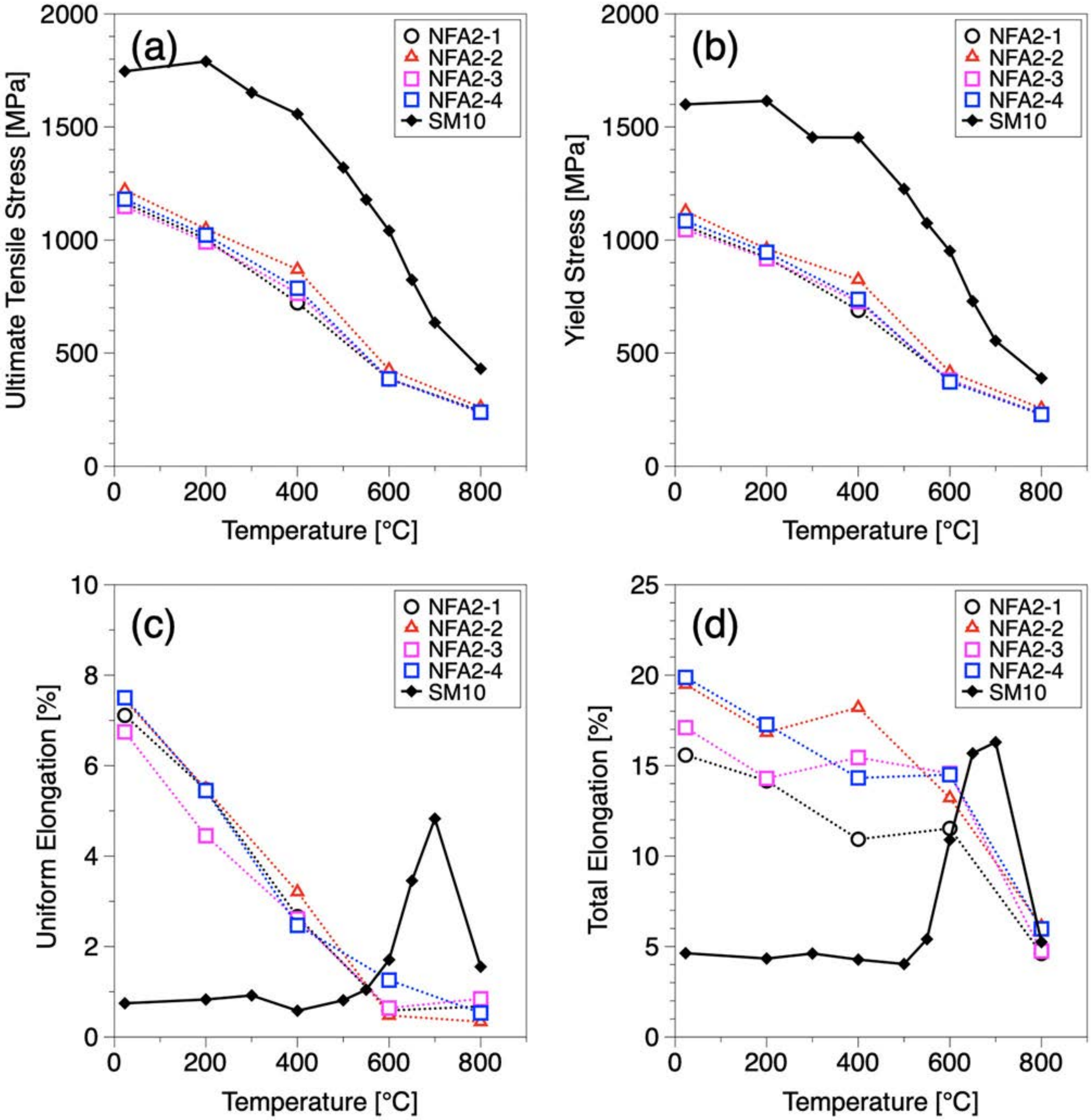



*Figure 2. Summary of high-temperature properties for 14YWT NFA-2 bars, including (a) ultimate tensile strength, (b) yield strength, (c) uniform elongation, and (d) total elongation. Each heat is plotted along with the widely cited SM10 heat of 14YWT, which had the highest strength and smallest grain size of any of the 14YWT materials produced.*

# Ion Irradiation for Accelerated Advanced Reactor Cladding Evaluation

*Principal Investigator: Hyosim Kim (Los Alamos National Laboratory [LANL])*
*Team Members/Collaborators: Nan Li, Ben Eftink, Pengcheng Zhu, Dongyue Xie, Ben Derby, Matthew Chancey, Ellis Kennedy, Yongqiang Wang, Winson Kuo (All LANL)*

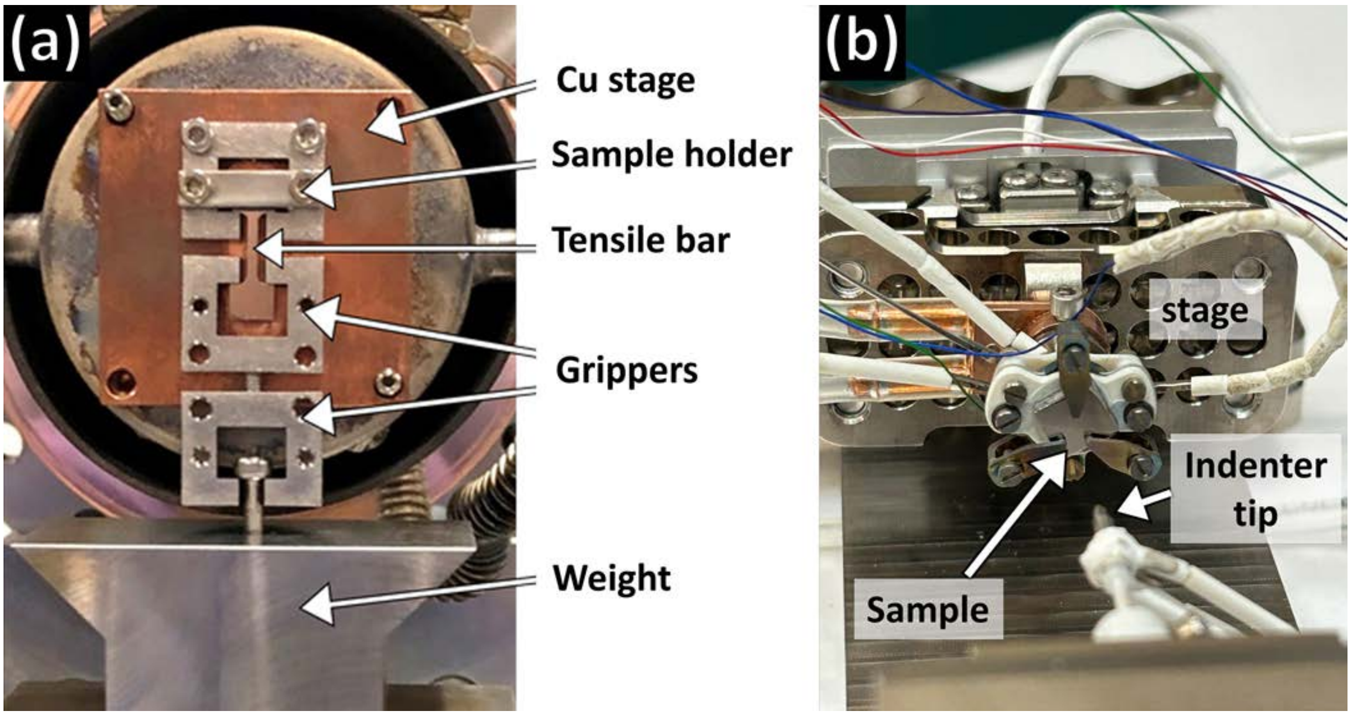



*Figure 1. (a) In-situ straining device developed at Los Alamos National Laboratory. Tensile bar sample is mounted on the stage and irradiated using ions at various temperature. (b) In-situ small-scale mechanical testing device that fits inside the scanning electron microscope with sample mounted on the stage. Ductile-brittle transition temperature can be measured before and after irradiation at various temperatures.*

This research focuses on accelerated evaluation of advanced reactor cladding candidate material, HT9, using ion irradiation and small-scale mechanical tests. Previous study on optimizing HT9 by controlling nitrogen (N) content funded by the Advanced Fuels Campaign successfully showed significantly improved void swelling resistance with higher nitrogen content after 600 peak dpa self-ion irradiation. In this study, two accelerated evaluation methods were used to test the newly developed composition engineered HT9 alloys using ion irradiations. First, the HT9 alloys were irradiated under combined extreme conditions (ion irradiation + stress + high temperature) and void swelling was characterized. Second, to evaluate the ductile-brittle transition temperature (DBTT) change after irradiation, small-scale mechanical testing was conducted via cantilever beam bending tests at different temperatures.

### Project Description

Next generation reactor cladding materials are expected to receive high dose irradiation at high operational temperature. During reactor operation cycles, materials also experience stresses from

**_Accelerated material evaluation via ion irradiation expedites material testing and development process and combined extreme effects and ductile-brittle transition temperature were successfully studied on newly developed HT9 steels._**

various sources such as temperature gradients and fuel swelling which can lead to irradiation assisted stress corrosion cracking. Through this research, the stress effect on void swelling in the advanced reactor candidate material, HT9 alloy, under high dose irradiation (200 peak dpa) was investigated via an in-situ straining capability developed at LANL for coupled extreme effect studies. The low nitrogen (.001 wt%) and the high nitrogen (.044 wt%) HT9 alloys were ion irradiated with and without tensile stress and the post irradiation characterization was conducted using a transmission electron microscope to compare void swelling behavior. The HT9 alloys have been considered as potential structural materials for future advanced reactors, however, irradiation-induced embrittlement has been a major obstacle of their deployment. For example, it is important to know the change in DBTT after irradiation. The accelerated testing we performed through a combination of ion irradiation and micro cantilever beam test at different temperatures provides insights on prediction of material failure. These accelerated coupled extremes effect and DBTT tests after ion irradiation have never been tested before on the new composition engineered HT9 alloys and help fast material screening for next generation reactor designs.

### Accomplishments

Accelerated testing on the composition optimized HT9 alloy was conducted to investigate the combined irradiation and stress effect on microstructure and DBTT of the pristine and irradiated HT9 alloys using a micro-cantilever beam bending test. Previous study showed that the HT9 alloy properties can change dramatically by optimizing minor solute such as nitrogen and the high N HT9 (.044 wt%) showed superior void swelling resistance compared to the low N HT9 (.001 wt%). Technical goals of the FY24 for the next generation reactor material research and development are 1) conducting high dose ion irradiation (200 peak dpa) on both low and high N HT9 alloys using an in-situ straining device developed at LANL as shown in Figure 1a to understand the effect of combined extreme conditions on these alloys and effect of nitrogen under such environment and 2) developing a small-scale DBTT measurement process for ion irradiated samples by utilizing a micro-cantilever beam bending test at various temperatures. To achieve the first goal, the N varied HT9 alloys were irradiated using 3.5 MeV Fe ions at 450 °C with and without tensile stress and the stress was roughly 8 % (38 MPa) of the 0.2 % offset yield strength. Tensile bar samples were mounted on the Cu stage using sample holders and grippers as shown in Figure 1a and a thermocouple was clipped between the tensile bar head and the sample holder to measure the accurate temperature during irradiation. A total of 3 kg of weights was used to provide tensile stress and irradiation time was 11-13 hours to reach 200 peak dpa. Contrary to expectations that tensile stress will promote void swelling, the low N HT9 showed suppressed void swelling when irradiated with the tensile stress as shown in Figures 2a and 2b. No voids were observed in the high N HT9 after 200 dpa

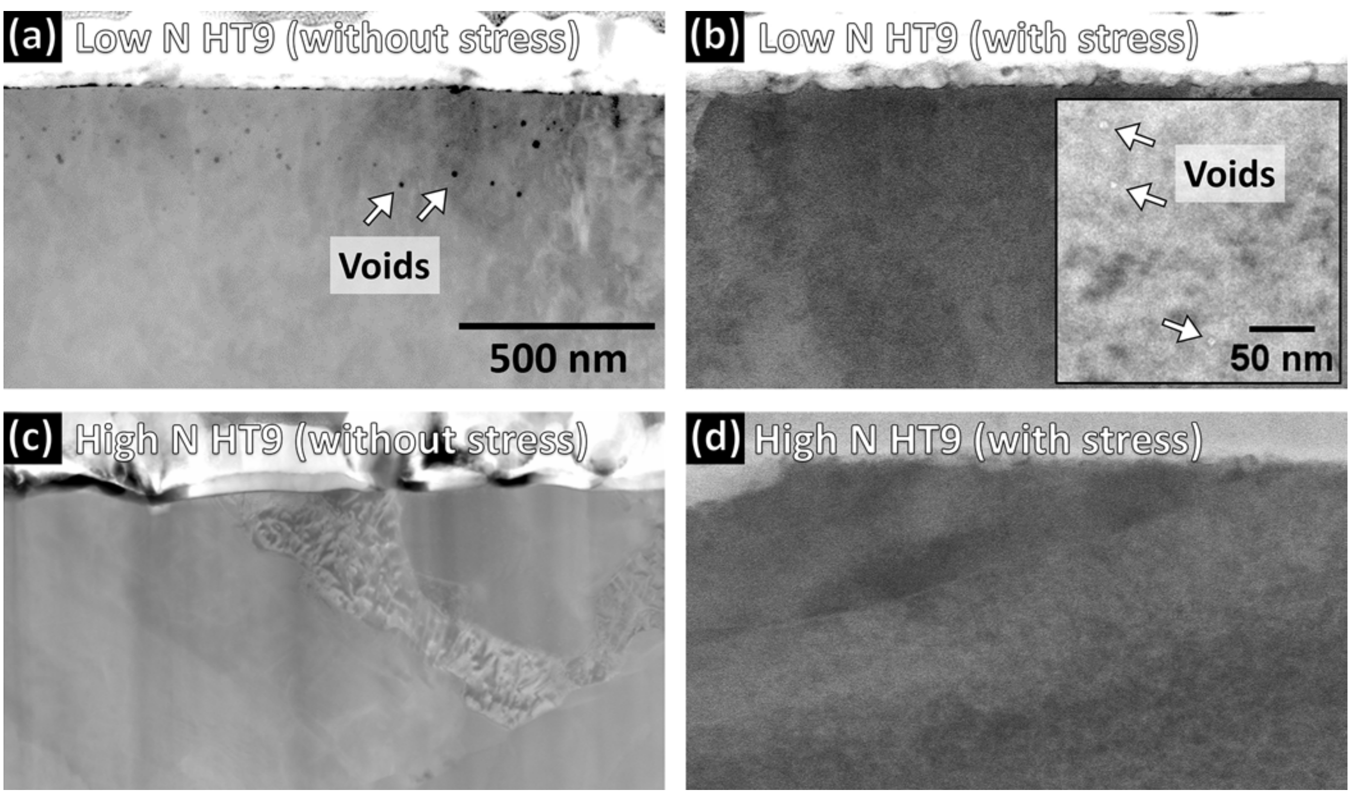
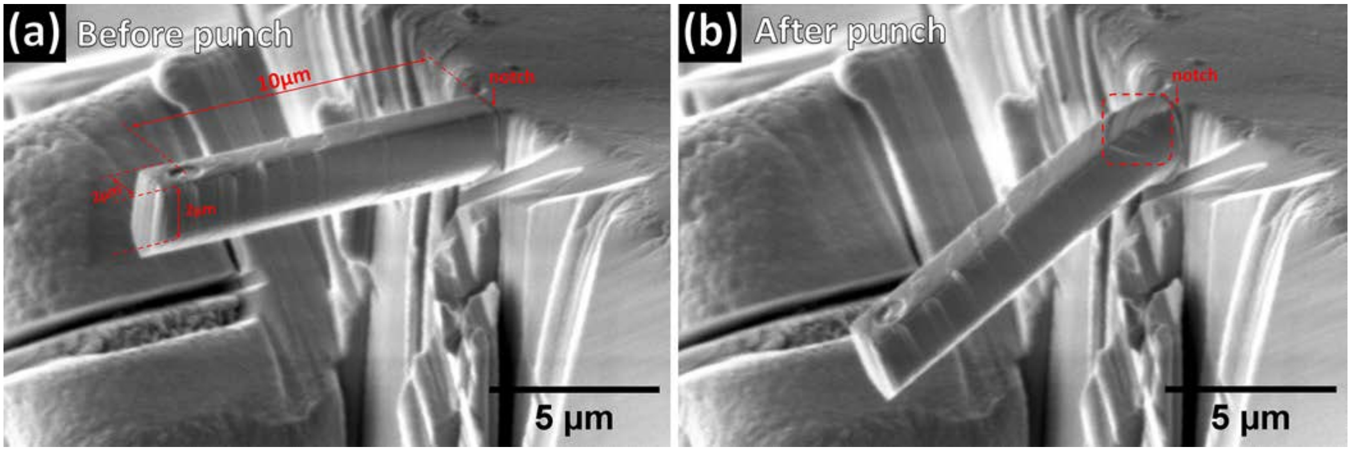



*Figure 2. Scanning transmission electron microscope cross-section images of N varied HT9 alloys all irradiated to 200 peak dpa using 3.5 MeV Fe ions at 450 °C with and without stress applied. Scale bar in (a) applies to others. (a) Low N HT9 shows voids after irradiation when no stress was applied. (b) Low N HT9 irradiated with stress applied shows suppressed void swelling and the inset TEM bright field image shows few voids appeared at 500 nm depth. (c) High N HT9 shows high swelling resistance after 200 dpa when irradiated without stress and did not show any voids after irradiated with stress (d).*




regardless of stress (Figures 2c and 2d). Detail characterizations are on the way to reveal more microstructure changes related to stress and the current result provides us important state-of-knowledge that low level tensile stress can suppress the void swelling. To estimate the DBTT before and after irradiation, a separate ion irradiation was conducted using 9 MeV Fe ions at 450 °C to provide more irradiated volume for micro-cantilever beam fabrication. Roughly 10 dpa was achieved within the depth where the beams were made. Approximately five cantilever beams were fabricated from both pristine low N and high N HT9 alloys. Each cantilever had an average width of 2 µm, a thickness of 2 µm, and a length of 10 µm as shown in Figure 3. During punch testing, the tip traveled at a speed of approximately 100 µm/s, corresponding to a strain rate of 50/s. At room temperature, the pristine sample beams exhibited plastic failure. The exact fracture toughness is being calculated. For the irradiated HT9 alloys, more than 10 individual cantilever beams were prepared. Punch tests were conducted at room temperature as well as at 100, 150, and 200 °C. After heating, the samples were cooled back to room temperature, and additional punch tests were performed. In all cases, the beams failed but did not fully separate from the base alloy. At lower temperatures, the failure showed a greater tendency toward brittle fracture. A detailed quantitative analysis of fracture toughness is still ongoing.

*Figure 3. Scanning electron microscope images of (a) micro-cantilever beam before punch and (b) after punch. Pristine HT9 samples exhibited plastic failure at room temperature. Irradiated HT9 samples showed similar behavior but greater tendency toward brittle fracture when tested at lower temperatures.*

# Post-irradiation Examination of Advanced Reactor Cladding Candidates

*Principal Investigator: Caleb P. Massey (Oak Ridge National Laboratory [ORNL])*
*Team Members/Collaborators: Annabelle Le Coq, Kory Linton, Jesse Werden, Stephen Taller, David Hoelzer (All ORNL)*

One potential obstacle in qualifying new cladding candidates for advanced reactors is the lack of knowledge about how differences in composition and processing variables affect post-irradiation degradation behavior. Using the High Flux Isotope Reactor (HFIR), various heats of advanced-reactor cladding candidates can be screened, achieving doses as high as 10 displacements per atom (dpa) per year. Without a functioning fast-spectrum test reactor in the United States, the Advanced Fuels Campaign (AFC) program has leveraged HFIR to screen multiple variants of fast-reactor cladding candidates in an irradiation campaign that will achieve final doses of ~75 dpa over ~7 years of irradiation.

## Project Description

In fiscal year (FY) 2019, ORNL fielded a HFIR irradiation titled the Advanced Manufacturing, Oxide-dispersion strengthened (ODS), and Wrought (AMOW) campaign. The AMOW irradiation campaign included capsules irradiated within the sodium-cooled fast reactor (SFR) temperature regime (~350-550°C) at doses spanning damage levels of 8 dpa and 75 dpa. Ultimately, the data generated from this irradiation campaign will inform permissible alloy impurity levels with respect to N content in HT9 ferritic/martensitic (F/M) steel; these data will also inform processing and optimization efforts for advanced ODS ferritic steels for next-generation fuel cladding.

Specific materials investigated in this work include Fe-12Cr F/M alloy HT9 with low (0.001 wt%) and high (0.044 wt%) nitrogen content, as well as three different ODS alloys with varying Cr contents and thermomechanical processing histories. The first ODS alloy, called the Oak Ridge Fast Reactor Advanced Cladding (OFRAC), is a 12Cr-ODS alloy with a dual dispersion of oxides and carbonitrides throughout the matrix. The second alloy, 14-Cr ODS 14YWT, is the leading ferritic ODS advanced cladding candidate for SFRs in the United States, developed

***HFIR provides a meaningful testbed for providing comparative irradiation data for various compositions and processing pathways for cladding performance and uncertainty quantification.***

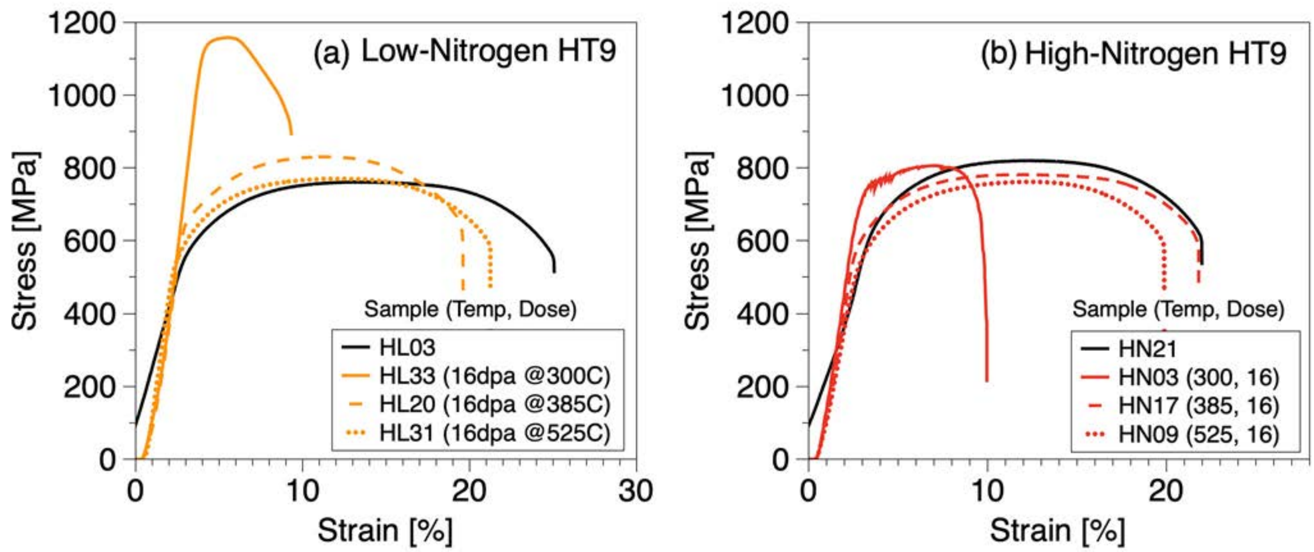



*Figure 1. Room-temperature tensile curves of (a) low-nitrogen HT9 and (b) high-nitrogen HT9 following irradiation in HFIR to 16 dpa at 300, 385, and 525°C target irradiation temperatures.*

by ORNL in the early 2000s. Finally, a 9Cr-ODS variant, 9YWTV, was also produced to provide a low-Cr and low-activation variant for comparison with the other reference alloys.

### Accomplishments

During FY24, mechanical tests were conducted on both the wrought HT9 materials and ODS FeCr materials following irradiation to 8 dpa (capsules AMOW01, 04, and 08) and 16 dpa (capsules AMOW02, 05, and 09) in the HFIR at target irradiation temperatures of 300°C, 385°C, and 525°C. Experimental irradiation temperatures for these capsules were evaluated using passive SiC thermometry specimens co-located within each capsule. Room temperature tensile tests of these specimens are summarized in Figures 1 and 2.

For the HT9 materials, although the nitrogen contents had only a minor impact on the pre-irradiation tensile response—namely, a higher yield strength (YS) and ultimate tensile strength (UTS) for the higher-nitrogen variant—the post-irradiation mechanical response shows significant differences in irradiation-induced degradation behavior, specifically at lower irradiation temperatures. An example of the tensile curves for the low-N and high-N HT9 variants, tested at room temperature, is shown

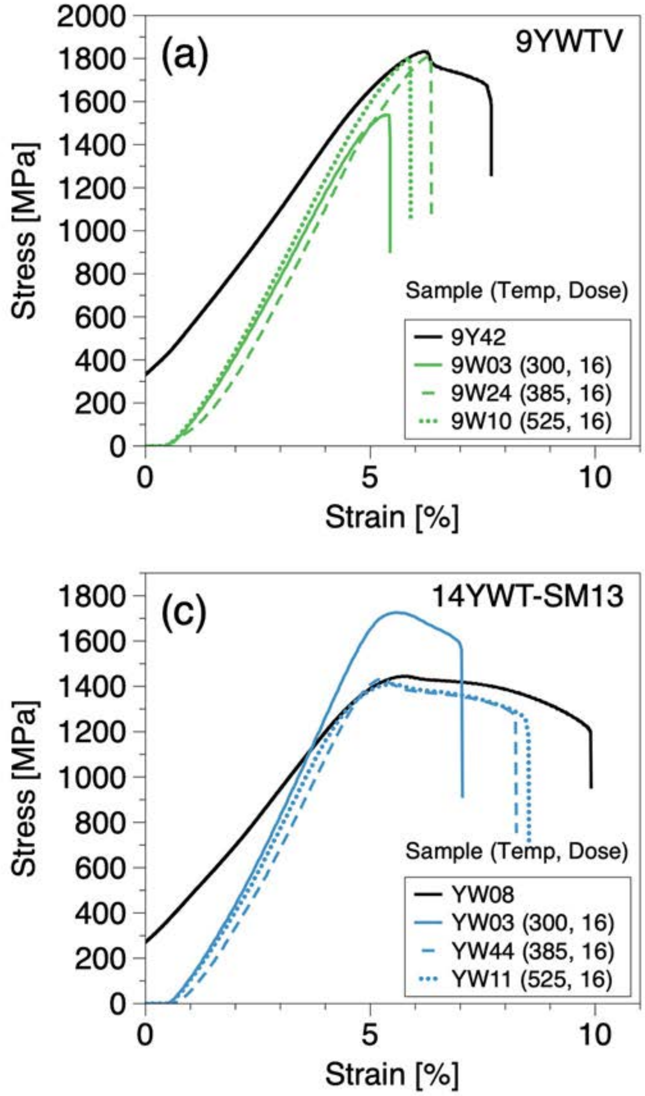



*Figure 2. Room-temperature tensile curves of (a) 9Cr-ODS alloy 9YWTV, (b) 12Cr-ODS alloy OFRAC, and (c) 14Cr-ODS alloy 14YWT following irradiation in HFIR to 16 dpa at 300, 385, and 525°C target irradiation temperatures.*

in Figure 1. For the low-N HT9, a marginal increase in YS and UTS can be seen at a target irradiation temperature of 385°C, representing the low-to-intermediate irradiation temperature for SFR operating conditions. This hardening is accelerated at the lower target irradiation temperature (300°C). Conversely, negligible effects on the strength and ductility are seen for the high-N HT9 variant irradiated at a target temperature of 385°C, and at a target temperature of 300°C the ductility reduces but the material retains greater strain hardening capacity in comparison to the low-N HT9 variant without the same drastic increase in YS and UTS. Additional microstructural examination of these specimens is needed to explain the vast differences in irradiation hardening behavior with the increased N content of the alloy.

The ODS alloys (in Figure 2) show very similar resistance to irradiation-induced changes to their mechanical properties, albeit with fundamentally different pre-irradiation tensile responses for each alloy. For the 9YWTV and 14YWT alloys, the materials before irradiation were already processed such that the alloys had extremely high YS (>1.8 GPa and >1.4 GPa, respectively, for 9YWTV and 14YWT), which also resulted in minimal strain hardening capacity when tested at room temperature before necking and failure. These trends continued following neutron irradiation but appeared to be more severe following irradiation. For the 9Cr alloy, no increase in hardness was observed at any of the irradiation temperatures, although the ductility was low in all test conditions. For the 14YWT alloy, following the lowest temperature irradiation condition, the YS did approach that of the 9Cr-ODS alloy, but some ductility remained. No significant change in strength or ductility was observed for the 14YWT alloy following irradiation at 385°C or 525°C target temperatures. Finally, the 12Cr-ODS alloy OFRAC provided the best balance of material properties following all irradiation conditions, although it is expected that this beneficial response was from the alloy's irradiation in the as-extruded condition rather than the work-hardened condition for the 9Cr-ODS and 14Cr-ODS alloys. This shows the importance of eventual irradiations of prototypic microstructures of components following the necessary shaping/forming processes, such as tube pilgering and annealing, for future accelerated fuel cladding qualification activities in AFC.

## Status Overview of FAST-01 Experiment

*Principal Investigator: Boone Beausoleil (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Chris Murdock, Luca Capriotti, Sobhan Patnaik, Andrew Bascom (All INL)*

***FAST has successfully accelerated the irradiation of several innovative metallic fuel designs from low to high burnups with initial PIE results indicating no major divergence from standard irradiation behavior.***

The Fission Accelerated Steady-state 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 completed non-destructive post irradiation examination (PIE) in 2022 and metallography and gas analysis in 2023. This year, sectioning and mounting for electron microscopy was completed along with scanning electron microscopy (SEM). The SEM results are used to provide details on composition and microstructure features within the fuel as well as fuel-cladding interface behavior. The data is useful to assess potential impacts from the accelerated irradiation and make comparisons to material behaviors in more conventional tests.

### Project Description

The Advanced Fuels Campaign (AFC) FAST experiments demonstrate an exceptional opportunity for advanced fuel research and development 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 separate effects). 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 FAST project functionally comprises four tests of uranium-zirconium (U-10Zr, 10 weight percent) metallic fuel: 1) control pins to correlate scaled testing behaviors to historical Experimental Breeder Reactor-II tests, 2) annular fuel geometries without bond sodium to investigate fuel-cladding chemical interactions (FCCI) and swelling behavior from low to high burnup, 3) the ability of fuel additives Sb, Sn, and Pd to arrest lanthanide mobility and mitigate the onset of lanthanide based FCCI, and 4) the use of Zr liners within the HT9 cladding to inhibit FCCI from low to high burnup. The tests are designed to demonstrate qualitative improvements in the metallic fuel designs and the ability to safely reach very high burnup targets (>20%FIMA). PIE activities have been underway on low burnup tests with moderate burnup tests currently undergoing disassembly for non-destructive examinations. Irradiations are expected to be complete for all tests during fiscal year 2025.

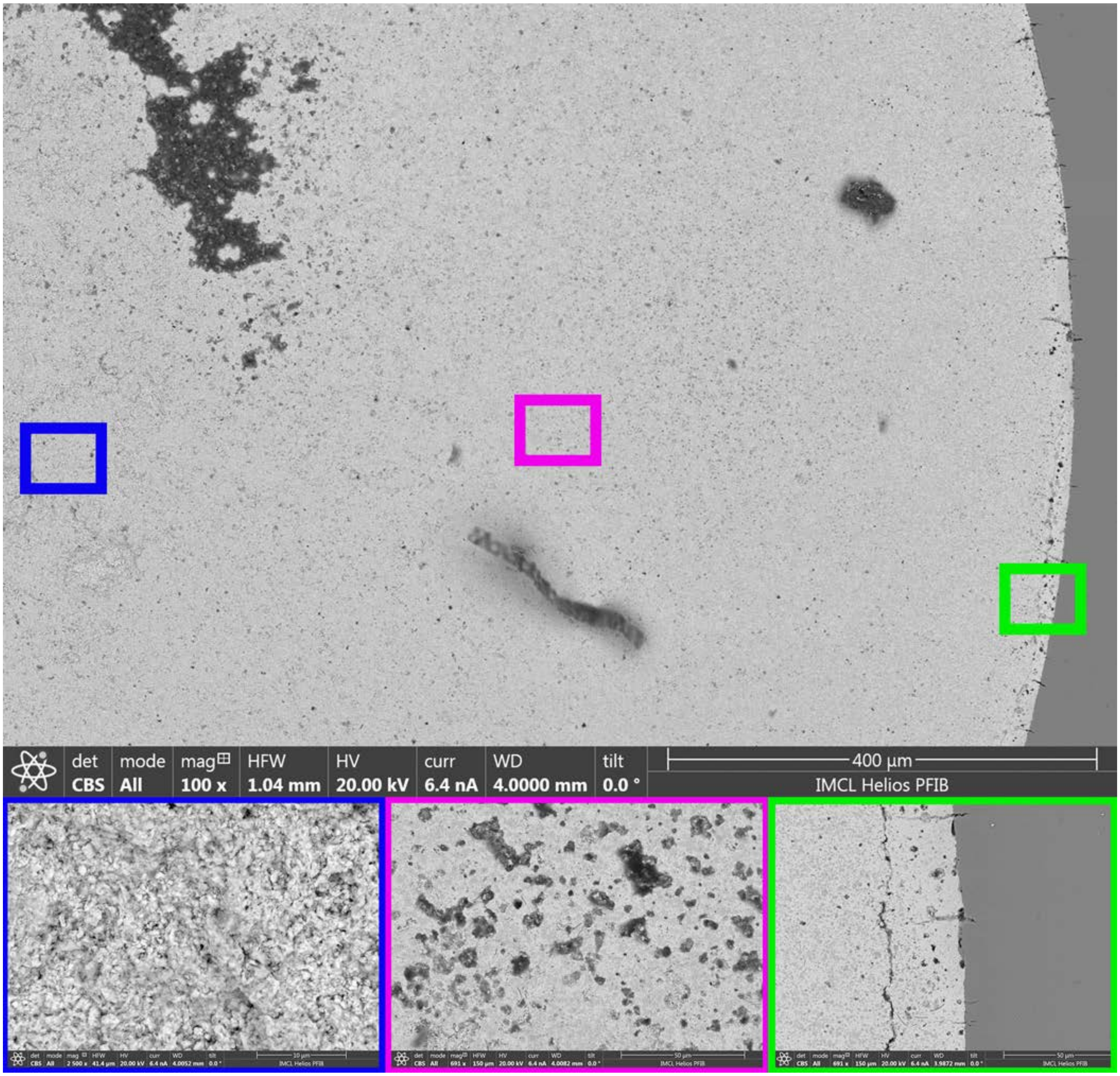



*Figure 1. SEM image of FAST-008 with emphasized images showing the center, mid-radius, and fuel-cladding interface. Each region shows the different stages of microstructure within the fuel and the change in porosity. The fuel-cladding interface shows the adherence of fuel to cladding but does not indicate any mass exchange indicative of FCCI problems.*

### Accomplishments

The FY24 technical goals of the FAST project were three-fold. The first goal was to maintain and continue irradiations within the ATR. The experiments were able to complete all planned irradiation cycles with minimal interruptions from ATR operations. Most of the tests have, at this point, completed their targeted irradiation objectives and have been removed from the ATR and are awaiting shipment later in FY25. A few high burnup tests remain in ATR and are scheduled to be removed in early summer of FY25. Once all tests are removed, they will all be shipped to the Hot-Fuels Examination Facility (HFEF) for PIE in the upcoming years. The second goal for the FAST project was to transfer samples from HFEF to the Irradiated Materials Characterization Laboratory for SEM characterization. This was completed for two samples: FAST-008, a low burnup control rodlet, and FAST-035, a low temperature rodlet. The SEM results for FAST-008 (Figure 1) showed interesting behaviors in how the rapid swelling of the fuel occurred and created distinct regions of porosity. It also showed interesting behavior along the fuel-cladding interface with the formation of several radial cracks. The SEM results for FAST-035 (Figure 2) showed that the fuel had very little changes to the microstructure during irradiation with only a few indications of nano-scale pores forming. Both rodlets were included in Nuclear Science User Facility rapid turnaround experiments to use transmission electron microscopy to investigate nano-scale microstructural features. The SEM work for FAST-007, a low burnup annular rodlet, is planned for fiscal year (FY)25. Lastly, the third goal of the project was to transfer the moderate burnup tests removed from ATR last FY to HFEF so that we could initiate the non-destructive examinations and disassembly of the capsules. The shipment of the rodlets, eleven rodlets in total, was completed in November of 2023. Visual examinations, precision gamma scanning, and neutron radiography of the capsules were completed by May of 2024. It was found from neutron radiography that one of the control rodlets failed and ruptured during irradiation. All capsules have since been disassembled and the rodlets cleaned and prepared for profilometry, gamma scanning, and gas analysis during FY25.

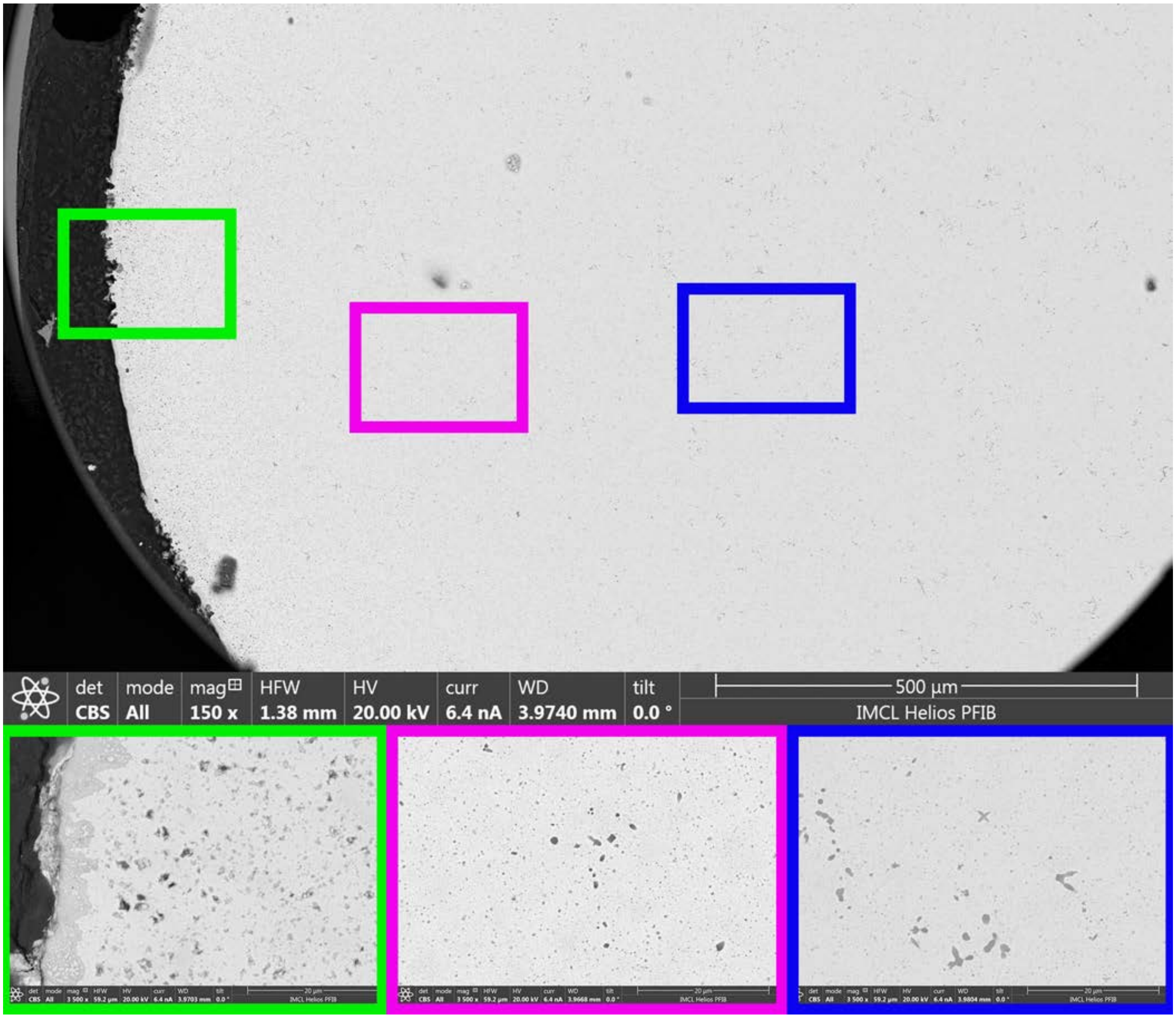



*Figure 2. SEM image of FAST-035 with emphasized regions showing the center, mid-radius, and edge of the fuel rodlet. Each region shows different variations of delta phase UZr2 and possible nano-scale porosity. The edge of the fuel rodlet shows the residual Zr rind typical of the as-cast condition of U-10Zr fuel.*

## HFIR Irradiations

*Principal Investigator: Annabelle Le Coq (Oak Ridge National Laboratory [ORNL])*
*Team Members/Collaborators: Denise Adorno Lopes, David Bryant, Nathan Capps, Jonathan Chappell, David Collins, Jacob Gorton, Jason Harp, Christopher Hobbs, Andrew Kercher, Kory Linton, Caleb Massey, Christian Petrie, Mackenzie Ridley, Nick Russell, Amber Telles (All ORNL)*

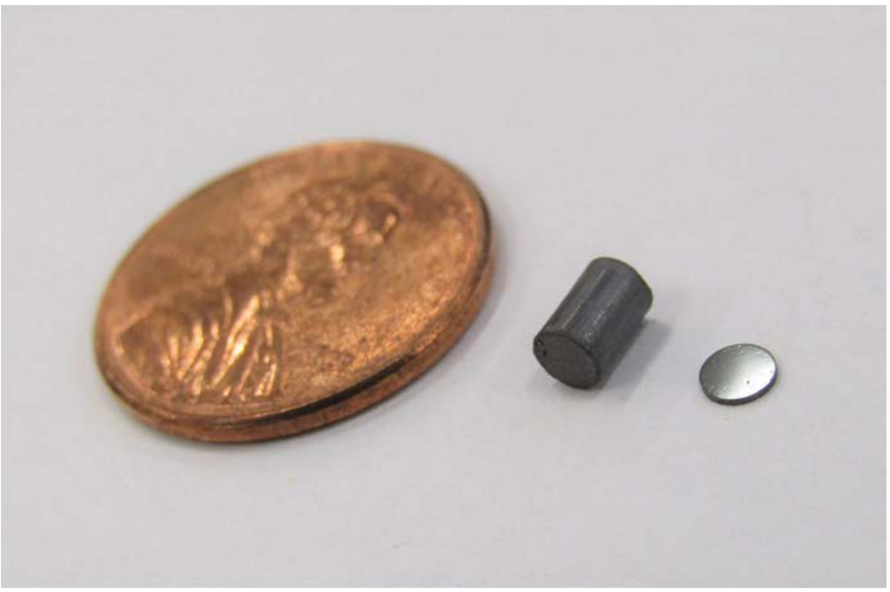


*Figure 1. Image of (from left to right) a penny for scale, a sintered fuel pellet, a MiniFuel disk specimen.*

Accelerated irradiation testing in the High Flux Isotope Reactor (HFIR) allows rapid collection of performance data on irradiated fuels and structural materials specimens. A HFIR fuel irradiation was performed to understand the microstructural evolution of $UO_2$ fuel at high burnup and isothermal conditions. Advanced cladding material concepts are also being studied for their potential application as accident-tolerant fuel cladding in nuclear systems and include FeCrAl alloys, alumina-forming austenitic (AFA) alloys, oxide dispersion strengthened (ODS) alloys, and chromium-coated Zr alloy.

### Project Description

A HFIR MiniFuel irradiation campaign has been developed to study the effects of temperature on the microstructure of $UO_2$ fuel specimens irradiated at high burnup. The goals of this experiment are to (1) investigate the burnup and temperature thresholds for dark zone formation in $UO_2$, (2) explore fission rate effects on microstructure formations, and (3) assess other fuel performance considerations such as fission gas release and unconstrained swelling under various irradiation conditions. This campaign will improve the understanding of the conditions driving the formation of microstructural features in high-burnup $UO_2$ fuel and mitigate safety concerns such as fuel fragmentation, relocation, and dispersal during a loss-of-coolant accident. Ultimately, the data generated from this irradiation will support high-burnup extension of $UO_2$ fuel in current commercial nuclear reactors. In addition, other MiniFuel experiment campaigns are studying the irradiated performance of advanced fuel concepts for potential application in advanced reactor systems.

Accident-tolerant fuel cladding material candidates have been studied using HFIR irradiation experiments. AFA alloys are examples of such candidates because of their potential higher resistance to embrittlement and high-temperature steam oxidation. HFIR irradiation of FeCrAl and AFA specimens aims to (1) study the impact of minor alloying elements on the neutron-irradiated mechanical properties of FeCrAl alloys and (2) collect neutron-irradiated mechanical properties on AFA alloys for comparison with those of FeCrAl alloys.

### Accomplishments

During fiscal year 2024, seven MiniFuel targets containing natural $UO_2$ specimens were successfully assembled and inserted into HFIR. The fuel specimens were fabricated at ORNL and pre-characterized to ensure that their density and grain size represented commercial $UO_2$ fuel pellets. Figure 1 shows a $UO_2$ MiniFuel specimen as well as a sintered $UO_2$ pellet and a penny for scale [1]. The irradiation conditions selected for this experiment include an average fuel irradiation temperature from 600°C to 1,000°C in 100°C increments and a specimen burnup of at least 50 MWd/kgU. These conditions are consistent with those observed in the dark zone region of commercially irradiated high-burnup fuel. Targets were inserted in the reactor reflector in removable beryllium (RB) and vertical experiment facility (VXF) positions in HFIR cycle 505 (February 2024) and cycle 507 (June 2024), respectively. Figure 2 shows the RB MiniFuel basket being inserted in HFIR [1]. These two irradiation locations differ by their thermal and fast flux, leading to a burnup accumulation rate more than two times faster in the RB compared to the VXF position. Ultimately, this difference will be used to determine the impact of the fission rate effects on the high-burnup structure formation. The targets will go through 7 and 16 irradiation cycles and are expected to complete irradiation by June 2025 and July 2027, respectively. During post-irradiation examination (PIE), fission gas release will be measured using ORNL's puncturing setup. Irradiation-induced swelling will be assessed on the recovered fuel specimens, followed by more detailed microstructure analysis. Additionally, seven MiniFuel targets containing advanced fuel specimens (Uranium Nitride (UN) kernels, UN TRISO particles, $U_3Si_2$ disks, compacts with UN and UCO TRISO particles in SiC matrix) irradiated up to 6% fissions per initial metal atom (FIMA) started PIE at ORNL's 3525 hot cell facility along with the first RB MiniFuel target containing $UO_2$ fuel specimens irradiated to 3% FIMA and 1100°C.

***HFIR irradiation testing supports the development of accident-tolerant fuel and material concepts for current and advanced nuclear systems.***

*Figure 2. Insertion the RB MiniFuel basket loaded with high-burnup targets into HFIR.*

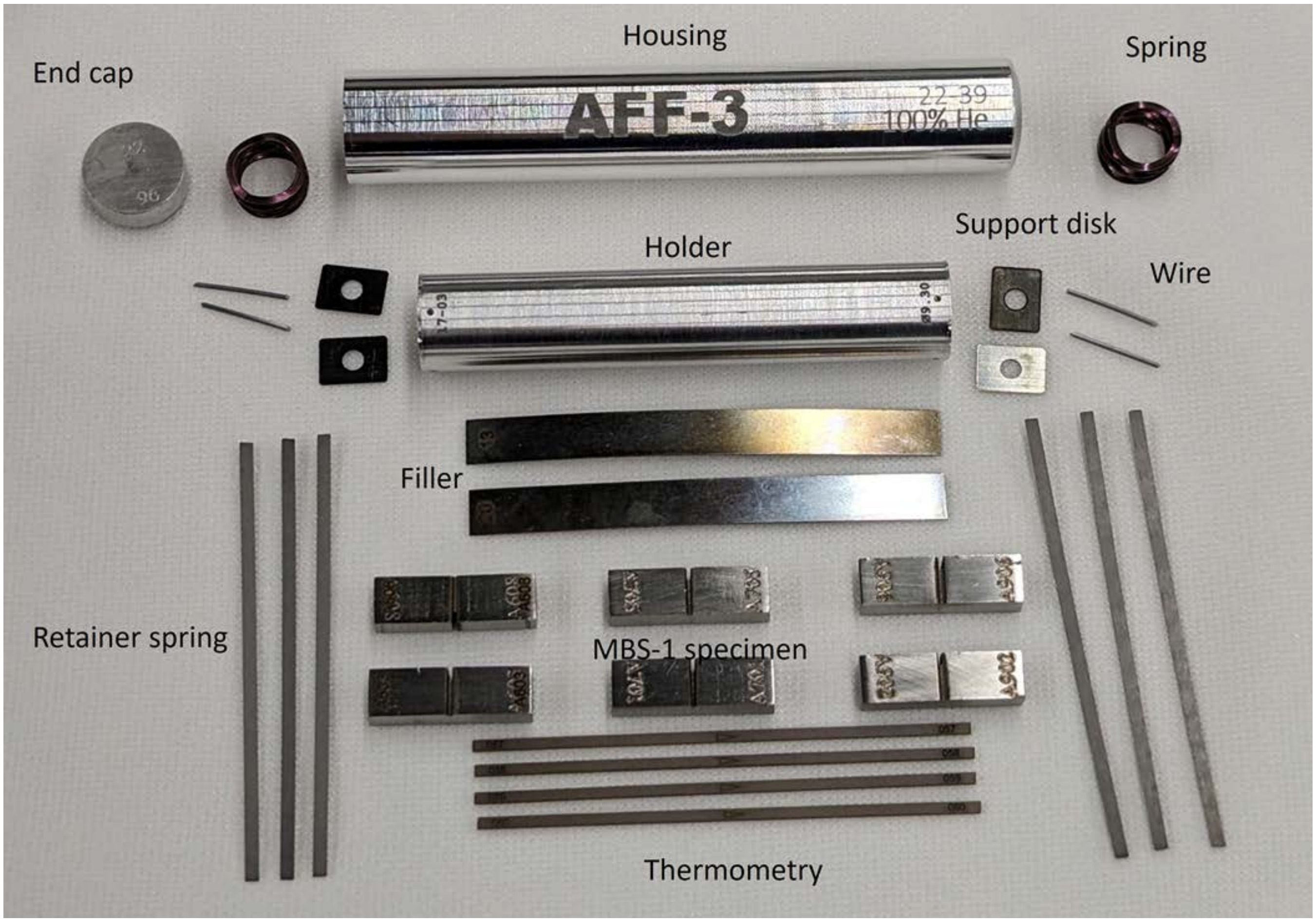



*Figure 3. Parts layout of a HFIR irradiation capsule containing MBS-1 specimens made of FeCrAl and AFA alloys.*

Six HFIR irradiation capsules containing specimens made of FeCrAl alloys with various minor alloying elements and AFA alloys were successfully assembled and inserted into HFIR's flux trap. Figure 3 shows an example of a capsule part layout [2]. Various alloys were fabricated at ORNL including wrought material, laser powder bed fusion processed material, and an alloy produced using hot-isostatic press. Each material was pre-characterized using scanning electron microscopy. The specimen geometries include small tensile specimens with a thickness of 0.5 mm (SS-J2 specimens) and miniature bend bar slotted-1 specimens (MBS-1 specimens). These two types of specimens are included in the existing general tensile and miniature bend bar HFIR capsule designs. The capsules target an irradiation damage of either 2 or 8 dpa and a specimen irradiation temperature of 315°C, a temperature representative of normal operating conditions in light water reactors. Mechanical properties were collected pre-irradiation to provide baseline performance data and included tensile testing and fracture toughness testing at room temperature. The capsules were inserted in HFIR in cycle 506 or 507 (April–June 2024). All capsules will complete irradiation by November 2024. PIE will include mechanical testing (tensile and fracture toughness) at room and irradiation temperature and microstructure analysis.

Additional accomplishments on cladding-material HFIR irradiations include the completion of the ODS tube irradiation campaign started under the campaign in 2019, making available highly valuable thin-walled advanced alloys tube specimens irradiated between 2 and 40 dpa. PIE on these specimens could include mechanical testing, oxidation testing, and microstructure analysis. Finally, 12 irradiation capsules containing coated Zr- and advanced Zr-alloy axial-tension tube and ring specimens were successfully assembled and inserted into HFIR. This irradiation campaign is the first to use the newly developed capsule design including pre-irradiation fabricated 3-mm-thick ring specimens, allowing a ring-pull test to be performed directly on irradiated ring specimens. The test matrix targets an irradiation temperature of 330°C and an irradiation damage between 4 and 10 dpa. Some of the low-irradiation-damage capsules completed irradiation, and the first ring specimen capsules were successfully disassembled in the ORNL 3025e hot cell facility. Figure 4 shows the first ring capsule fully disassembled along with all the irradiated specimens recovered.


**References:**

[1.] A.G. Le Coq et al., Design and Assembly of MiniFuel Targets for Characterization of High Burnup $UO_2$ Specimens Irradiated in the High Flux Isotope Reactor, ORNL/TM-2024/3396, Oak Ridge National Laboratory, Oak Ridge, TN, June 2024 (https://doi.org/10.2172/2394725).

[2.] A.G. Le Coq et al., Status Report on HFIR Irradiation of Optimized Alumina Forming Alloys, ORNL/TM-2024/3309, Oak Ridge National Laboratory, Oak Ridge, TN, March 2024 (https://doi.org/10.2172/2333784).

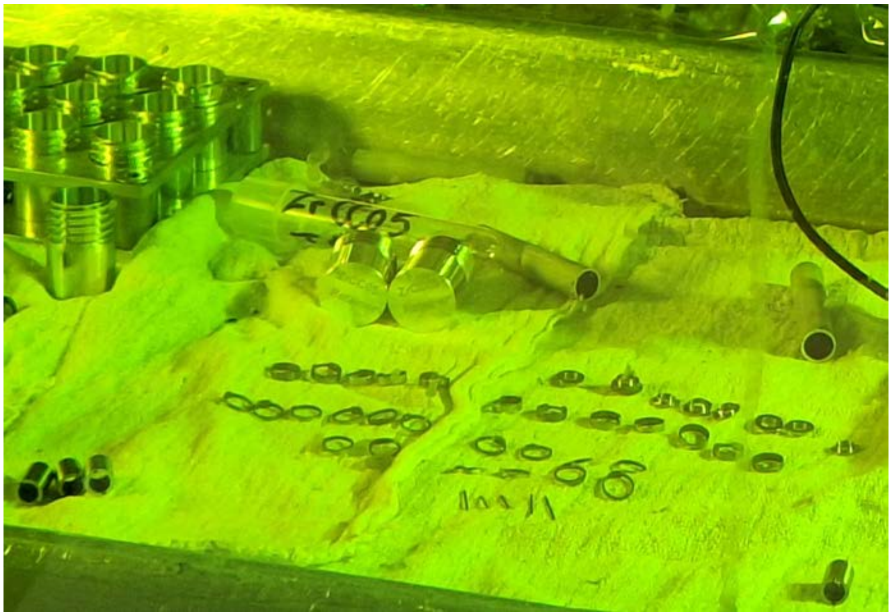



*Figure 4. First coated Zr-alloy ring specimen capsule disassembled at 3025e.*

# Fuel Cladding Chemical Interaction Characterization for U-10Zr Legacy Metallic Fuel: A Holistic Approach to Support Fuel Qualification

*Principal Investigator: Luca Capriotti (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Fidelma G. Di Lemma, Yachun Wang, Daniele Salvato, Kyle Paaren, Fei Xu, Yalei Tang, Jake Stockwell (All INL)*

Metallic fuel (based on U-10Zr) has regained attention as a promising nuclear fuel candidate for sodium-cooled fast reactors. Fuel/cladding chemical interaction (FCCI) has been recognized as a limiting phenomenon to deploy metallic fuel at high burnup and under harsh temperature conditions. While this phenomenon has been extensively studied, systematic and reliable data under several conditions of temperature, power, burnup are lacking. In order to acquire new data and to enable a better mechanistic understanding of FCCI, legacy fuel materials from MFF experiments irradiated in the Fast Flux Test Facility have been retrieved and analyzed with Scanning Electron Microscopy (SEM) and Small-Scale Mechanical Testing (SSMT). Furthermore, a novel machine learning method was developed, validated and used to quantify FCCI wastage thickness.

## Project Description

Advanced characterization techniques have been used in this study to effectively evaluate FCCI on a variety of MFF metallic fuel pins. Different combinations of temperature and burn-up were investigated, focusing on the effect of temperature on FCCI. To achieve a reliable analysis, FCCI wastage layers was evaluated via SEM which provides

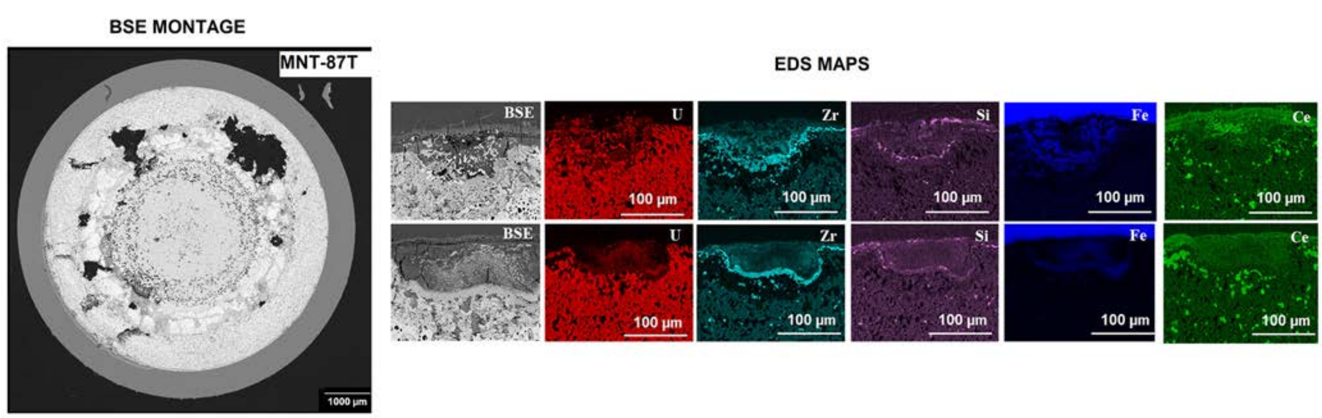



*Figure 1. Example of analyses for an MFF-5 high temperature pin sample (MNT-87T). On the left, in the full cross section backscattered electron (BSE) montage a clear FCCI layer is observed highlighted by red arrows. On the right is a high magnification BSE micrograph with energy dispersive x-ray spectroscopy maps of two FCCI regions. Both sets of maps show high Fe penetration and intensive lanthanide (Ln) attack including intergranular on the top set of micrographs.*

| MFF experiment | Axial Height (x/L) | Cladding wastage (μm) | Manual | | | | ML | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| // | // | Sample | Average | Standard Dev | Min | Max | Average | Standard Dev | Min | Max |
| MFF-2 | 0.50 | MNT-31G | 6.0 | 3.1 | 1.1 | 11.1 | 7.3 | 2.8 | 3.8 | 17.0 |
| MFF-2 | 0.63 | MNT-21G | 10.5 | 13.0 | 0.8 | 35.3 | 8.1 | 4.9 | 3.0 | 32.9 |
| MFF-2 | 0.95 | MNT-39G | 2.4 | 2.0 | 0.5 | 10.2 | 2.9 | 2.0 | 0.0 | 17.9 |
| MFF-3 | 0.44 | MNT-54C | 2.0 | 2.0 | 0.1 | 10 | N/A | N/A | N/A | N/A[1] |
| MFF-3 | 0.98 | MNT-83T | 117.4 | 56.8 | 1.3 | 186.8 | 109.1 | 66.5 | 1.5 | 210.2 |
| MFF-5 | 0.48 | MNT-86T | 16.0 | 9.5 | 4.9 | 35.1 | 16.5 | 5.4 | 4.9 | 38.9 |
| MFF-5 | 0.72 | MNT-87T | 18.7 | 26.5 | 0.4 | 88.6 | 16.8 | 21.8 | 0.3 | 81.1 |
| MFF-5 | 0.96 | MNT-88T | 22.8 | 14.8 | 2.3 | 50 | 25.4 | 12.7 | 2.5 | 61.3 |

*Table 1. FCCI cladding wastage values are reported for both methods (manual and machine learning). The homogenous lanthanide attack and the intergranular attack (if present) are both accounted as cladding wastage.*

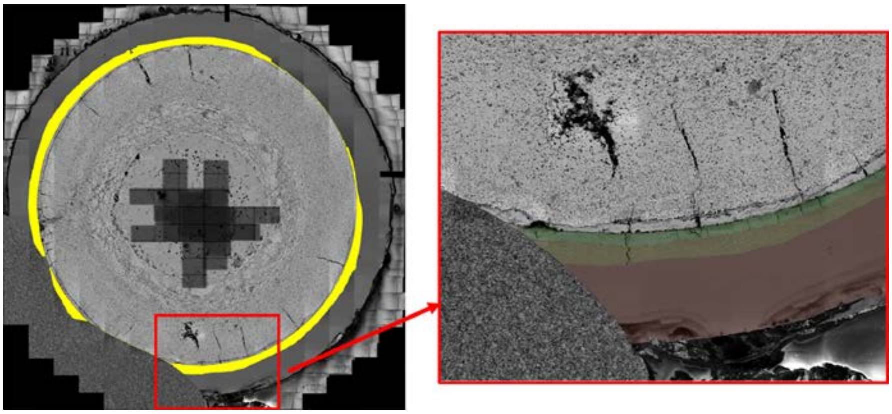


*Figure 2. Example of machine learning method applied to a full cross section. Automated stitching of high-resolution images was used in these analyses leveraging the use of the INL High-Performance Computing Platform. In the right image the FCCI (cladding wastage) is highlighted, in green the homogeneous Ln attack and in yellow the intragranular attack.*

high resolution microstructural and elemental information. A prescribed standardized procedure was used to ensure the data could be compared and processed. Moreover, a machine learning model was developed and applied to provide statistical relevance and avoid subjectivity in the measurement of FCCI thickness. The FCCI information were following correlates to local mechanical properties measured via SSMT for one examined sample. The SSMT method is not just suitable to examine the FCCI region formed in the cladding, it is as well applicable to investigate the potential mechanical property change throughout the entire cladding thickness without losing the fuel slug. This permits to assess not only the effect of each FCCI interaction layer on cladding mechanical properties, but also the irradiation effect on the performance of cladding matrix. This systematic data collection that connects the FCCI extension and microstructure

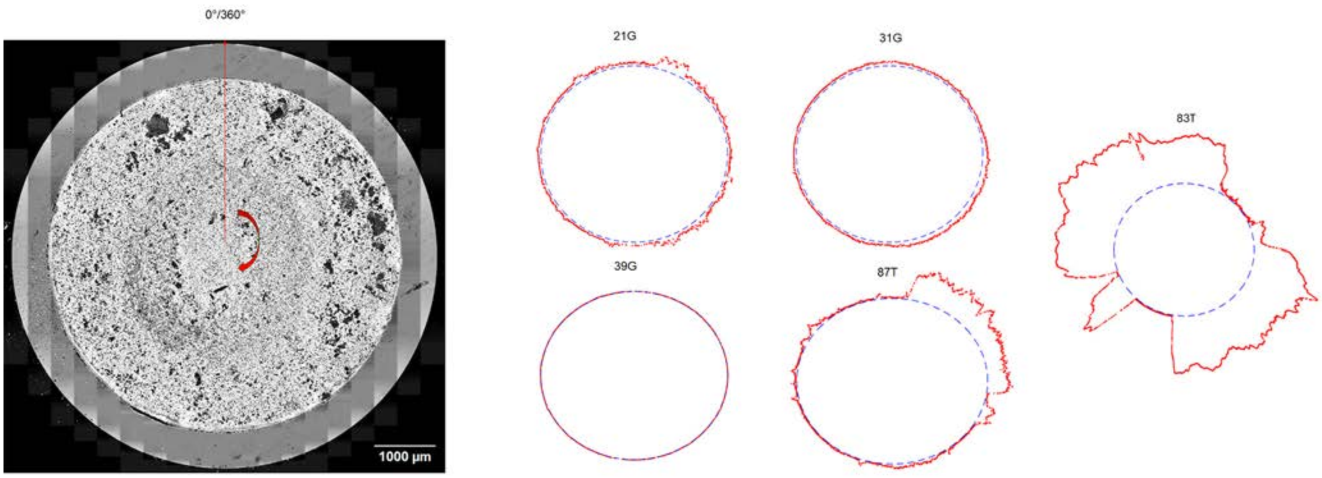



*Figure 3. These plots illustrate the thickness variation of the FCCI region for each fuel cross section from 0 to 360 ° as shown in micrograph on the left. Data from the machine learning have been normalized to 100 µm to permit to observe together samples presenting high FCCI (such as MNT-83T or MNT-87T) and samples with low FCCI (MNT-21G, 31G, 39G). The outer boundary of the detected FCCI region (red circle represents) is shown along with a schematization of the fuel region (blue dashed circle not to scale).*

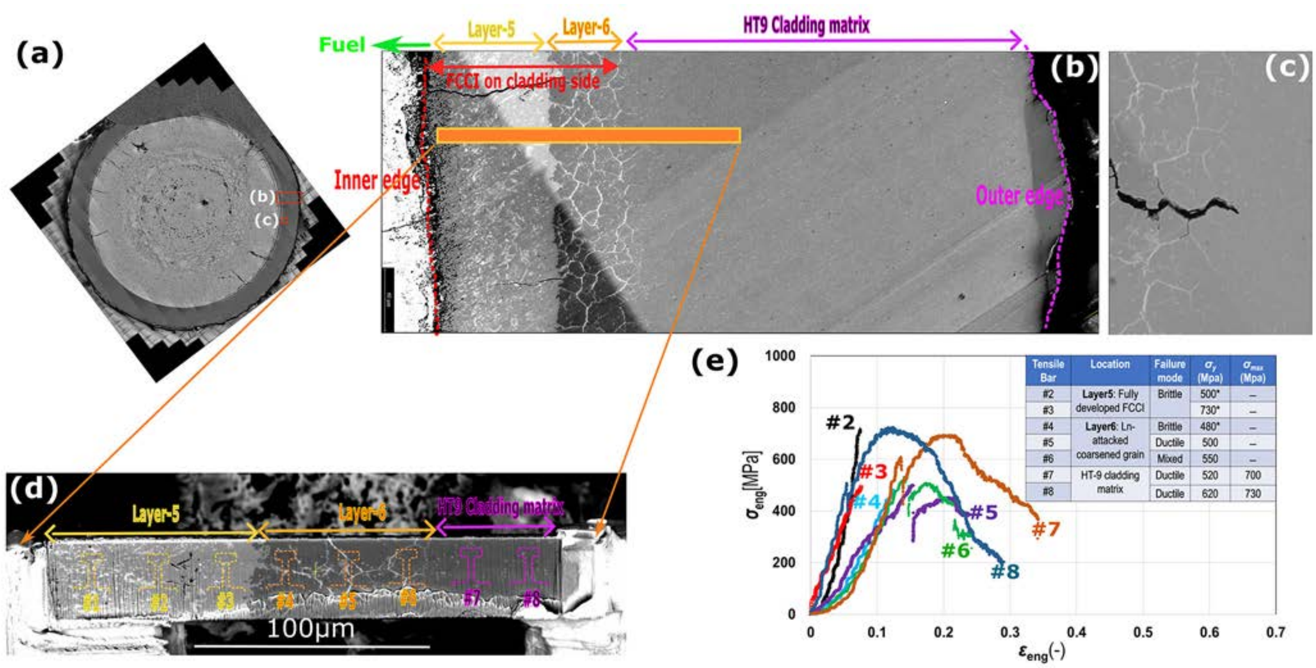



*Figure 4. SSMT results for MFF-3 sample MNT-83T (a), where two interaction layers (layer 5 and 6) formed on the cladding side (b). (c) A high magnification micrograph shows a crack propagated through layer 6 into HT9 cladding matrix. (d) shows the LALO and fabricated micro-tensile bars. (e)the SSMT results found that two tested specimens (#2-#3) in layer 5 failed in a complete brittle mode, three specimens (#4-#6) in layer 6 failed in a mixed mode, and the two HT9 cladding matrix specimens showed ductile behavior.*

with its mechanical properties provide important information for mesoscale models of FCCI effect on cladding behavior and its essential for the future qualification of high burn up metallic fuel.

## Accomplishments

SEM characterization was performed on four full fuel cross sections during this fiscal year from low temperature (MFF-2) and high temperature (MFF-5) pins, which were compared to data collected in previous campaigns, see Table 1 and Figure 1. The data collected via an optimized methodology provided the opportunity to develop and utilize machine learning models, as these necessitate reliable and homogeneous data input. The machine learning model was based on interactive segmentation model using a click-based and mask-guided interactive model and it was combined with Cascade-Forward Refinement and augmentation techniques (Figure 2). Its performance was evaluated against manual method and chemical elemental analyses data, obtained via Energy Dispersion Xray analyses. This model achieved a very high performance as indicated by a popular metric F1, which combines precision and recall, up to 0.99 (F1 can reach a maximum of 1). Moreover, machine learning can provide a local measurement of the FCCI wastage on the full cross section, which is time consuming if performed by an operator. A summary of the FCCI thickness results from the machine learning is shown in Table 1 and Figure 3.

Furthermore, to correlate the FCCI microstructural and elemental information to its mechanical properties, a large-area-lift-out (LALO) technique was used to extract selected FCCI area. Such LALO was used to fabricate multiple micro-tensile bars for in-situ mechanical testing. The force-displacement data was simultaneously recorded using the piezoelectric actuator in the capacitive transducer in each test for post analysis. The morphological evolution of tensile specimens was recorded as a video of SEM frames. The results obtained for MNT-83T (from MFF-3 experiment) are shown in Figure 4. Specifically, the homogenous lanthanide attack region (layer-50 has lost ductility and completely embrittled as a result of chemical interaction. In the intergranular attack (layer-6), the observed large, coarsened grains formed after carbon loss and the subsequent grain growth account for mechanical softening in the cladding. This is accompanied by lanthanides intergranular transport which weakened grain boundary cohesion, leading to intergranular cracking under stress during operation.

***Fuel cladding chemical interaction in U-10Zr metallic fuel: a novel approach FCCI wastage characterization and quantification that provides reliable and repeatable data combining SEM and SSMT to characterize the various wastage layers for microstructure and local mechanical properties. Furthermore, a novel machine learning method was developed, validated and used to quantify FCCI wastage.***

## Next Steps in Safety Testing Fast Reactor Fuels

*Principal Investigator: Colby Jensen (Idaho National Laboratory [INL])*

Metallic fuels for sodium fast reactors (SFR) are a strategic technology in the United States offering unique advantages to compared to other SFR fuel designs. The transient performance of metallic fuel is a key reason for its appeal as it inherently compatible with passive safety strategies while offering high reliability and performance. Mixed oxide (MOX) fuel designs are a well-proven fuel system worldwide, more than metallic fuels. Fuel performance during transient conditions is an important distinguishing feature of the two fuel systems, where a much more extensive database exists for MOX. However, a more limited database for metallic fuel validates its case for passive safety mechanisms in the SFR system. Uncertainties and optimization of the MOX system is driving high international interest in continued accident testing. A few remaining uncertainties are driving high domestic interest to complete a solid case for its qualification, especially to justify its role in passive mitigation of challenging reactor accident conditions. The Advanced Fuels Campaign (AFC) program is working towards establishing specific testbed capabilities and partnerships with France, Japan, and Korea, to support successful deployment of these most important fast reactor fuel types.

### Project Description

Following years of planning, the Advanced Reactor Experiments for Sodium fast reactor fuels (ARES) project started in 2020 as a collaboration between Japan Atomic Energy Agency (JAEA) and the Department of Energy (DOE; Figure 1). The project developed the Temperature Heatsink Overpower Response (THOR) capsule for the Transient Reactor Test Facility (TREAT) to study off normal conditions in a static sodium device, capable of holding full length Experimental Breeder Reactor (EBR)-II test pins. The project has progressed through testing multiple fresh fuel U-10Zr fuel pins and first tests on irradiated fuel pins, with near-term plans to finish testing advanced MOX and metal fuel pins in the next year.

***Safety testing of fast reactor fuels has become a significant and growing focus of international and domestic stakeholder collaborations with the AFC program.***

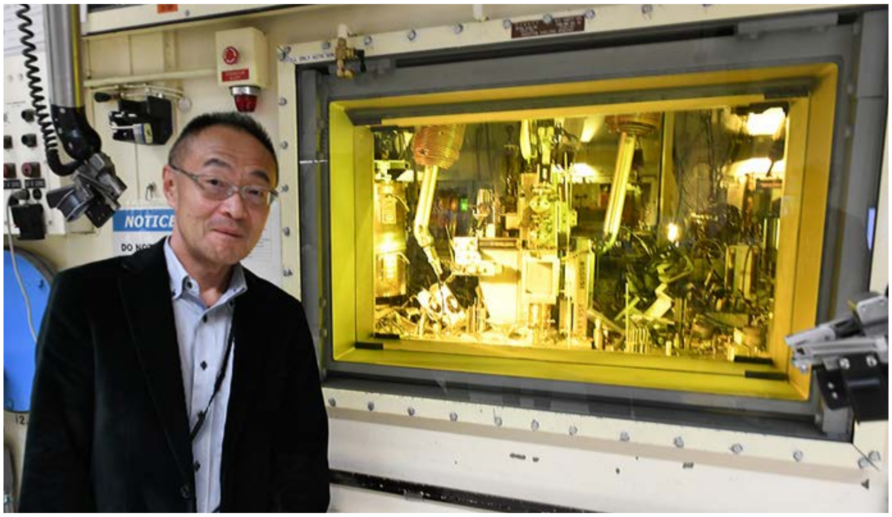



*Figure 1. Photo of JAEA collaborator at TREAT during Mixed Oxide Transient Over Power-1 experiment execution.*

Several years ago, INL and TerraPower won a project to develop the Mk-IIIR sodium loop for TREAT. The Mk-IIIR system is nearing design completion, along with supporting infrastructure across INL, and is planned for near-term use starting in fiscal year 2026. This latter work is primarily supported by significant investments from an Advanced Reactor Demonstration Program project. The AFC program is supporting the TREAT loop deployment through investment in an out-of-pile test loop facility for thermal-hydraulic and instrumentation studies. A transient, hot-cell furnace testing capability was identified a few years ago as an important capability gap, which can achieve conditions representative of important SFR transient conditions, at intermediate overtemperature and long-time scales.

During the recent year, the transient testing program for sodium reactor fuels advanced through continued maturation of testbed capabilities and input with stakeholders from industry and international collaborators on testing plans. These developments are filling important testing gaps that are crucial to DOE's goal of enabling deployment of advanced reactors.

Figure 2. SFR fuels testbed for three distinct time and temperature regimes.

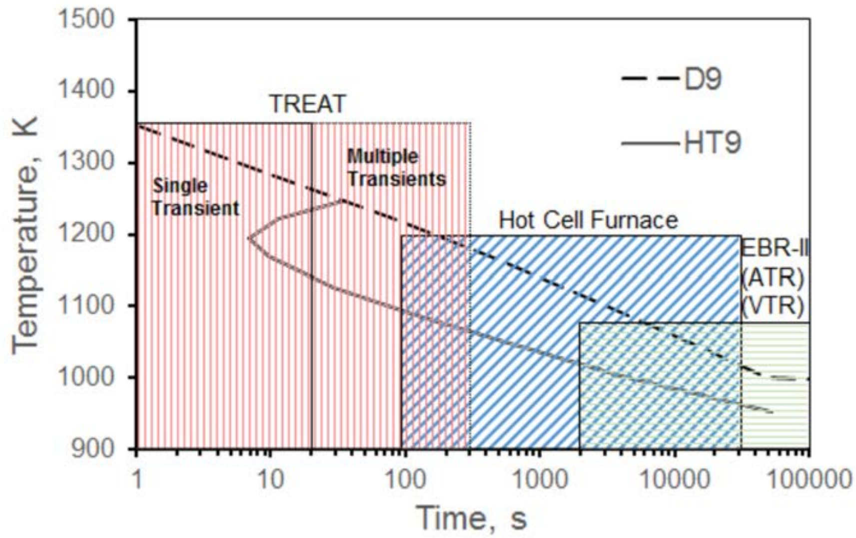
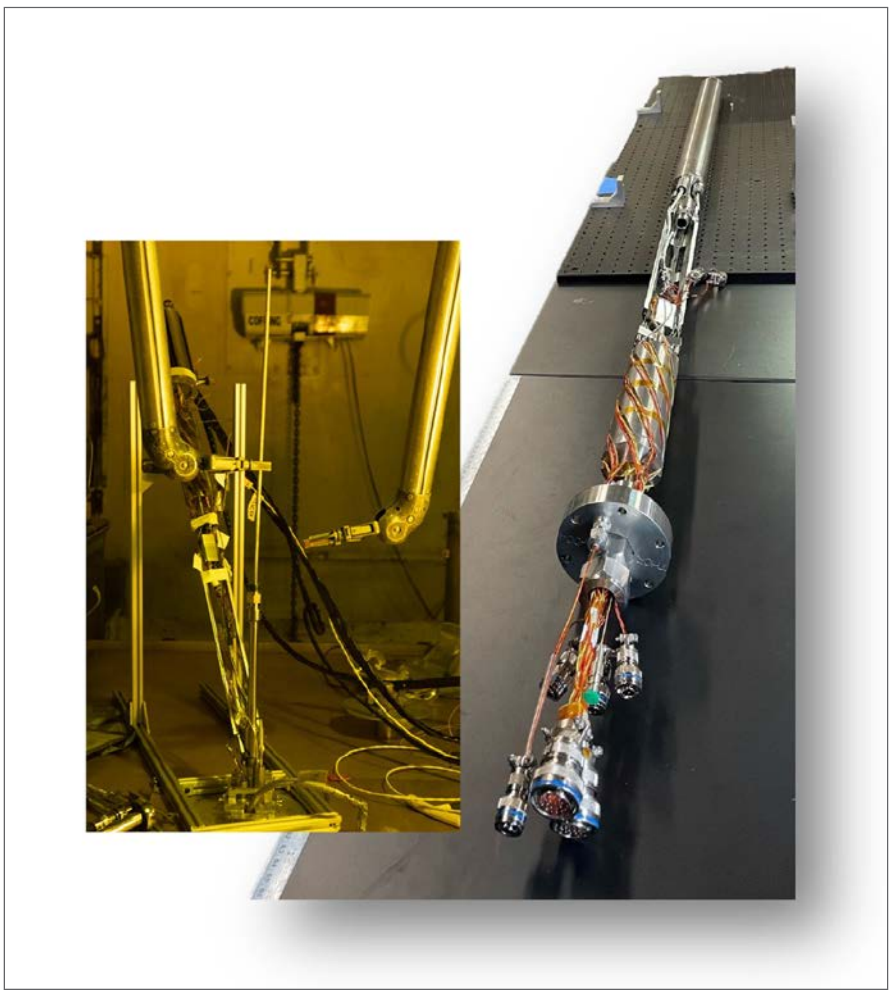



Figure 3. Photos of a recent THOR experiment assembly and loading an EBR-II pin in the hot cell.



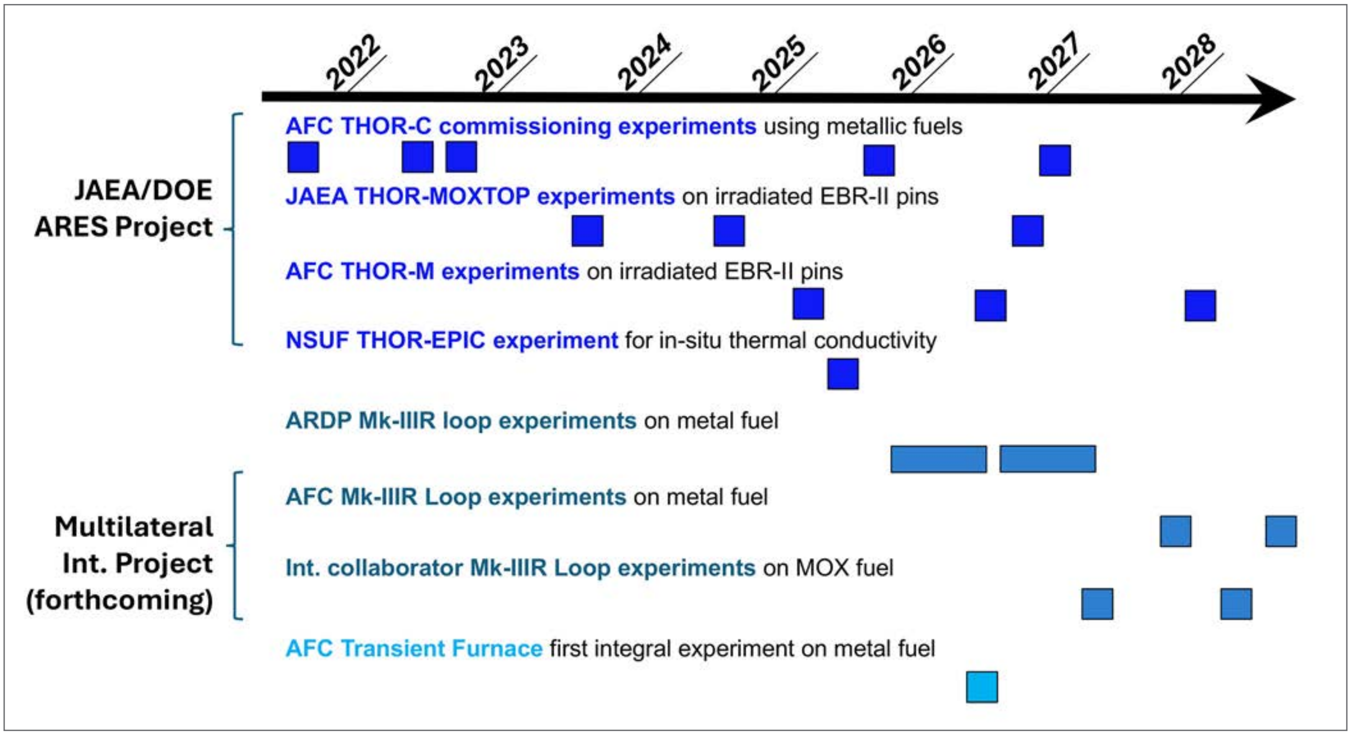



*Figure 4: Overview of recent, planned, and developing plans for transient testing fast reactor fuels.*

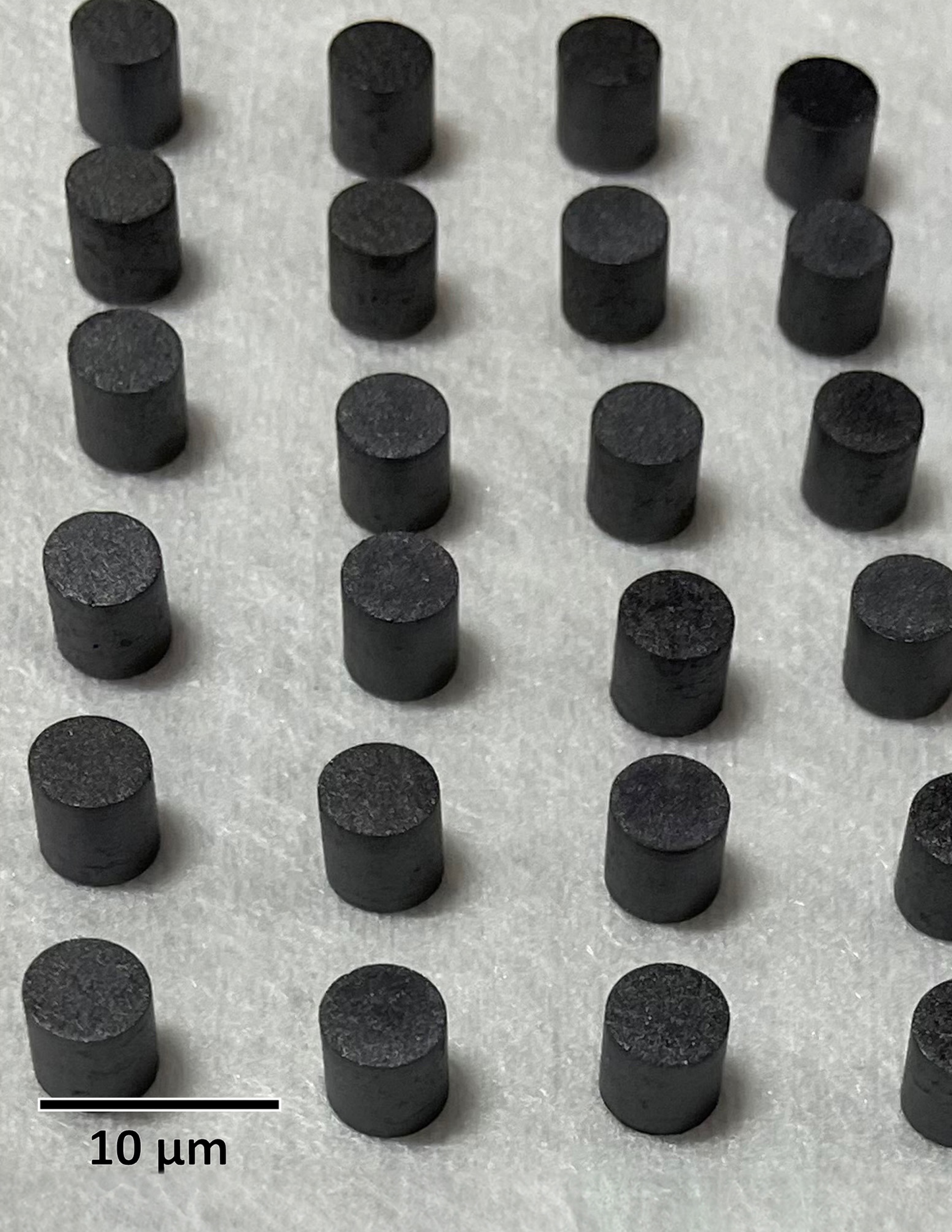
**Accomplishments**

The SFR transient testing testbed is composed of three main facilities including the THOR capsule and Mk-IIIR sodium loop in TREAT and a transient furnace testing capability in the hot cell. Figure 2 illustrates the time-temperature need for these capabilities. In recent stakeholder reviews of the AFC metallic fuels research plan, the furnace was identified as a key interest by some stakeholders, reinforcing its priority. The next steps in these areas are to redeploy THOR soon to perform multiple experiments on irradiated fuels in Big-BUSTER, for programs including JAEA, AFC, and the Nuclear Science User Facility. The sodium loop will continue towards completion of design and installation of all systems needed to support experiments. The transient furnace capability requires development of a detailed test plan to drive capability for testing, using existing or new furnace systems.

The data from THOR experiments completed to date is being used to develop fuel failure criteria for fresh fuel and is of high interest to industry and international institutions. Important results and hardware from recent THOR experiments (Figure 2 shows the THOR assembly and hot-cell loading). At the same time, the upcoming experiments, especially THOR-M-LOF-1, have become important points of stakeholder engagement towards ensuring the experiments achieve their broader needs. Multiple organizations are beginning to model THOR experiments to support their objectives.

The availability of the testbed is driving several test planning exercises between several stakeholders as bilateral and multilateral efforts. International interests have grown for testing both MOX and metallic fuels, building on the success of the ARES project. Irradiated heterogenous MOX fuel designs, now in INL fuel storage, are of high interest for study under overpower conditions, since modern MOX reactor designs are implementing such features and little transient performance data is available. TREAT is the only transient reactor in the world with easy access to relevant materials and supporting infrastructure to do such tests. Meanwhile, many international interests are peaking for conducting the first in-pile Loss-Of-Flow experiments on metallic fuels and AFC has partnered with the Fast Reactor Program to develop an experiment design to successfully meet U.S. goals for establishing criteria for passive safety. The MFF experiment materials stored at INL are crucial test subjects to explore the impacts of prototypic length fuel on current understanding of fuel degradation, failure, and post-failure consequences. Figure 4 provides an overview of fast reactor fuel safety testing program planning.

Finally, while completing testing on legacy MOX and metallic fuel designs is paramount for near-term deployment goals worldwide, testing advanced fuel designs under power-cooling mismatch conditions remains largely unexplored. Important fuel design variations considered for transient testing include recycled fuel compositions with potential impact of high TRU content, fuel forms removing in-pin sodium with potential for fuel-cladding mechanical interaction, and the effect of fuel-cladding interaction barriers on ultimate failure thresholds and post-failure consequences.


10 µm

# Assembly and Irradiation of the THOR-MOXTOP-2 Experiment

*Principal Investigator: Matthew Ramirez (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Colby Jensen, Trevor Smuin, Todd Birch, Chase Case, Ashley Lambson, Austin Fleming, Sarah Khan (All INL)*

The Advanced Reactor Experiments for Sodium Fast Reactor Fuels (ARES) project is a collaboration between the U.S. INL and the Japanese Atomic Energy Agency 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 Over-power Response (THOR) Mixed Oxide Transient Over Power2 (MOXTOP-2) capsule is an experiment in a static sodium capsule designed for irradiation in TREAT. The THOR capsule is sized to house a single EBR-II style fuel pin which is sodium bonded to an internal heat sink. The capsule includes two cable heaters (Figure 1) 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 2 shows an overview of the instrumentation leads through the top flange that are installed in the THOR-MOXTOP-2 capsule.

The THOR capsule is also 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 in the MOXTOP-1 experiment in 2023. In 2024, the THOR capsule design was updated to allow assembly into the new larger test vehicle

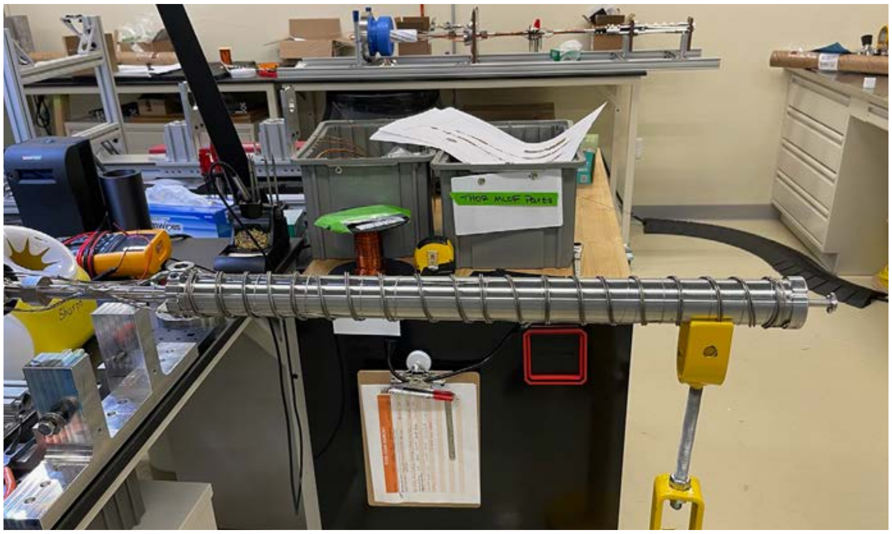


*Figure 1. THOR-MOXTOP-2 cable heaters.*

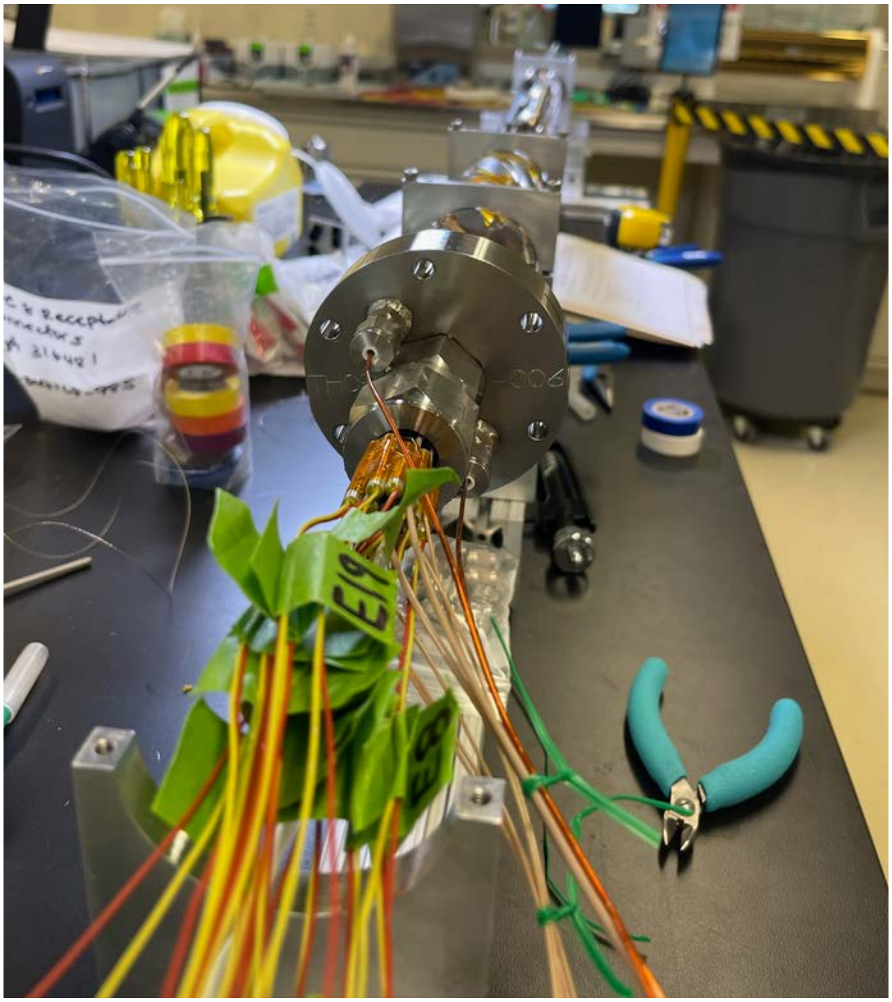



*Figure 2. THOR-MOXTOP-2 instrumentation.*

Figure 3. THOR-MOXTOP-2 sodium insertion.

in TREAT and with modifications for improved assembly and performance of the capsule. The first experiment assembly of the revised THOR design was for the THOR-MOXTOP-2 experiment.

### 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 pre-irradiated experiments in TREAT. Modules were assembled to support the first THOR-Metallic (THOR-M) and THOR-MOXTOP experiments.

The THOR-M experiment was the first THOR capsule to go through fuel loading and assembly inside HFEF. The THOR-M experiment was successfully loaded and assembled in March of 2023 and further proved the design of the remote handling equipment developed for assembly at HFEF. The THOR-M experiment provided expertise to HFEF operations and the THOR project team which has been leveraged for experiments such as THOR-MOXTOP-1 and now THOR-MOXTOP-2.

The THOR-MOXTOP-2 experiment module features the same hinged design as THOR-MOXTOP-1 which allows the experiment components above the capsule to rotate out of the way to facilitate sodium and fuel loading as can be seen in Figure 3. 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 Measurement Science Laboratory at the INL, where the instrumentation package was tested and calibrated.

Following the module and instrumentation assemblies, the experiment hardware was

> ***The completion of THOR-MOXTOP-2 experiments will help to further develop a high-burnup safety threshold for transient overpower conditions in advanced mixed oxide fuels for future international SFRs.***

shipped to the Materials & Fuels Complex for sodium loading in the pyrochemistry glovebox (Figure 3). In the inert atmosphere of the glovebox, shown in Figure 4, solid sodium was rolled into cylinders and loaded into the capsule. Using one of the two internal cable heaters (Figure 1), 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.

THOR-MOXTOP-2 had multiple design updates to mitigate the lessons learned from THOR-MOXTOP-1 and THOR-M. The fabrication and assembly of THOR-MOXTOP-2 were successful, and the project is currently awaiting the transfer to HFEF to load the fuel pin followed by transfer to TREAT for irradiation in late October/early November.

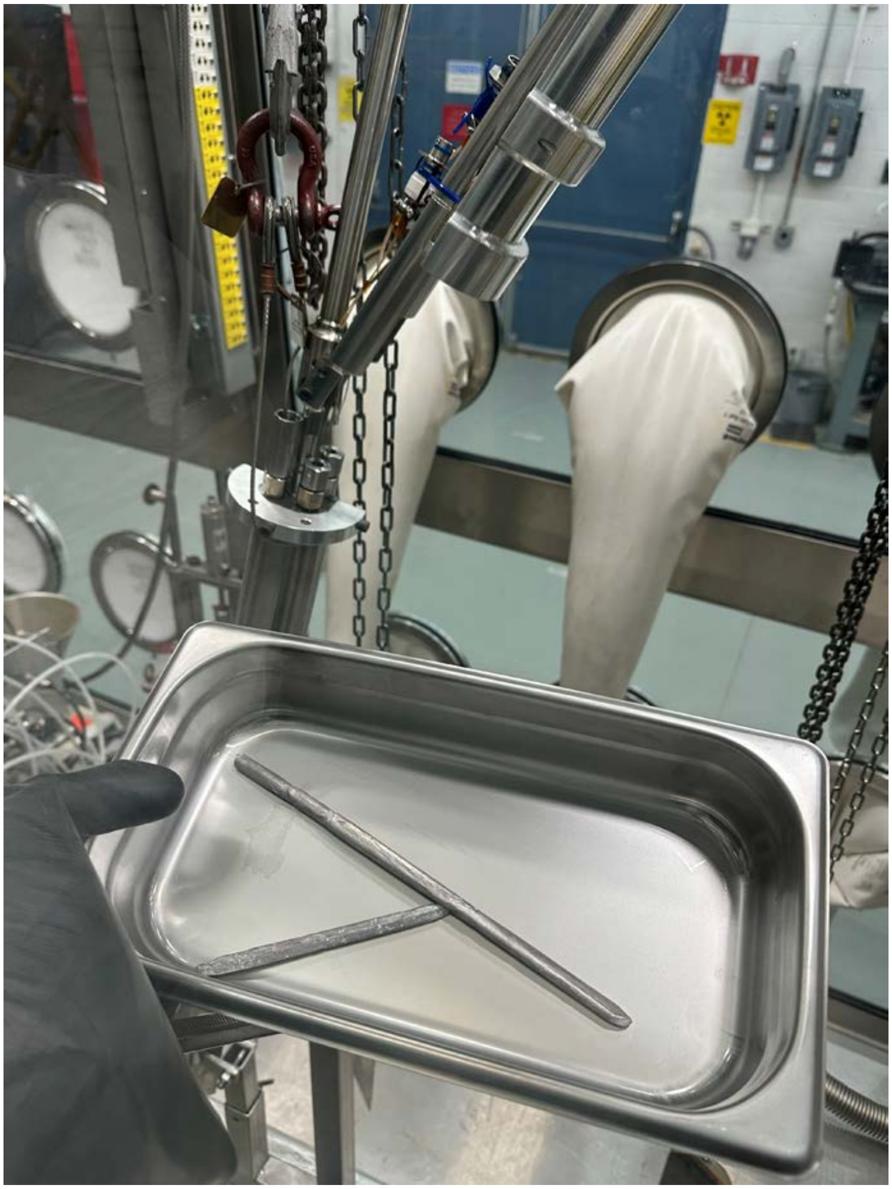


*Figure 4. THOR-MOXTOP-2 Sodium.*

# Design Updates for THOR Testing Device

*Principal Investigator: Matthew Ramirez (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Colby Jensen, Trevor Smuin, Todd Birch, Austin Fleming (All INL)*

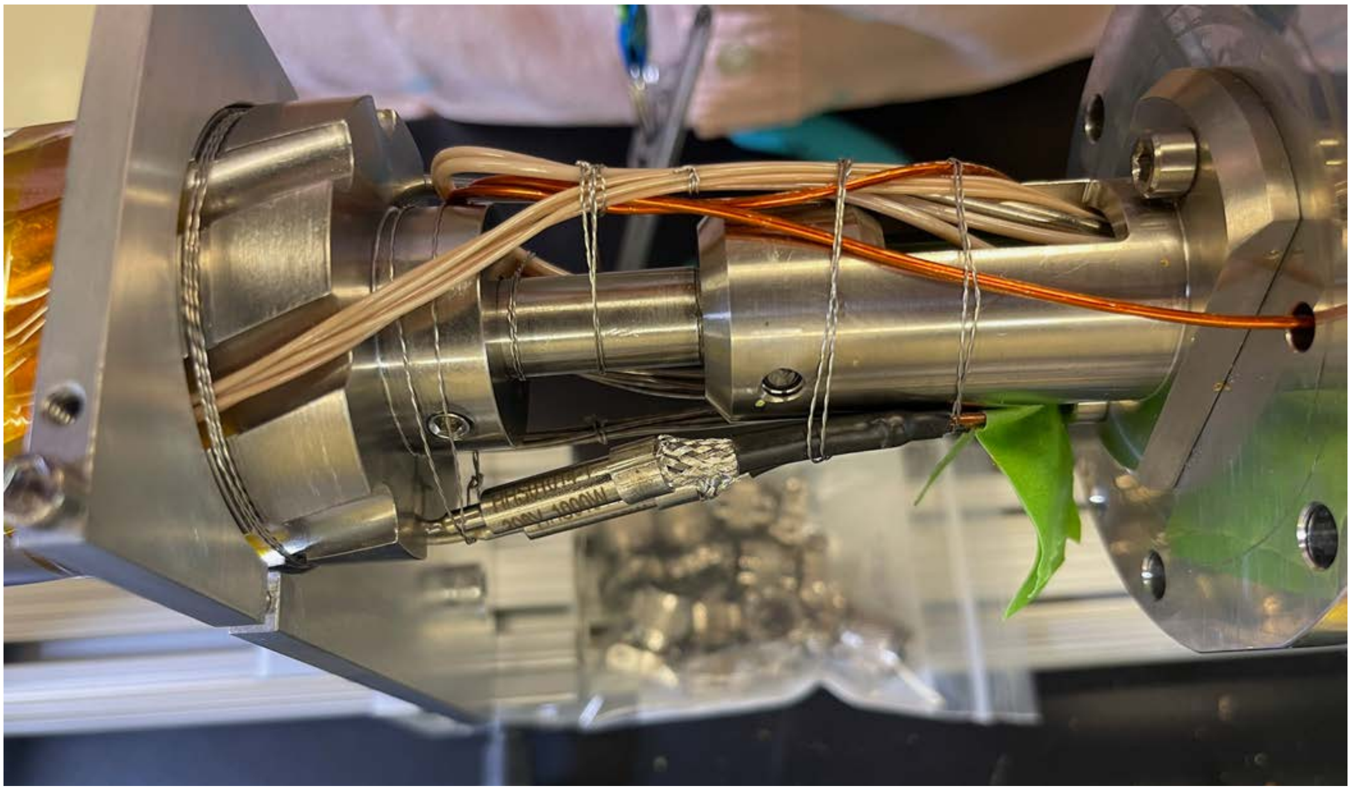


*Figure 1. Cable heater connections above tungsten shielding.*

The Advanced Reactor Experiments for Sodium Fast Reactor Fuels (ARES) project is a collaboration between INL and the Japanese Atomic Energy Agency 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) was 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 fuels to support continued improvement in fuel performance and economics. After completion of multiple capsules assembled through 2023, several design modifications have been made to streamline assembly and enhance performance of the capsule for many future tests already planned in the capsule.

## Project Description

The Temperature Heatsink Over-power Response (THOR) Mixed Oxide Transient Over Power (MOXTOP) capsule is a static sodium capsule designed for irradiation in TREAT. The capsule is sized to house a single full-length, pre-irradiated EBR-II style fuel pin which is sodium bonded to an internal heatsink.

The challenges encountered during execution of preirradiated experiments in 2023 and previous fresh fuel tests going back to 2021 produced many design updates from the lessons learned during assembly and installation as well adaptation to integrate into the new TREAT Big- Broad Use Specimen Transient Experiment Rig (BUSTER) test vehicle. These design updates were engineered into the system over the past year producing a much-improved 2nd generation THOR test device. During this year, the THOR-MOXTOP-2 capsule was refabricated and assembled to fully demonstrate the benefit of these design changes.

## Accomplishments

After the unexpected sodium leak from the heatsink inside the capsule for THOR-MOXTOP-2, it was decided that design changes would be required to successfully assembly, install, and irradiate THOR-MOXTOP-2 and subsequent THOR devices. The changes included: upgrades to the cable heaters, mitigation for sodium leaking out of the heatsinks, Conax sealing changes to decrease flange leak rate, cable changes for grounding issues, and mitigation of sodium loading blocking during loading.

**Cable heaters:** The THOR-MOXTOP-1 experiment was completed in 2023 as the final THOR experiment in the first-generation capsule tested in TREAT, also in the smaller TREAT test vehicle before TREAT upgraded to Big-BUSTER in fall of 2023. However, an issue with the cable heater in TREAT was identified before loading into the core, leading to the development and employment of an alternate heating approach using thermocouples.

There appeared to be two different failure mechanisms between two different cable heaters. One cable heater failed with loss of continuity and the other failed with one leg grounding out initially and the other leg grounding out later in the experiment. Testing was performed to better understand the failure mechanisms including cyclical and maximum temperature testing, coiling and bending, and radiation testing of cable heater components and materials in the Advanced Test Reactor gamma tube.

The cable heaters were tested using new cold lengths, lead lengths, and wire coating types. It was also found the original connections were made below the tungsten shielding increasing the potential of failure due to radiation damage. The cable heaters where reconfigured using a longer lead length to place the

*Figure 2. Single piece heat sink design.*

connection point above the tungsten shielding (Figure 1) and the wire type was changed from Teflon to Kapton as testing showed the Kapton coated wires sealed better than the Teflon. A redundant cable heater was also installed in case of a failure of the first cable heater.

**Sodium Leaking/Blockage:** The MOXTOP-1A experiment was successfully heated and executed, but it was found during the test and later confirmed that sodium had leaked from the heatsink to the bottom of the outer capsule shell. Around the same time, an unexpected sodium leak was also found in the THOR-MOXTOP-2 experiment during final assembly, before it was sent into the hot cell and halting further assembly. These leaks came after extensive development and prototype testing of the heatsink joint seals and led to the design being deemed flawed and requiring changes, including a new heatsink part. Meanwhile, the MOXTOP-2 test pin remained untouched in the Hot Fuel Examination Facility awaiting the updated hardware.

Thermal expansion of sodium during wetting/bonding on the THOR-M experiment caused the sodium level to rise and block the argon purge flow path, which caused the pin to be forced out of its loaded position after loading. This was addressed by adding vent holes to the well above the heatsink to provide an argon flow path above the sodium level in the heatsink. The sodium wetting/bonding temperature was also decreased, and the wetting time increased to ensure no sodium spilled or leaked out past the top of the heatsink. The counterbore in the heatsink was also increased to allow for thermal expansion of the sodium during the wetting/bonding process in the heatsink.

The original two-piece heatsink design allowed for small holes to be fabricated using an Electrical Discharge Machining hole popper along the length of each heatsink segment to install a distributed temperature sensor (DTS) fiber. To install the DTS fiber over the length of the entire heatsink, the heatsink needed to be a two-piece heatsink with a metallic knife edge with a crush gasket installed between the two mating surfaces. The crush gasket was tested with both titanium and iron heatsinks to ensure the sodium was sealed and would not leak.

The design was changed from a two-piece heatsink design to a one-piece heatsink design (Figure 2) with a gun drilled hole for all future irradiations. The THOR-wells (loading area for sodium, pin and linear variable differential transformer (LVDT)) was also redesigned to remove any potential ledges for areas of concern where sodium could be obstructed, as was encountered in some previous assembly instances. The DTS fiber is currently not included in upcoming THOR experiments but concepts for alternative installation methods are being considered for future builds.

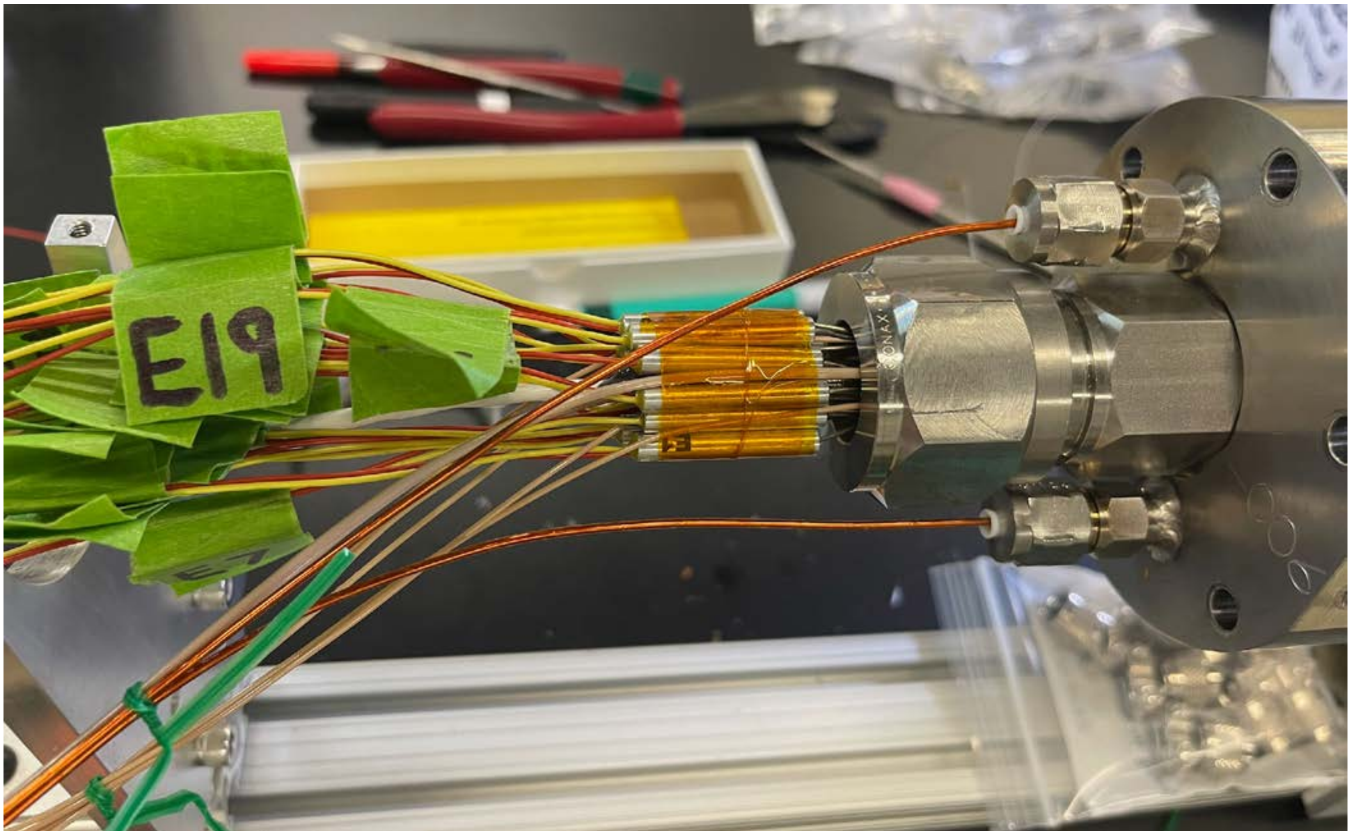



*Figure 3. Leads and connections in Conax for THOR-MOXTOP.*

**Conax Sealing:** The center compression fitting on the THOR-BUSTER flange was designed with 57 feedthroughs due to the high number of instruments in the experiment module. The compression fittings used are purchased from a vendor with Grafoil sealants. Achieving leak rates in the required 1E-6 to 1E-5 std cc/sec range with the compression fitting was difficult. Epoxy application on the compression fitting has been an effective method decreasing leak rates, but this method is labor intensive and time consuming.

More than two-dozen tests were completed with new sealants including fluoroelastomer (Viton) materials. These seals should seal better though they do have reduced temperature rating compared to the Grafoil. Test setup included application of leads or wires within the Conax, torquing of the Conax's, insertion into BUSTER primary containment pipe, pressurization using helium to 50 PSIG, holding pressure at 50 PSIG for at least 30 minutes, and lastly leak checking for each test configuration performed. Leak rates were gathered by helium leak detector probe method. The results indicated that the performance of the Viton sealant for configurations tested can deliver a greater sealing method than previously utilized graphite sealants. In addition, reducing the number of flow paths and the number of elements within the Conax resulted in lower observed leak rates. The Viton sealants will be used wherever temperature ratings do not demand the Grafoil type.

THOR-MOXTOP-2's design was modified to include a Viton sealant along with reduced numbers of feed throughs (from 57 to 30, see Figure 3) allowing for a much lower observed leak rate during testing after reassembly of the THOR-MOXTOP-2 capsule during this year.

**Cable changes for grounding issues:** Kapton (polyimide)-coated copper wires have been used for the LVDT and Acoustic Emitter sensor because the coating minimally increases the wire diameter. On the earlier assemblies these wires were found to have grounded in the BUSTER flange during torquing of the Conax compression fitting. To fix the issue on earlier assemblies, several thermocouples' leads were repurposed to replace the grounded wires.

THOR-MOXTOP-2 has replaced the Kapton coated copper wires with Polyether Ether Ketone (PEEK) insulated wires. During earlier tested it was noted that the PEEK insulated wires observed slightly lower leak rates when performing the tests. The Peek wires also provide a greater radiation tolerance than Teflon. The PEEK coated wires can be seen in Figure 4 just above the tungsten shielding.

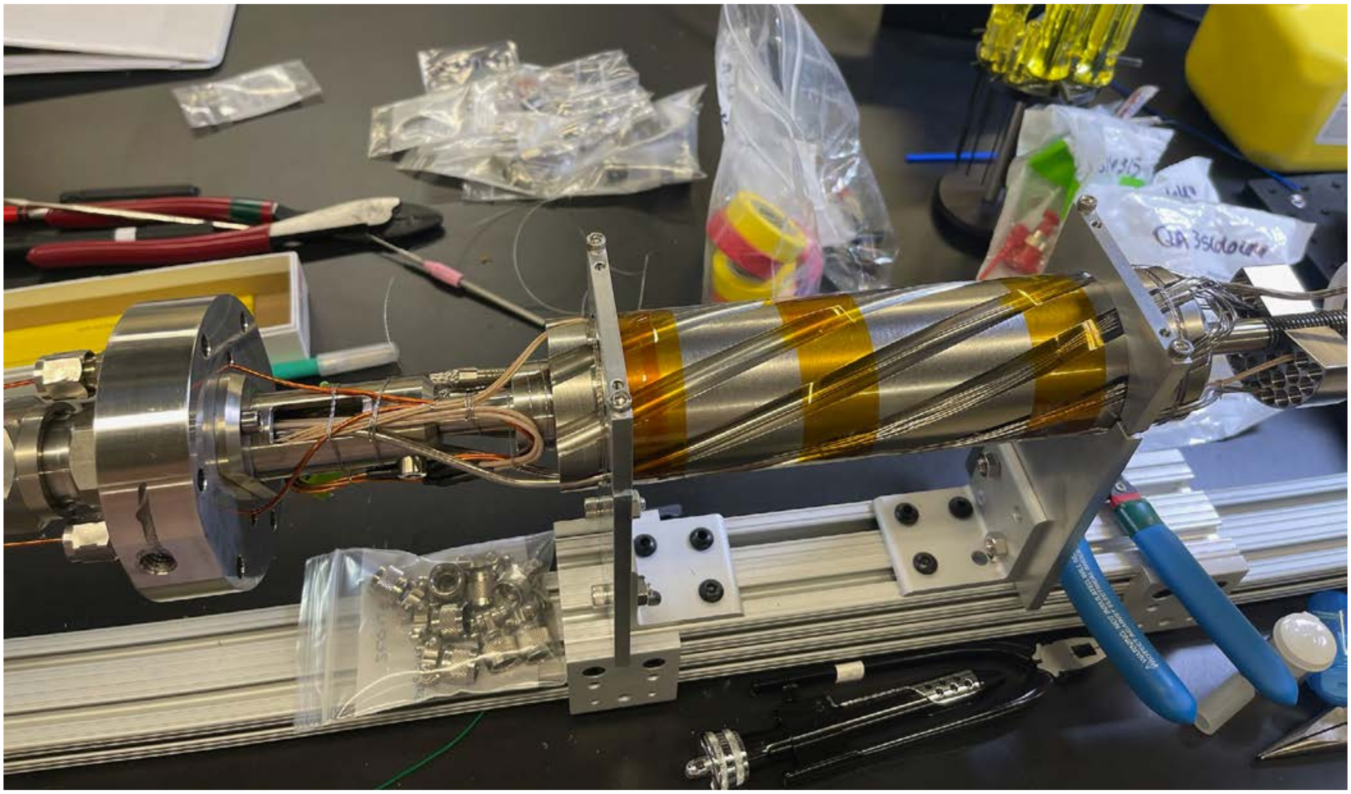


*Figure 4. Wire configuration changes for THOR.*

***The lessons-learned from full execution of several THOR devices from 2021 to 2023 resulted in several engineering refinements implemented during this year, first into a reassembled THOR-MOXTOP-2, greatly benefiting many future experiments.***

# Out-of-Pile Sodium Testing and Mk-IIIR Infrastructure

*Principal Investigator: Colby Jensen (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Mark Cole, Dan Crush, Aaron Feinauer, Jon Littlejohn, Brion Pearson, Cody Race, Emily Seegrist, Richard Skifton, Ian Stites (All INL)*

***The out-of-pile sodium loop and supporting infrastructure will provide researchers with the ability to perform experiments and instrumentation studies in a non-radiological facility that directly support the commercialization of SFR technology.***

Historically, Mk-series flowing sodium loops were used in the Transient Reactor Test Facility (TREAT) to provide prototypic sodium fast reactor (SFR) thermal-hydraulic conditions for transient testing of SFR fuels. Recently, there has been a resurgence in SFR research and fuels qualification to support commercial reactor designs. TerraPower and INL designed a new Mk-series sodium loop, Mk-IIIR, to support research and qualification of SFR fuels for emerging commercial applications such as TerraPower's Natrium design. The Mk-IIIR sodium loops provide a reactor-based test bed for SFR fuel research. The project also identified the need for Mk-IIIR sodium loop infrastructure to support sodium filling and system checkout in a non-nuclear facility and an out-of-pile sodium loop to support research and development of instrumentation, perform instrument calibrations, and validate the thermal hydraulic models used for Mk-IIIR experiments.

### Project Description

The project is currently working toward establishing equipment and infrastructure to support operation of both the Mk-IIIR sodium loops and the Modular Sodium Test Loop in the Idaho Engineering Demonstration Facility (IEDF). The IEDF provides a non-nuclear location to perform unirradiated experiments in support of SFR research. Establishing sodium operations and a sodium testbed in a non-radiological facility will result in significant cost and schedule savings relative to performing operations inside the TREAT or another radiological facility.

The MSTL originally operated at TerraPower's facility to support instrumentation development for the Mk-IIIR project. Following the project, the MSTL shipped to IEDF to support continued instrumentation research and Mk-IIIR test train out-of-pile thermal hydraulic studies. Unfortunately, the original MSTL design and equipment configuration did not support the temperature and pressure requirements for Mk-IIIR instrument calibrations and out-of-pile testing needs. Therefore, the project is upgrading the system to provide thermal-hydraulic conditions that more closely match the Mk-IIIR sodium loops for out-of-pile testing and instrument calibrations.

### Accomplishments

Due to the unique challenges of handling sodium, it was determined that Bay W-4 in the IEDF would be set up specifically to house the sodium loops and perform out-of-pile sodium experiments. In fiscal year (FY) 2024, modification to the W-4 mezzanine were completed to increase floor space for the Mk-IIIR and MSTL. Electrical panels and outlets were upgraded and routed where necessary to support the power requirements of the sodium loops. A fire enclosure will be used to house large experiments and sodium loops during molten sodium operations. In FY2024, a contract was placed with Sinisi Solutions to design and fabricate the fire enclosure. A glovebox will be used for smaller scale sodium operations and experiments. The sodium handling glovebox was received in FY2024. The project looks forward to installing the fire enclosure, glovebox, and initiating sodium operations in the IEDF in FY2025.

The MSTL preliminary design configuration is shown in Figure 1. The highlight of the system is the modular design which will allow researchers to insert, remove, and alter sections of the loop to support individual experiments. Modular sections of the loop include removable sodium pumps, removable in-flow test spools, flanged penetrations, and a Mk-IIIR specific test leg for full scale Mk-IIIR test train experiments. In FY2024, the project completed the preliminary design review of the MSTL. Long lead material procurements have started. The MSTL design will be finalized, and fabrication initiated in FY2025.

The first Mk-IIIR sodium loop is currently being fabricated at Petersen, Inc. Following fabrication, new Mk-IIIR sodium loops will arrive at the IEDF where the project plans to perform final assembly of the Mk-IIIR loops, load sodium, and perform initial system checkouts. Mk-IIIR sodium loop support equipment includes an assembly stand to complete assembly of the Mk-IIIR sodium loops, a stand to support the Mk-IIIR during operation, and a sodium fill/drain system. Both Mk-IIIR stand final designs were completed in FY2024 and fabrication initiated. The Mk-IIIR sodium drain/fill system design will be finalized, and fabrication initiated in FY2025.

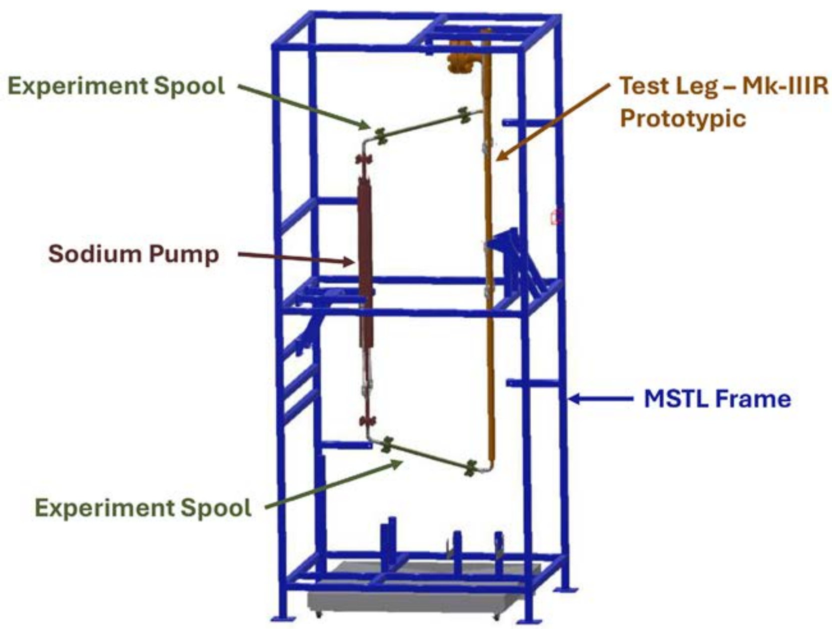



*Figure 1. The Modular Sodium Test Loop preliminary design configuration.*

## Preliminary BISON Fuel Performance Assessment for a Reference Metallic Fuel Design During Normal Reactor Operation

*Principal Investigator: Pavel Medvedev (Idaho National Laboratory [INL])*

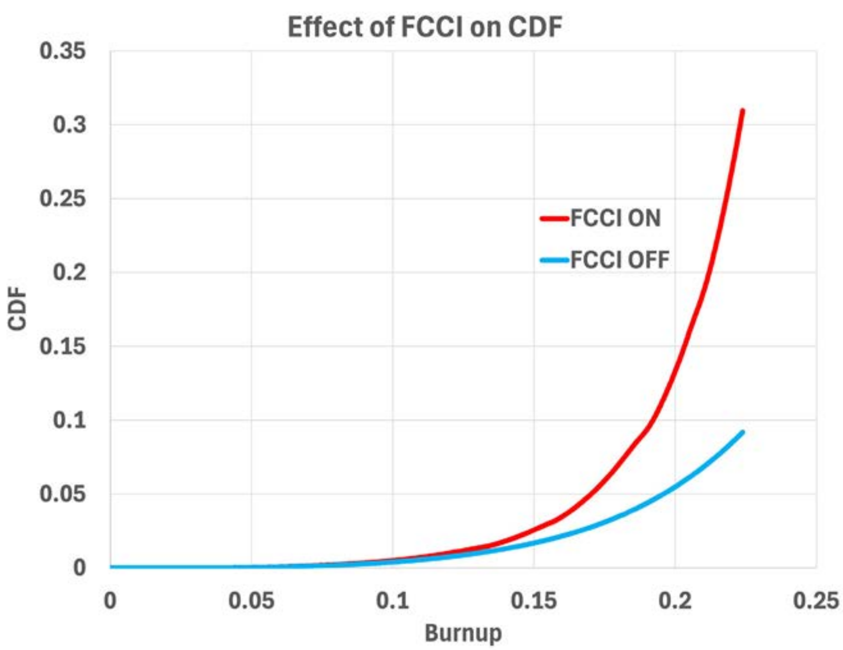



*Figure 1. Effect of FCCI on CDF.*

In 2023 the United States Nuclear Regulatory Commission (NRC) performed metallic fuel assessment using NRC NUREG-2246, "Fuel Qualification for Advanced Reactors" which was documented in NUREG/CR-7305. It was concluded that up to 10 at% burnup limit, life-limiting and safety-related fuel behaviors are well known and predictable. Fuel cladding chemical interaction (FCCI) was identified as the life-limiting phenomenon, which leads to cladding and barrier degradation.

It has become apparent that 10 at% burnup limit suggested by NUREG/CR-7305 does not meet performance expectations of the newly proposed advanced reactors. For example, Natrium reactor is designed to reach a burnup of 146 GWd/t or 15.2 at%. ARC-100 is designed to reach a burnup of 14 at%.

### Project Description

To address this issue steady state fuel performance calculations have been carried out using BISON fuel performance code for the reference fuel design and operating conditions. Performance results were compared to the Experimental Breeder Reactor (EBR)-II Mark-V fuel performance limits where appropriate.

### Accomplishments

The results obtained indicate Reference Fuel Design is compliant with EBR-2 Mark-V safety limits during steady state operation for peak burnup of 17% when the cumulative damage fraction (CDF) limit of 0.05 is reached. Recognizing that proposed advanced reactor will utilize high assay low enriched uranium fuel enriched to 20%, it is unlikely that their peak fuel burnup will exceed 15%. As shown in Figure 1 the reference fuel design will exhibit a CDF of 0.026 when it reaches 15% burnup. If FCCI is fully mitigated the CDF is reduced to 0.017. Both CDF values are indicative of extremely low fuel failure probability, and it may be concluded that during steady state operation to 15% burnup FCCI has no impact of fuel reliability.

*Figure 2. Pavel Medvedev is honored by INL director John Wagner for innovative nuclear fuel design patent.*

***Results indicate that metallic fuel performance much better than initially proposed by NUREG/CR-7305 and fuel failure probability is extremely low, while FCCI has no impact of fuel reliability during steady state operation to 15% burnup.***

BISON analysis indicates End of Life FCCI values of 172 micrometers at the upper region of the fuel pin for the peak fuel burnup of 22%. FCCI results in 3% reduction of the fuel burnup limit of the fuel, and successful mitigation of FCCI would allow burnup limit to increase to 20%. The study determined that FCCI is a localized event occurring in the cladding just below the plenum. Based on this observation an array of FCCI mitigation strategies is proposed including liners, dopants to inhibit Rare Earth Elements (RE) migration, adjusting axial power profile to reduce burnup at the fuel top, reducing cladding temperature at the fuel top, increase cladding thickness, decreasing smeared density, for recycled fuel using non-RE containing slug or for peak power assemblies use non-RE containing fuel, flipping assembly upside down to relocate FCCI prone area to colder region of the core, reversing coolant flow, cross flow/horizontal fuel assembly, spherical fuel elements, conductive inserts. Because FCCI is a localized phenomenon, FCCI mitigation strategies can be applied only to peak power pins and to the FCCI-prone areas of the pin, specifically fuel cladding just below the pin plenum.

# Joint Analysis of Metal Fuel Experiments in TREAT

*Principal Investigators: Aydin Karahan (Argonne National Laboratory [ANL]); Matthew Mihelish (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Colby Jensen, Pavel Medvedev (All INL)*

Ongoing and planned metallic fuel pin tests in Transient Reactor Test (TREAT) facility will address a critical gap required for the licensing of Sodium Fast Reactors. Since handling of the irradiated fuel pins and the transient tests are very expensive and time consuming, utilizing available simulation tools to guide and optimize the experiments is very crucial. TREAT capsule tests, in which the fuel pin is surrounded by stagnant sodium coolant and titanium heat sink and is encapsulated in an Inconel capsule, are planned as a practical and efficient approach to assess metallic fuel transient fuel behavior in TREAT facility. Temperature Heat Sink Overpower Response (THOR)-C-2 test, which tested fresh U-10Zr fuel with Experimental Breeder Reactor (EBR)-II fuel pin geometry, has been completed and utilized in this study for preliminary validation purposes.

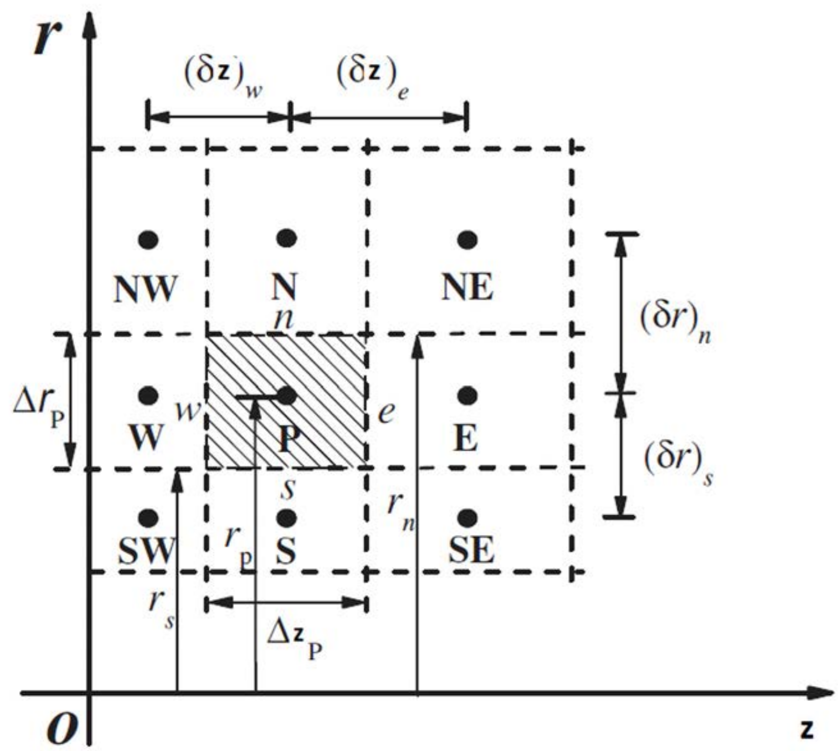



Figure 1. Schematics of a control volume used in finite volume discretization.

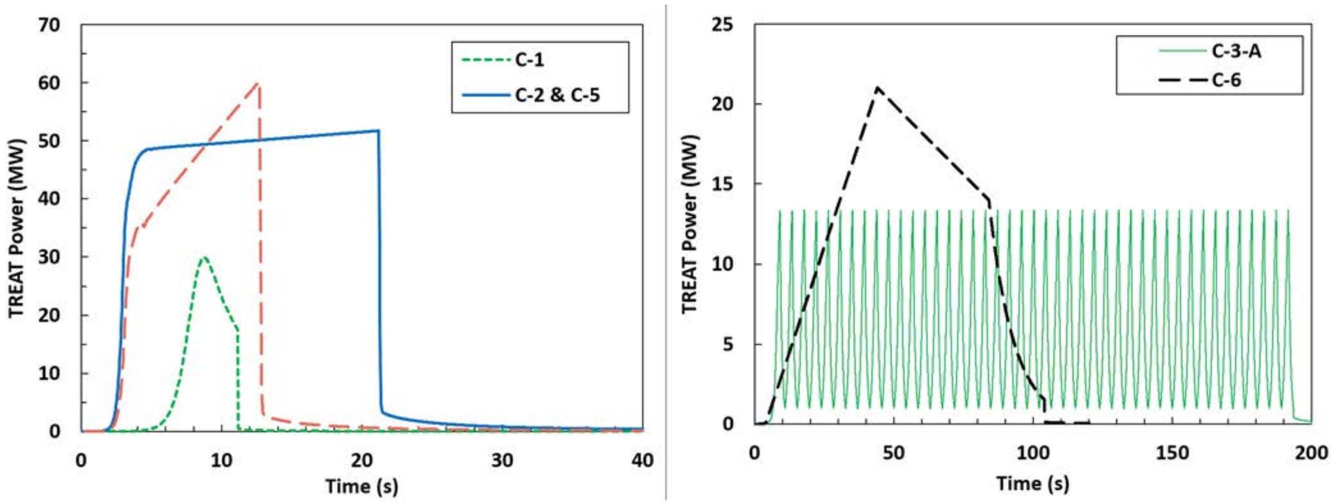



*Figure 2. TREAT power transients for all THOR-C experiments.*

### Project Description

MFUEL module of SAS4A/SASSYS-1 [1] is a physics-based metallic fuel performance model applicable to the normal operation, transient scenarios and fuel failure modeling including scenarios with bulk fuel melting. The model has been validated using EBR-II normal operation, separate effect transient tests as well as TREAT M-Series transient tests [2]. In this study, MFUEL models has been utilized together with a new capsule heat transfer model developed in this project. The new heat transfer model was necessary due to (1) significant amount of heat losses that required 2D heat transfer, (2) the presence of a titanium heat sink, rejecting a significant amount of heat, and (3) stagnant coolant conditions, which are inconsistent with SAS4A/SASSYS-1 (SAS) heat transfer model.

Updates to SAS4A/SASSYS-1 and MFUEL has been described below, followed by a preliminary validation effort using the results from THOR-C-2 fresh fuel capsule experiment. A previous study for THOR-C-2 analysis using BISON code is also utilized in this study to model this test [3].

### Accomplishments

#### Development of a Capsule Heat Transfer Model

A 2D time dependent finite volume model, developed in [4] for solving cylindrical heat conduction problems, has been utilized to compute the 2D transient temperature distribution of the fuel pin, stagnant sodium, and titanium heat sink, including the gas plenum. Figure 1 illustrates an axisymmetric control volume used in finite volume discretization.

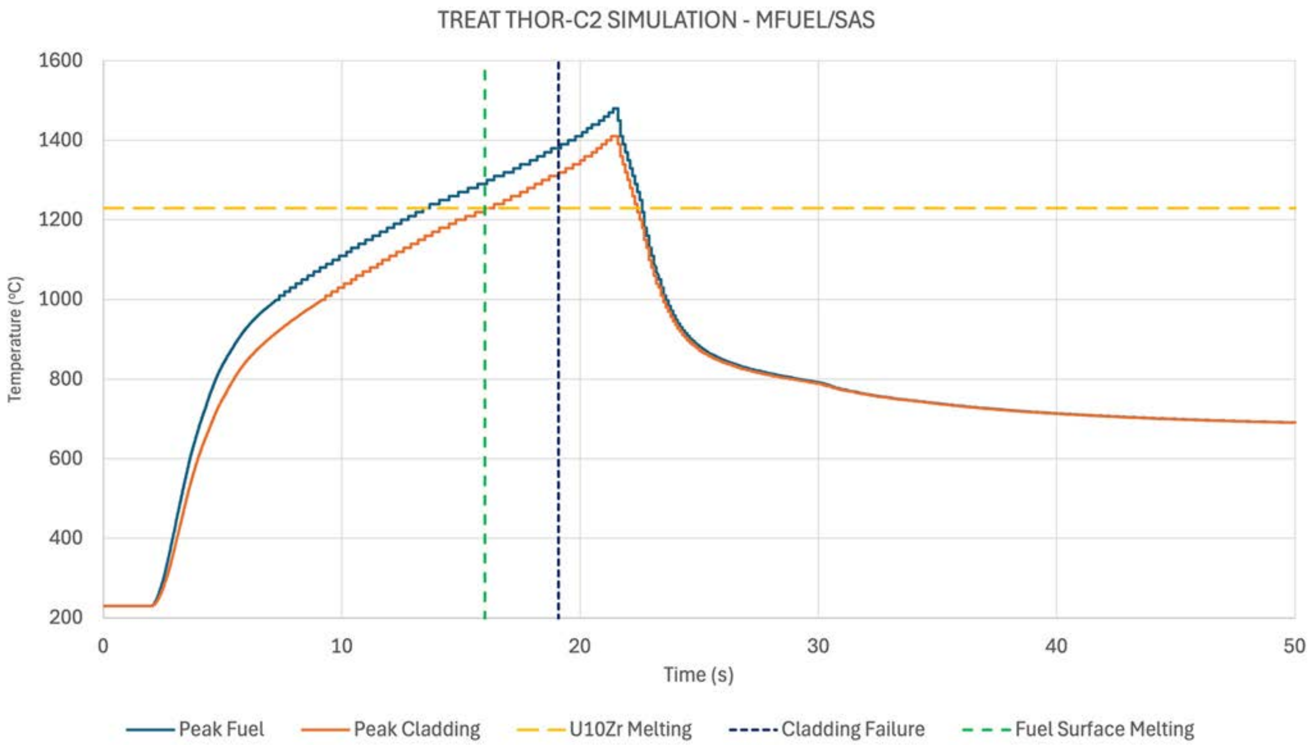



*Figure 3. SAS4A/SASSYS-1 (SAS) simulation results for THOR-C2 experiment.*

The transient heat conduction equation in axisymmetric coordinates is given in Equation 1. Integrating the heat equation over the control volume P shown in Figure 1 and subsequent discretization results in Equation 2.

The model has been applied to different radial regions by incorporating the heat source distribution and material properties.

### Updates on Fuel Clad Chemical Interaction Model

MFUEL's fuel cladding chemical interaction (FCCI) model for the fresh fuel is found too conservative, as it predicts eutectic formation at the cladding's inner surface as soon as the temperature exceeds the eutectic formation threshold. However, since the fresh fuel is not initially in contact with the cladding, FCCI may be significantly delayed. In MFUEL, fuel can be in free swelling, soft contact, or hard contact mode [1]. In this study, it is assumed that FCCI is delayed until the fuel surface begins to melt, prior to the soft contact phase. This assumption is found to produce excellent results compared to THOR-C-2 experiments.

### Preliminary Validation of SAS4A/SASSYS-1 using THOR-C2 Capsule Tests

The results of the TREAT THOR-C-2 capsule experiment were used to validate the SAS4A/SASSYS-1 code with MFUEL and the new 2D capsule heat transfer model. The variation in TREAT power during the THOR-C-2 experiment is shown

**Guiding simulation tools like BISON and the MFUEL module of SAS4A/SASSYS-1 for planned TREAT experiments will optimize these experiments, significantly reducing time and funding needed to generate relevant data, which will be highly beneficial for licensing metallic-fueled fast reactors.**

in Figure 2 (blue line). After a power ramp up to nearly 50 MW, the power moderately increased over 20 seconds before the reactor was shut down. Measurements indicated cladding failure approximately 20 seconds into the transient.

The SAS4A/SASSYS-1 simulation results for THOR-C-2 are shown in Figure 3. The fuel pin heats up, and the temperature of the rapid eutectic reaction is reached at around 10 seconds, but the rapid eutectic model was not activated because there was no predicted contact between the fuel and cladding. Fuel melting initiates, and surface melting occurs near 16 seconds. This is when it is assumed that the fuel loses mechanical integrity and contacts the cladding. The rapid eutectic penetration model is activated at around 16 seconds. Cladding failure, due to thermal creep rupture augmented by eutectic wastage formation at the cladding's inner surface, is predicted to occur at around 19.7 seconds, which is very consistent with the experimental observations. Finally, after the reactor shutdown, the capsule cools down toward the equilibrium temperature. The predicted equilibrium temperature aligns well with BISON and ABAQUS predictions.

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# Parameterizing Irradiation Effects of HT9 and Fuel Performance Modeling of Accelerated Metal Fuel Tests

*Principal Investigator: Boone Beausoleil (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Alexander Swearingen, Jacob Hirschhorn (All INL)*

***This work establishes the applicability of FAST methodology acceleration to the understanding of HT9 cladding behaviors in irradiation conditions by defining the divergences from conventional irradiation tests.***

The Fission Accelerated Steady-state Test (FAST) experiments have been designed to accelerate integral testing of next generation reactor fuel concepts. This methodology is key to the Advanced Fuels Campaign (AFC) irradiation testing portfolio as it reduces the time cost of irradiation testing. However, questions have been raised concerning the applicability of the FAST methodology for understanding cladding behaviors. This work compares HT9 cladding material behavior of metallic fuel FAST tests with Experimental Breeder Reactor (EBR)-II experiments using BISON fuel performance modeling to parameterize the irradiation effects on HT9 and provide insight into the differences between the FAST methodology and full-size experiments.

## Project Description

The technical objectives of this work are to perform an analysis of the applicability of the FAST metallic fuel experiments to understanding HT9 cladding behavior when comparing to EBR-II experiment HT9 cladding performance. The FAST experiments utilize scaling behaviors to accelerate burnup of the fuel, and due to this, the cladding is subjected to approximately 1/4 of the neutron fluence experienced in a conventional EBR-II irradiation test. Also, EBR-II experiments exhibit meaningful thermal creep behavior due to the long duration of the experiments. This behavior is diminished in the FAST experiments as the duration of the test is shortened and the stress imparted on the fuel more rapidly. These discrepancies between the conventional irradiation tests performed in EBR-II and the FAST experiments require

*Figure 1. Summary of simulated FAST and EBR-II experiments with burnup, PCLT, and PICT.*

| Rodlet | Irradiation Time (days) | Average Burnup (%FIMA) | PFCT (°C) | PICT (°C) |
|---|---|---|---|---|
| FAST-008 | 63.23 | 3.46 | 529 | 429 |
| FAST-016 | 128.23 | 7.58 | 634 | 522 |
| FAST-031 | 128.23 | 7.79 | 597 | 492 |
| FAST-048 | 188.23 | 11.8 | 617 | 508 |
| FAST-050 | 248.23 | 11.8 | 618 | 509 |
| FAST-052 | 188.23 | 11.4 | 618 | 509 |
| FAST-053 | 248.23 | 13.8 | 613 | 505 |
| X425-T423 | 549.93 | 9.20 | 707 | 616 |
| X425-T424 | 951.71 | 14.7 | 706 | 612 |

additional analysis to ascertain the effect these differences have on the final cladding performance results. This work will attempt to understand these differences and identify any correlation that can be made between the two methodologies. This work will use as-run simulations of the FAST experiments and EBR-II experiments to compare typical cladding performance criteria, such as diametral strain and Fuel Cladding Chemical Interaction (FCCI) wastage thickness.

### Accomplishments

This work has concentrated on the FAST control experiments for simulation as they facilitate comparison between the novel fuel designs and the traditional EBR-II experiment design. By comparing the control experiments with both the novel design experiments and EBR-II experiments, conclusions can be drawn about the novel design performance in relation to the traditional EBR-II design performance while decreasing the time in-reactor These control experiments are all solid pin, sodium bonded, 75% smear density U-10Zr fuel with HT9 cladding. The EBR-II experiments that were compared were chosen with these same parameters. The FAST and EBR-II experiments are simulated using the BISON fuel performance code and are listed in Figure 1 along with simulated burnup, simulated Peak Fuel Centerline Temperature and simulated Peak Inner Cladding Temperature (PICT). All FAST experiments have been simulated to the most recent cycle using the BISON fuel performance code. This modeling was performed to understand the comparability of the HT9 cladding performance in the FAST experiment results and the EBR-II experimental results. To accomplish this, the diametral cladding strain and FCCI wastage thickness of the simulated FAST experiments have been compared to the simulated EBR-II experiments. Both results are displayed as a function of the length of the experiment normalized to the length of the fuel pin. The diametral cladding strain results are shown in Figure 2 while the FCCI wastage thickness results are shown in Figure 3. The diametral cladding strain results show small strains in the FAST simulations, around 0.05% while EBR-II pin X425-T423 shows simulated strains around 0.6% and

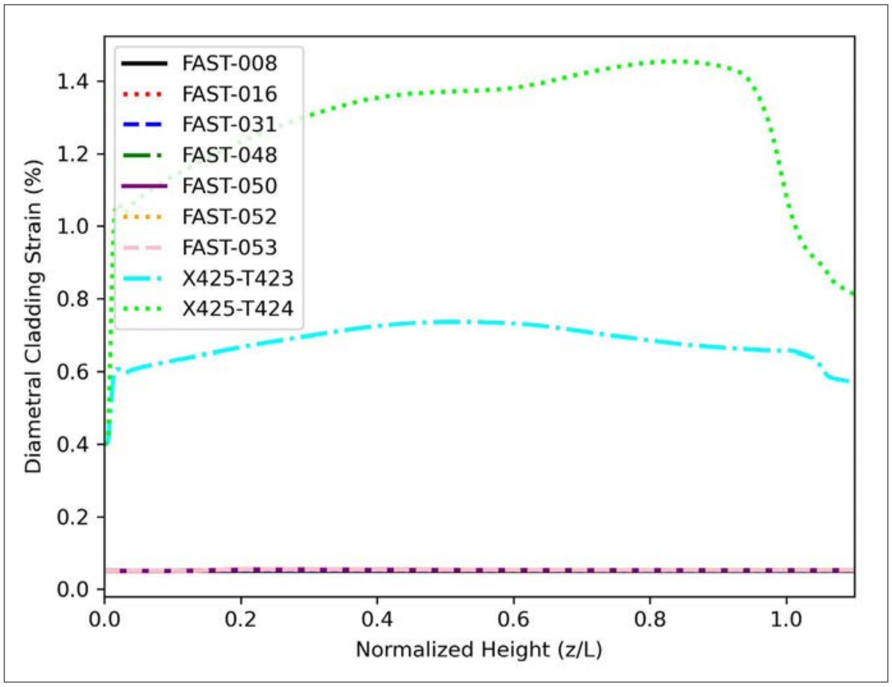
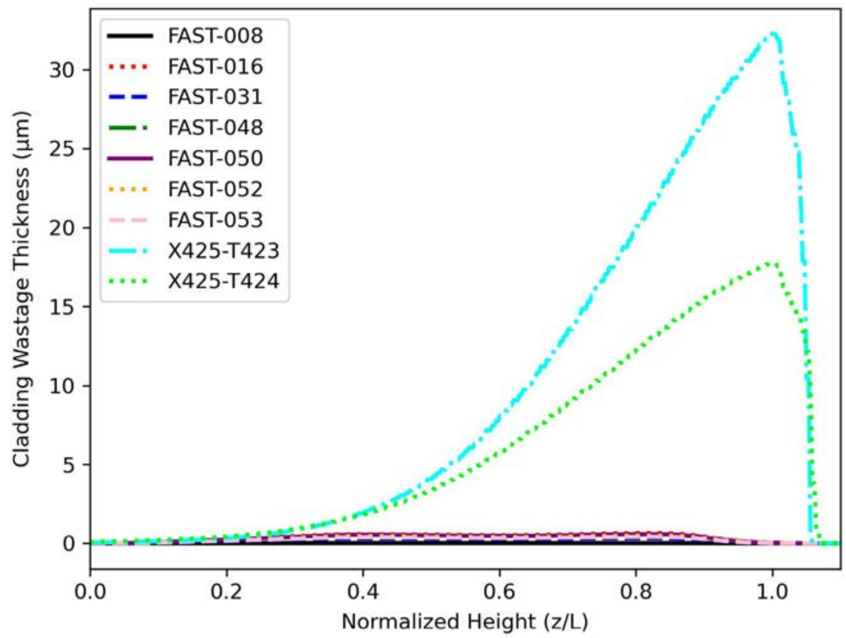



*Figure 2. Comparison of simulated diametral strain for FAST and EBR-II experiments.*

Figure 3. Comparison of simulated FCCI wastage thickness for FAST and EBR-II experiments.




X425-T424 shows simulated strains from 1.2 – 1.4%. This demonstrates the impact that the thermal creep has on the cladding performance, as the EBR-II experimental HT9 experiences temperatures at least 100°C higher at the peak compared to the FAST experiment counterpart, for significantly longer periods of time. Another concern arising from the thermal differences can be seen in the FCCI wastage thickness results, which are driven significantly by the temperature at the inner cladding boundary. The results seen in Figure 3 show that there is significantly more wastage developed in the EBR-II experimental HT9 cladding than the FAST experimental cladding, on the order of 30 micron larger. The weakening of the cladding due to FCCI wastage directly contributes to increased diametral cladding strain, further indicating that adjusting the temperature to conditions like EBR-II is a key component in future FAST experiments focused on cladding behavior. These comparisons highlight discrepancies in the performance behavior of HT9 cladding of prototypic fast reactor tests compared to FAST experiments. This provides the groundwork for establishing FAST experimental methodology as a tool for accelerated testing of next generation nuclear fuels.

# CAPABILITY DEVELOPMENT

## I-Loop Design and Installation

*Principal Investigator: Nate Oldham (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Kendell Horman, Vince Tonc, John Naughton, Madison Edwards, Kelly Ellis, Carlos Estrada-Perez, Michael Worrall, Jason Barney, Thomas Berti, and R. Dale Kepler (All INL)*

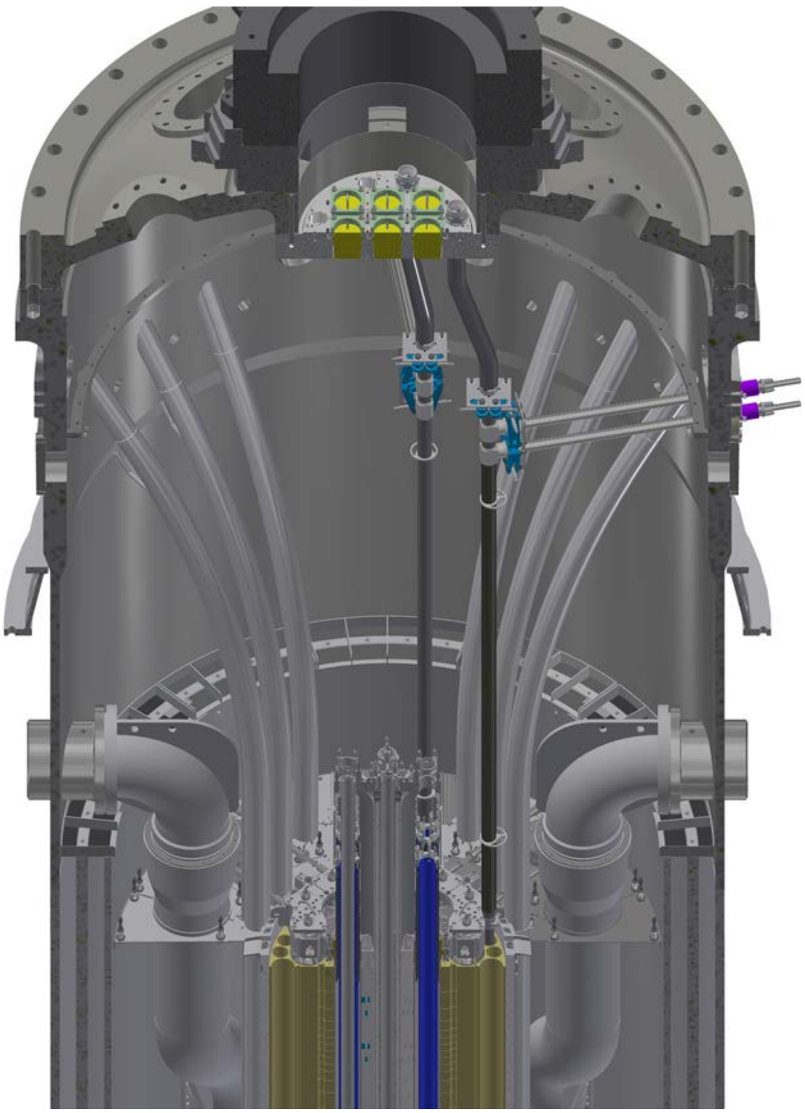
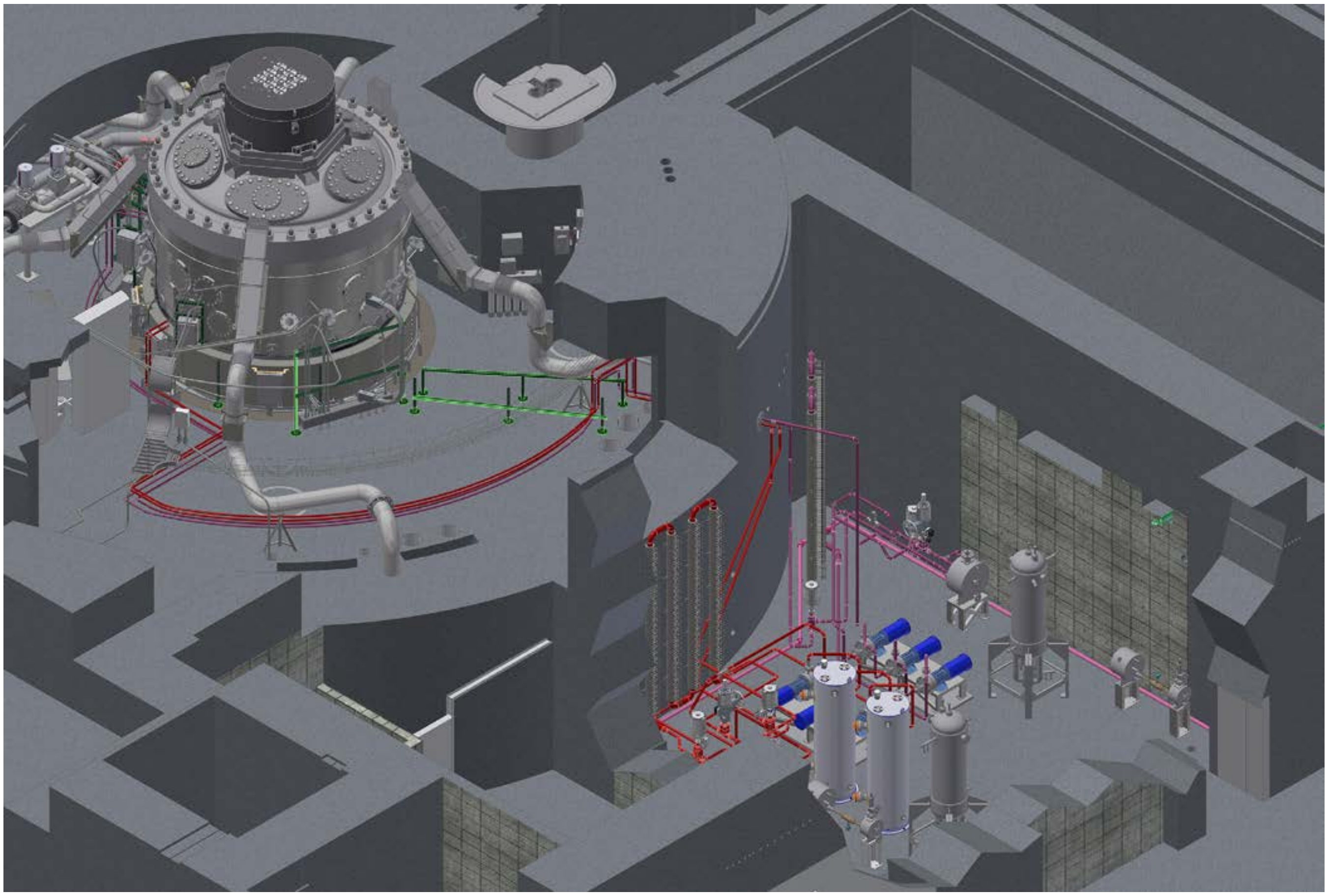


*Figure 1. 3-D model of I-Loop tube with quick-connect flanges.*

The primary objective of the I-Loop project is to expand water loop capability for light water reactor (LWR) fuels testing. This is accomplished with project scope to install two I-Loop test facilities in the Advanced Test Reactor (ATR) medium-I positions I-13 and I-14. One loop will be primarily dedicated to prototypic boiling water reactor (BWR) conditions while the other will be dedicated to prototypic pressurized water reactor (PWR) conditions.

### Project Description

The closure of the Halden Boiling Water Reactor (HBWR) and growing demand for extending 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 INL and support the industry and DOE desire for these capabilities

*Figure 2. I-LOOP X-Core system layout.*


*Figure 3. 1A Cubicle for I-Loop X-Core equipment.*

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 has progressed with the engineering design by completing a majority of tasks well into preliminary design (60%) and some into final design (90%). Additionally, all long lead procurements have been adequately specified to the point that contracts are in-place for the items to be received in fiscal year (FY) 2025. The objective is to have the components on-site and ready for installation into the reactor facility during a maintenance window.

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 / Shield Cylinder – 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. The I-Loop Tube design is nearing completion and is expected to complete a final design review in the first quarter of FY2026. Significant progress has been achieved in identifying the proper system safety classification and seismic requirements.

The I-Loop Tube is a first-of-kind design with complex features that are challenging to fabricate. To mitigate this risk, extensive prototyping is needed. A development contract is underway with Leidos Dynetics Co. to determine optional features and welding techniques to ensure a successful fabrication. Leidos Dynetics has the capabilities to electron-beam weld a ~three meter long I-Loop Tube as well as the expertise with stainless steels and zircaloys. The weld parameters are now developed to the point that a full-scale mockup will be fabricated in FY2025.

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 2). This system includes a series of sub-systems such as loop makeup, ion exchange/filtrations, sampling, and loop chemistry. A major accomplishment in FY2024 has been the completion of the final design of a purification system. The purification system is designed to clean-up a

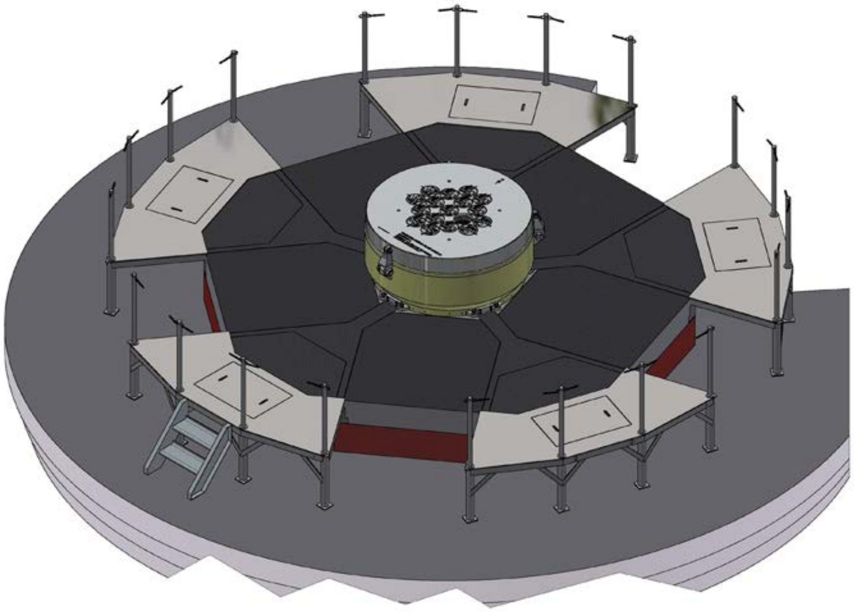


*Figure 4. 3D model of TSP-Mark II installed.*

*Figure 5. Fabrication of Shield Cylinder-Mark II.*

fission product release from testing specimens in off-normal conditions. This represents a major capability of the I-Loop system. The X-Core work also saw the completion of the major task of renovating the 1A cubicle (Figure 3) paving the way for installation of larger I-Loop equipment.

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 and Shield Cylinder. Major fabrication of the Transfer Shield Plate – Mark II and Shield Cylinder – Mark II (Figures 4 and 5) was performed in FY2024 with expected completion in the first quarter of FY2025. Once installed, this will complete the modifications to the reactor top allowing test insertion/extraction in medium-I positions.

***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 and a prototypic BWR test environment.***

# I-Loop Test Rig Design Concepts and Capabilities

*Principal Investigator: Nate Oldham (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Madison Tippet, Brian Durtschi, Ryan Sandbeck, Michael Worrall, Andrew Prince (All INL)*

The I-Loop team is designing specific boiling water reactor (BWR) and ramp test rig design concepts. These concepts will be utilized as a basis for the first commissioning experiments. Experimenters can utilize the base designs as examples of capabilities design boundaries. The test rig design concepts also inform the facility designers of necessary support equipment is available.

## Project Description

The I-Loop project is designing and deploying two separate loops: one prototypic Pressurized Water

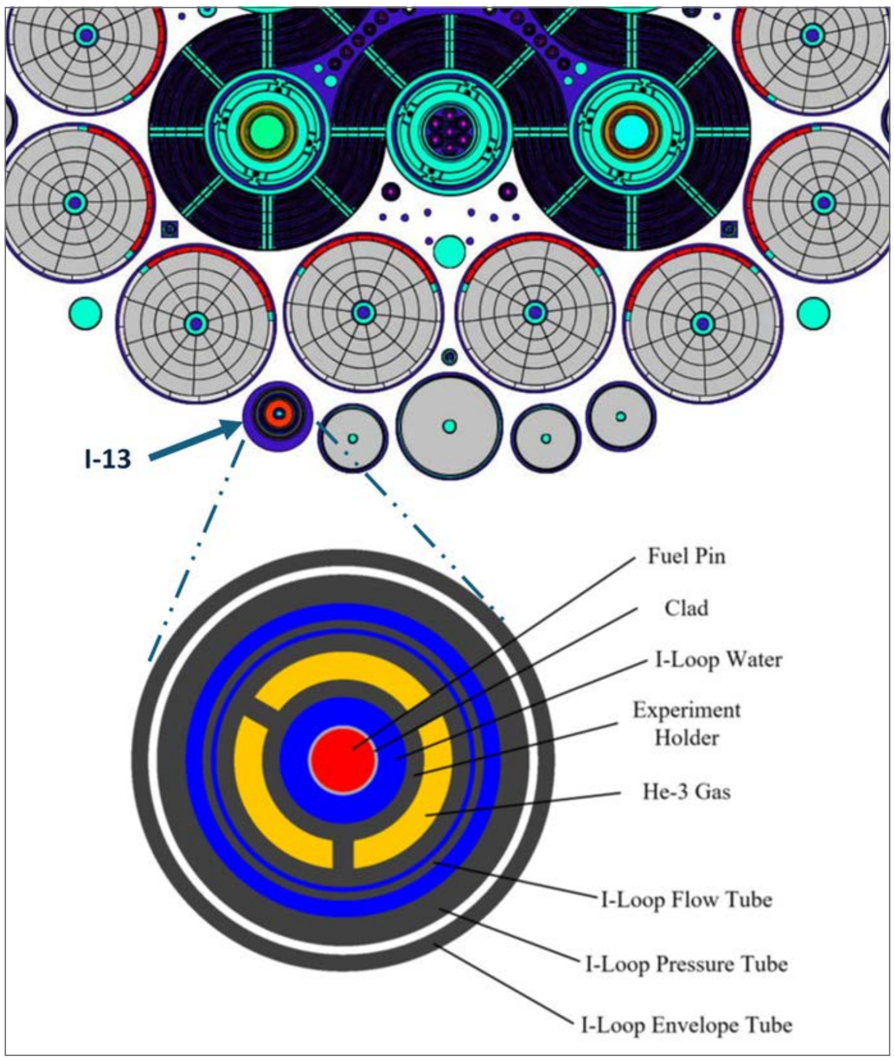



*Figure 1. I-Loop Ramp operated in ATR's high-power cycles to conduct power manipulations.*

*The ramp and BWR experiment test rigs provide experimenters the overall framework for typical testing capabilities in the I-Loop.*

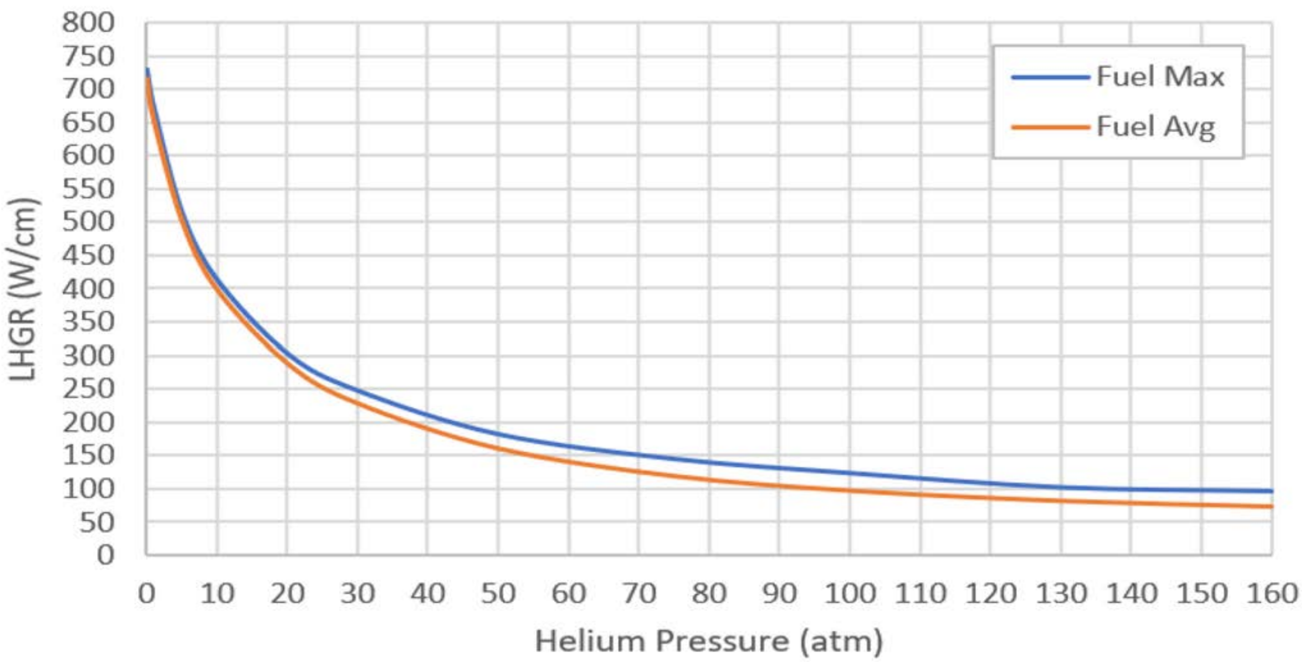



*Figure 2. Fuel pin LHGR as a function of He-3 pressure.*

Reactor (PWR) and another BWR environment. The team has been developing two different experiment rigs for these loops. The first is a ramp test rig that is capable of manipulating the localized rod power to simulate off-normal operational transients. The second is a test rig capable of a two-phase flow that simulates a prototypic BWR environment.

### Accomplishments

**Ramp Test Rig:** The ramp test rig design team has developed an experiment concept that conducts crucial performance on the pellet cladding interaction (PCI). A used "burned" fuel pin with a starting enrichment will be irradiated in a standard ATR cycle to a burnup of 19.5 MWd/kgU. It will then be operated in ATR's high-power cycles to conduct power manipulations using a Helium-3 screen (Figure 1) around the fueled specimen. The pressurization of the Helium-3 reduces the fuel pin's linear heat generation rate allowing the experimenter to tailor the ramps independently of the test reactor power (Figure 2).

**BWR Test Rig:** Currently, pressurized water loops are the only testing facilities available to test BWR fuel concepts. The test environment is at a non-prototypic higher pressure/temperature at a single-phase fluid flow condition. This void in the LWR test bed capabilities is one that the I-Loop is uniquely situated to provide. The BWR test rig design (Figure 3) team has developed an experiment concept that creates a two-phase flow test environment. The design concept shows that not only is two-phase flow achievable, but a wide range of void fractions is available to the experimenter.

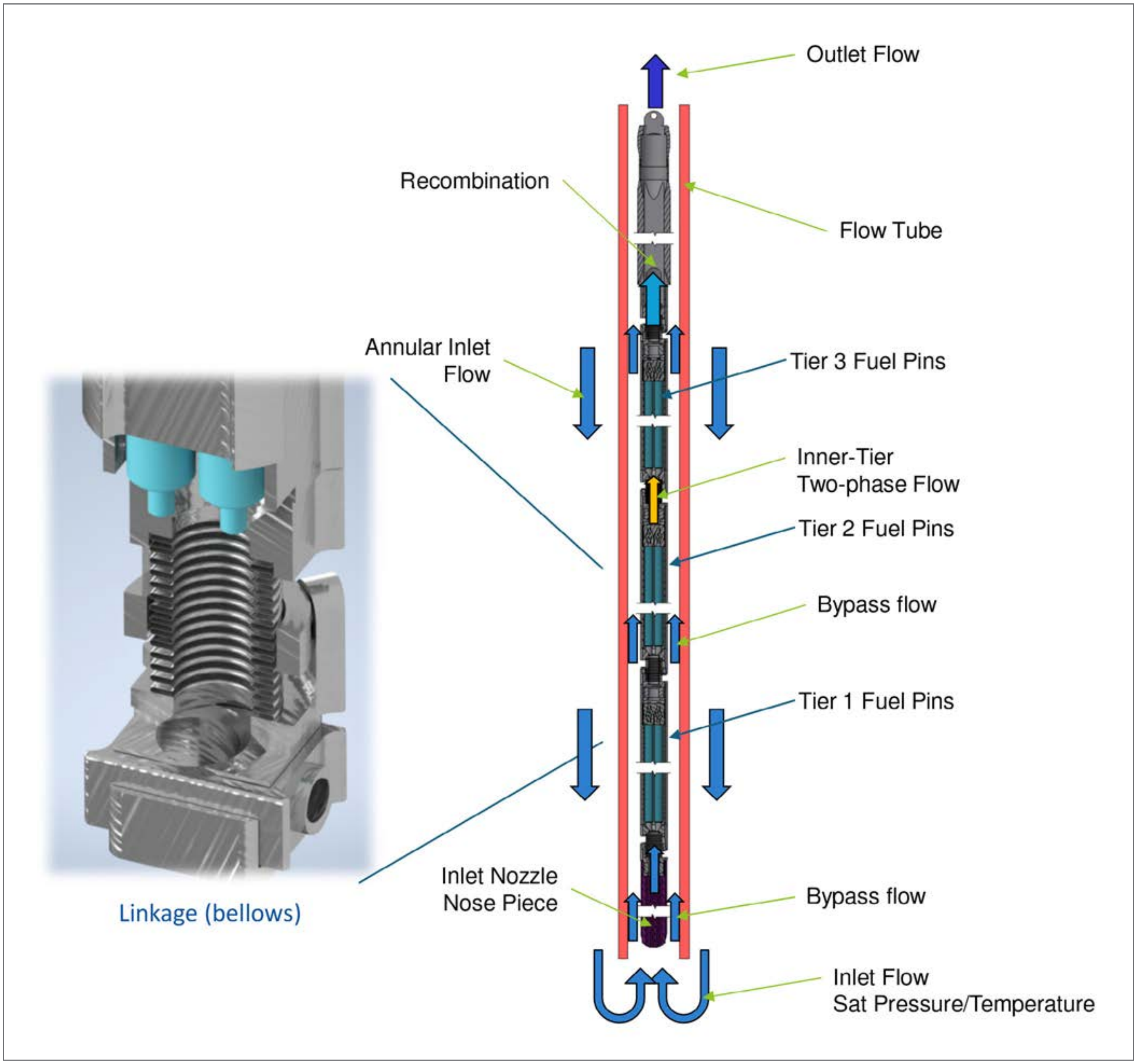



*Figure 3. BWR test rig design.*

## TWIST Device Commissioning Results in TREAT

*Principal Investigator: Colby Jensen (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Klint Anderson, Austin Fleming, Charles Folsom, Robert Armstrong, Cindy Fife (All INL)*

***The commissioning testing of the TREAT TWIST device is successfully generating needed results and data, completing the first goals of the industry-consensus AFC LOCA plan to support industry goals for extended burnup.***

Transient performance, including design basis accidents, is a crucial component of nuclear fuel safety bases. Integral experiments are planned at the Transient Reactor Test Facility (TREAT) to evaluate Reactivity Initiated Accident (RIA) and Loss of Coolant Accident (LOCA) conditions on high burnup (HBu) and accident tolerant fuels (ATF). The Transient Water Irradiation System for TREAT (TWIST) has been designed to create applicable boundary conditions and diagnostics for testing irradiated fuels, especially from commercial nuclear power plants. The new device is a next evolution beyond the Static Environment Rodlet Transient Test Apparatus, already in service for simple water-based testing. The TWIST device will be utilized for several high visibility HBu experiments including the Nuclear Energy Agency Second Framework for Irradiation Experiments (FIDES-II) Joint Experimental Programme High Burnup Experiments for Reactivity initiated Accident as well as the AFC-led consensus LOCA plan endorsed by the Electric Power Research Institute Collaborative Research Fuel Technologies group and now included as a new FIDES-II project called LOC-HBu. A series of experiments are now underway to validate design predictions of the TWIST device performance, called the TWIST-Commissioning or TWIST-C experiments.

### Project Description

TREAT LOCA and RIA testing of ATF and HBu fuels is crucial to Department of Energy goals of sustaining the current fleet of light water reactors (LWRs) by providing data crucial to extending burnup and qualifying ATF designs. The TWIST device will provide world-leading RIA and LOCA testing capability. It provides a controlled environment to (1) drive a water blow-down event, loss-of-coolant, by actuating a valve to release water from the upper capsule containing a fuel specimen into an expansion tank below, and (2) tailor the neutron heating profile for RIA experiments in pool water conditions, enabling minimization of the TREAT power pulse width. Figure 1 provides an overview schematic of the TWIST capsule highlighting key components and instrumentation.

The design is new and unique requiring careful development and qualification of the systems in their deployed state. Therefore, a series of commissioning experiments were designed to accomplish the goal of quantifying and assuring the TWIST device performance. Figure 2 shows

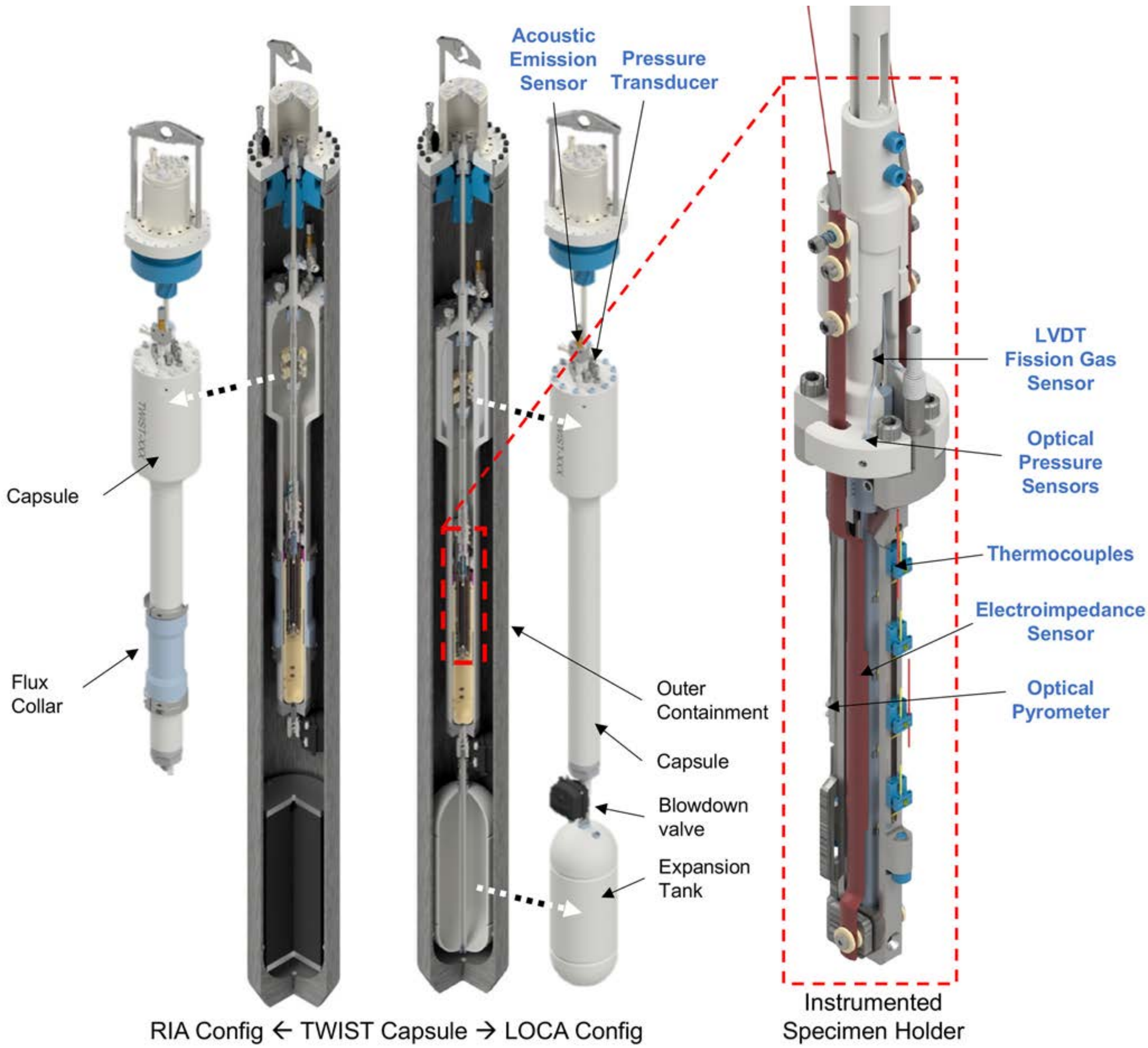



*Figure 1. Schematic overview of the TREAT TWIST test device with key components and in-situ instrumentation.*

| Test ID | Fuel Length (cm) | Rod Free Volume (cc) | Rod Internal Pressure (MPa) | Peak Cladding Temp. (K) | Purpose |
|---|---|---|---|---|---|
| LOC-C-1A to C | 25 | 15 | 0.1 | Cladding – 520 Fuel - 1600 | Fuel power calibration, pre-blowdown, centerline thermocouple |
| LOC-C-2 | 25 | 15 | 0.1 | 1173 | Fuel power calibration, post-blowdown, centerline thermocouple |
| LOC-C-3-A to E Completed in FY24 | 25 | 15 | 0.1 | multiple transients on same capsule | Detailed thermal hydraulic validation, Instrumentation validation, centerline thermocouple |
| RIA-C-1 | 25 | 15 | 0.1 | (1130 J/g-U02) | Balloon/burst Instrumentation validation |
| LOC-C-4 | 25 | 15 | ~12 | 1173 | Balloon/burst Instrumentation validation |
| LOC-C-5 | 50 | 15 | ~12 | 1273 | Long rod evaluation, balloon/burst, Instrumentation validation |

*Figure 2. TWIST device commissioning test matrix overview with competed experiments highlighted.*

a summary of the TWIST-C test matrix including both LOC-C and RIA-C experiments to fully demonstrate the TWIST capability for its intended use on high burnup fuels.

During the past year, the TWIST-C experiments have been fabricated, assembled, and tested under a wide range of conditions. These experiments utilize fresh fuel $UO_2$/Zry-4 rodlets with goals of measuring reactor-specimen power coupling, test-specimen boundary conditions, and fully demonstrate system diagnostics for evaluating irradiated fuel behaviors. To date, three capsules have been built and nine transient irradiation experiments performed on those capsules.

The TWIST device 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 in addition to a tailored flux profile for RIA experiments in pool water. 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 fuel fragmentation, relocation and dispersal, 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 device assembly and illustrates the axial flux profile and blowdown temperature history of a LOCA test specimen.

The TWIST capsule was designed to be the LWR testbed for TREAT experiments with pre-irradiated fuel. The TWIST capsule

*Figure 3. Assembled TWIST-LOC-C-3 experiment in TREAT with INL and French IRSN collaborators in background.*

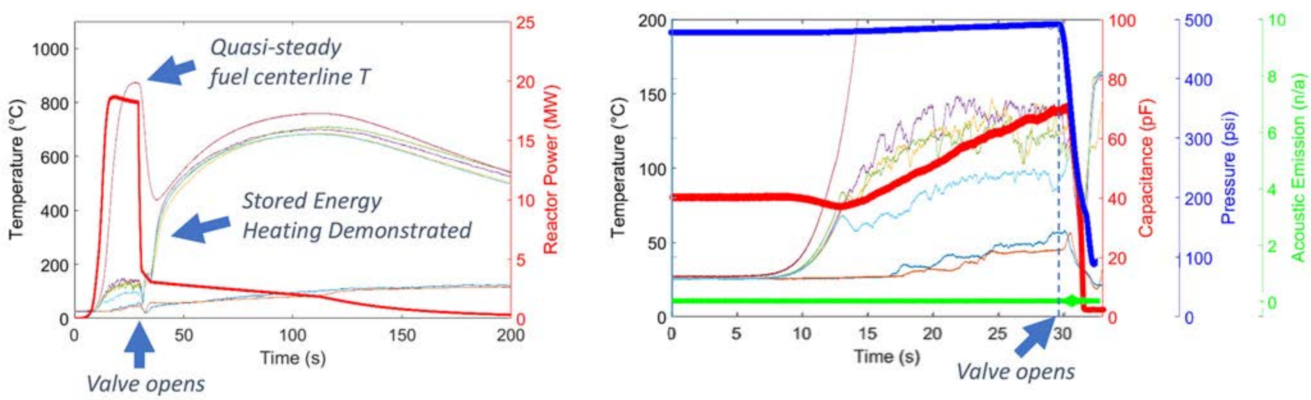



*Figure 4. Selected results from the LOC-C-3C experiment showing (left) overall temperature behavior demonstrating full stored energy heating thermal behavior in LOCA, and (right) zoom of instrumentation response during heatup and blowdown events.*

accommodates rodlets with fueled lengths ranging from 25 to 50 cm and features a state-of-the-art instrumentation package to collect relevant time-dependent data for post-test analysis of the experiment. The commissioning test instrumentation package includes an electro-impedance sensor to detect phase change events in the water and cladding radial dimension, an infrared 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. In addition, the TREAT hodoscope will provide real-time fuel tracking during the experiments to capture fuel relocation and dispersal behaviors, an important first for LOCA experiments.

### Accomplishments

For 2024, exceptional progress has been made towards the overarching goal of qualifying the TWIST device for testing HBu fuels. Specific technical goals of the project have revolved around the efforts to build two additional TWIST devices and execute and analyze nine experiments performed over the past year. These technical goals and a brief description of associated accomplishments are explained below:

1. Power measurement – The LOC-C-1 and LOC-C-2 experiments included four separate experiments on the two capsules. These experiments were designed to prevent damage to the test fuel specimens while measuring energy deposition in them using (a) fuel centerline thermocouples and (b) post-test axial gamma spectroscopy (to be completed in Fiscal Year 2025). The four experiments were performed over a range of reactor and capsule configurations that encompass its design basis. Thermocouple measurements found power input to be approximately 5-10% less than pretest predictions and a detailed sensitivity and uncertainty analysis was performed finding < 8% uncertainty at 2s in these measured power results.

2. Thermal hydraulic performance – The LOC-C-3 experiments were designed to exercise the TWIST system over a range of applicable power and thermal-hydraulic conditions to establish a known performance envelope for the system. Figure 3 shows a completed assembly of the TWIST-C-3 capsule in TREAT with INL TWIST team members and key collaborators from the French Institut de Radioprotection et de Sûreté Nucléaire (IRSN). The experiments included five transients with different capsule configurations. The third experiment, LOC-C-3C, was a full demonstration of the challenging stored energy heating simulation with results shown in Figure 4. The experiments performed very well and are now being analyzed with the new Blue-Comprehensive Reactor Analysis Bundle multiphysics capability.

3. Instrumentation – The TWIST instrumentation package shown in Figure 1 is an important technical challenge to its success. All sensors have demonstrated excellent performance. The foresight and prepared troubleshooting of the remote actuating valve was a major success, which now has a modified design to successfully perform under irradiation conditions. Select results from the LOC-C-3C experiment are shown in Figure 4. It provides a useful example of the integrated performance of cladding thermocouples, the capsule pressure sensor, thermocouples, acoustic emissions sensor, and (not shown) the hodoscope to characterize the water blowdown event. With the support of the Nuclear Science User Facilities Program, the TREAT hodoscope system was upgraded in summer 2024, from 96 channels to 192 channels, to allow full view of the TWIST internal volume. The TREAT hodoscope will provide real-time fuel tracking during the experiments to capture fuel relocation and dispersal behaviors, an important first for LOCA experiments.

### Next Steps

TWIST commissioning is not a completed story just yet. Some basic post-transient examinations will be done on LOC-C-1, -C-2, and -C-3 to further establish confidence in fuel power measurements and the state of the fuel in the experiments. A few remaining experiments shown in Figure 2 will fully demonstrate RIA and LOCA simulations under representative conditions. These experiments will be an exciting opportunity to benchmark fuel performance codes with design conditions that will significantly challenge their test specimens.

## TWIST High-Burnup Tests

*Principal Investigators: Klint Anderson, Cindy Fife (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Colby Jensen, Charles Folsom, Robert Armstrong, Jerry Kahn, Andrew Chipman, Changhu Xing, Ryan Sandbeck, Jordan Argyle, Austin Fleming, Ashley Lambson, Geran Call (All INL)*

Loss of Coolant Accident (LOCA) testing is critical to support ongoing efforts to extend the regulatory burnup limit in light water reactors (LWR) beyond 62 GWd/t. This safety testing allows continued research on fuel fragmentation relocation and dispersal in high burnup (HBu) fuels, which is the primary research and development need to extend the allowable fuel burnup limits. 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 has been designed, developed, and commissioned for use in the Transient Reactor Test Facility (TREAT) at INL.

### Project Description

The Transient Water Irradiation System in TREAT (TWIST) provides a controlled environment in which a blow-down event can be simulated in TREAT by actuating a valve to release water from the upper capsule containing a fuel specimen into an expansion tank below, as shown in Figure 1. This approach fills 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. The unique design of the TWIST device allows highly representative testing using nuclear heating, with achievable rod powers and temperatures prototypic of LOCA events.

Fabrication, assembly, and irradiation of the TWIST device have been demonstrated with fresh fuel specimens as part of a commissioning campaign as discussed in greater detail elsewhere in this report. The full potential of TWIST, however, will be realized when testing refabricated rodlets taken from segments of preirradiated HBu fuel rods. The experience, expertise, and knowledge gained by the experiment team throughout the commissioning campaign are critical in ensuring successful execution of the HBu experiments in the TWIST device. Based off this experience, the design of TWIST was updated to increase the capability of the device, including optimizing the design for remote assembly inside the Hot Fuel Examination Facility (HFEF) and designing a configuration of the device for Reactivity-Initiated Accident (RIA) testing, all while maintaining the state-of-the-art instrumentation package deployed in the fresh fuel experiment package.

### Accomplishments

Following completion of the first commissioning tests, the TWIST design was updated to incorporate refabricated rodlets made from segments of HBu fuel received from the Byron Fuel Generating Station.

Figure 1. TWIST device configured for LOCA testing.

The TWIST capsule, expansion tank, closure flange, and other main components were not changed. The design of the specimen holder, instrumentation fixtures, and other components surrounding the specimen were updated to simplify the specimen loading process of HBu fuel segments in HFEF while protecting sensitive instrumentation required to be in close proximity with the test specimen. Leveraging knowledge and expertise gained from the remote assembly of previous experiment campaigns, this design update features a hinge mechanism which allows the electroimpedence detectors and fiber optic pyrometer to hinge out of the way during specimen loading. This hinge feature, as shown in Figure 2, allows all instrumentation except cladding thermocouples to be assembled by hand prior to entering the hot cell, and minimizes the number of connections which must be made remotely.

Additional updates to the TWIST design include an additional configuration, as shown in Figure 3, which allows the device to be used for RIA testing of HBu fuels. This modification is minor in nature, and consists of removing the expansion tank, plugging the bottom port on the TWIST capsule, and installing a flux filter around the capsule at the axial location of the rodlet. These modifications increase the capability of the TWIST capsule to be the HBu test vehicle for all capsule type accident testing of LWR fuels in TREAT. A successful design review, incorporating the HBu and RIA design upgrades, was successfully completed in July 2024.

Remote handling equipment was designed, fabricated, and tested to facilitate assembly of the TWIST device in HFEF. Qualification of the remote handling equipment and HBu TWIST device commenced in 2024 with the first two phases being completed, proving the updated design and associated hardware can be assembled remotely in HFEF (see Figure 4).

***Design of the TWIST device for remote assembly has been completed, enabling the capability to perform critical safety testing on refabricated HBu fuel rods in TREAT.***

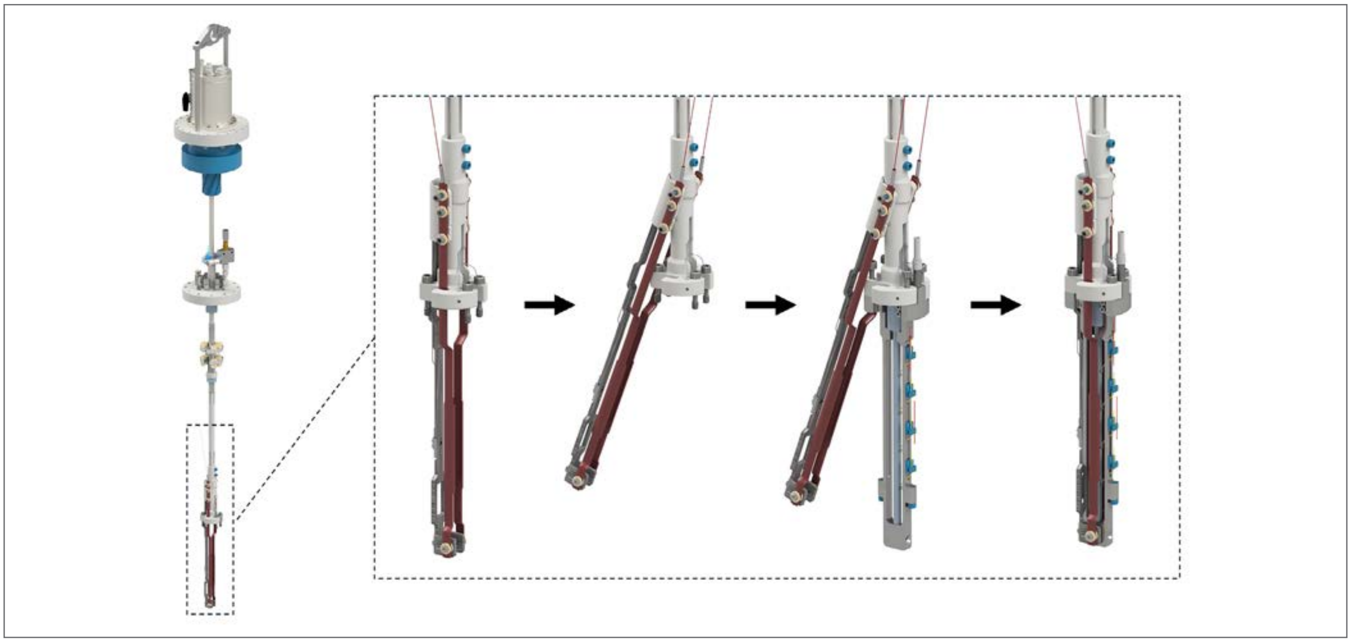


*Figure 2. Design of TWIST hinge mechanism to support remote assembly of HBu fuels.*

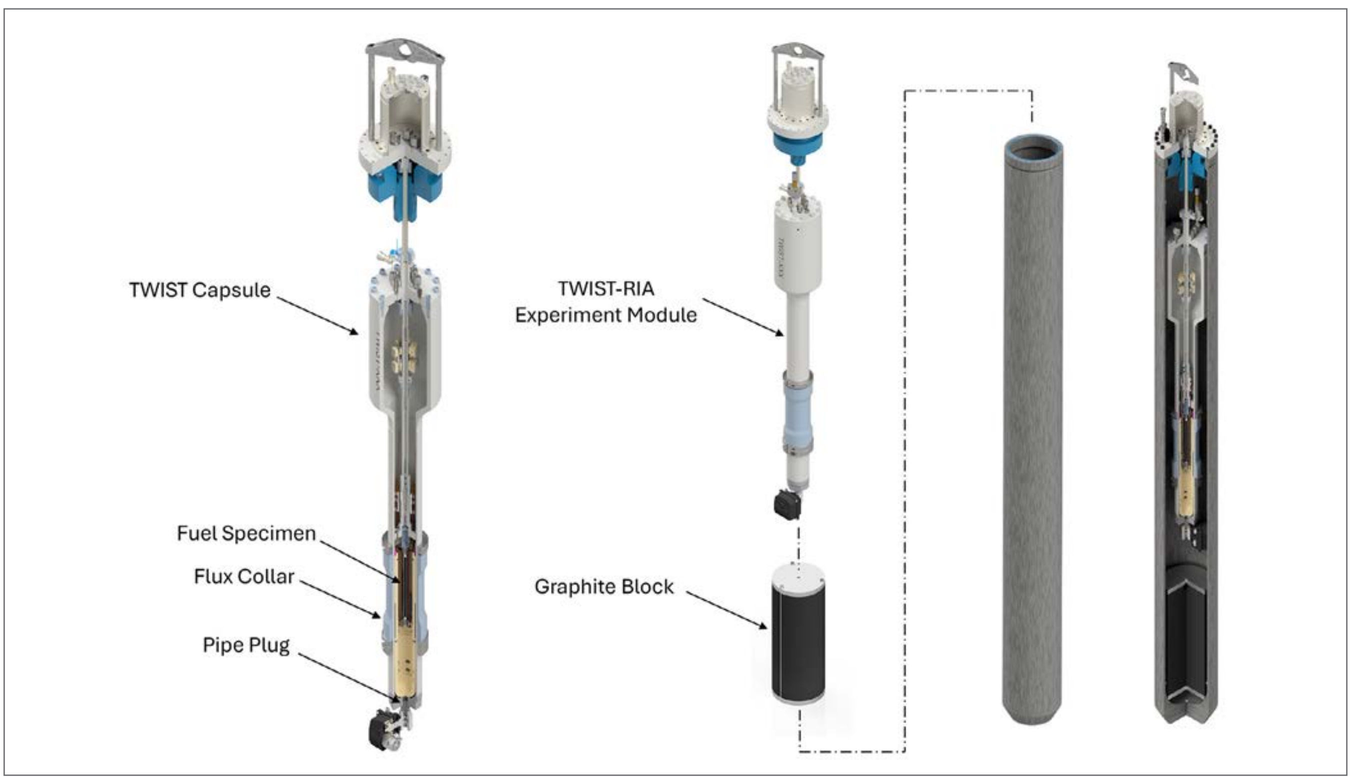



*Figure 3. TWIST module configured for RIA testing.*


With the accomplishments completed in 2024, the TWIST device and experiment team are well poised to execute on impactful experiments in 2025. The final phase of the remote handling qualification will commence shortly, which tests the experiment assembly inside HFEF to ensure operators are well trained and procedures in place for the coming HBu assembly. Fabrication of updated experiment hardware is currently underway and nearing completion. The TWIST capsule will be commissioned for RIA testing by completing a fresh fuel RIA qualification test in the coming months. All this leads to assembly and experimentation of the first refabricated rodlet (completed in 2024) that will be irradiated as part of the High-burnup Experiments in Reactivity Initiated Accidents test program in TWIST, to evaluate pellet-cladding mechanical interaction of HBu fuel.

*Figure 4. Remote handling qualification using a mockup of the TWIST design for HBu fuel specimens.*

## Refabrication of Byron Fuel Rodlet

*Principal Investigator: Jason Schulthess (Idaho National Laboratory [INL])*
*Team Members/Collaborators : Jordan Argyle, Spencer Parker, Klint Anderson, Justin Yarrington, Clayton Turner, Cad Christensen, Mark Cole (All INL)*

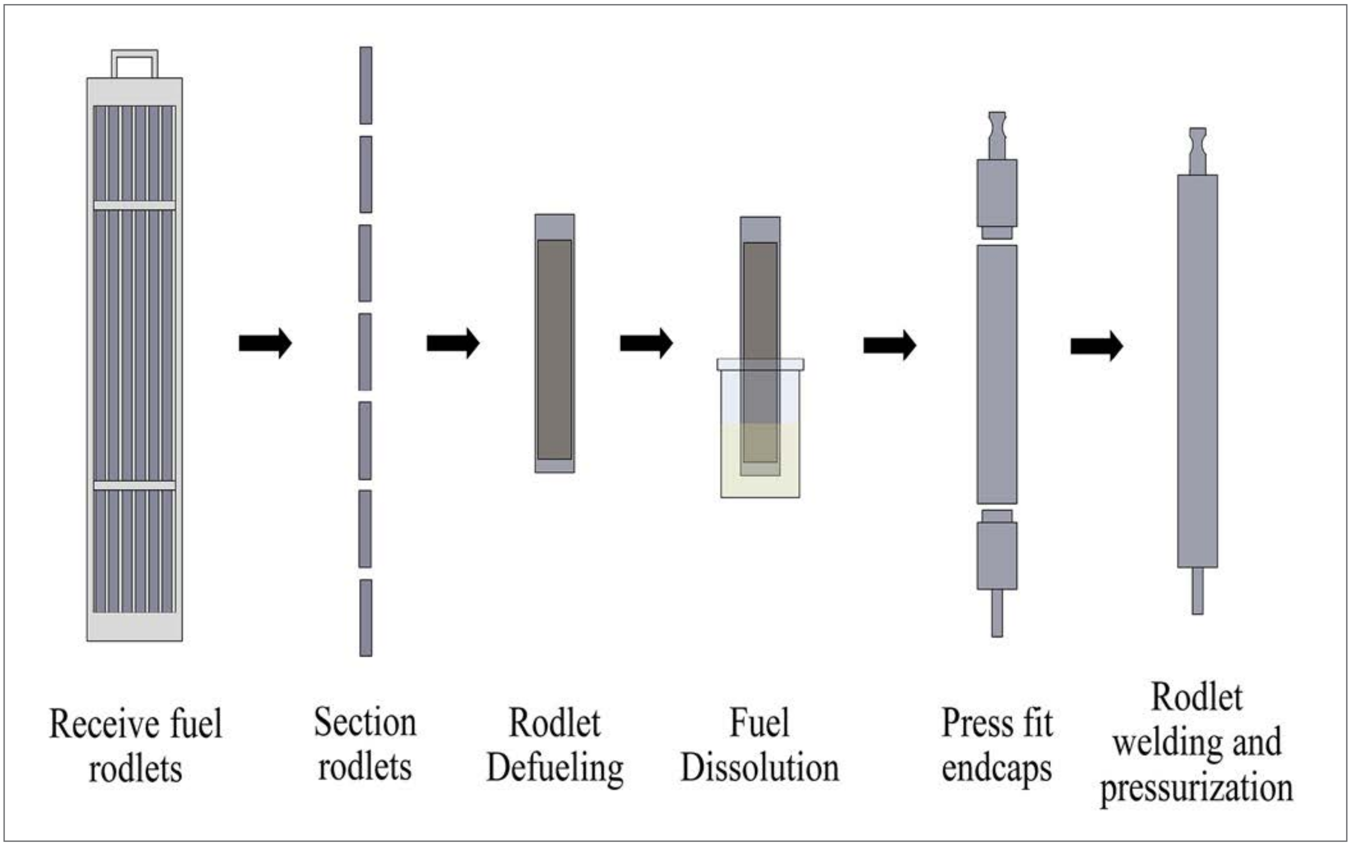



*Figure 1. Refabrication process steps.*

Irradiated fuel rod refabrication is a crucial enabling capability that bridges the area between base reactor irradiation and subsequent experimentation and/or re-irradiation. This process is key to performing meaningful research and development (R&D) on fuels with any level of burnup, especially at the Transient Reactor Test (TREAT) Facility and opens the door to R&D for materials tested in commercial nuclear power plants. Refabrication allows access to fuel at any point in its lifetime, allowing opportunity to apply instrumentation and perform experiments to measure performance under a variety of specified conditions. Secondly, the ability to repackage previously irradiated fuel into experiment vehicles is an enabling capability that ensures experiments are instrumented and

*Figure 2. Segment harvested from parent rod to be used for the HERA-HBU-1 test.*

*Figure 3. Bottom of rod segment after mechanical defueling.*

contain desired boundary conditions for experiment objectives. Measurements on high burnup fuels are impractical, if not impossible, without refabrication and experiment preparation capability.

### Project Description

The refabrication capability that was previously developed at INL's Hot Fuel Examination Facility was utilized to perform refabrication of the first segment from parent material irradiated in the Byron Nuclear Generating Station. This segment is intended for use in high burnup fuel testing in support of the High-burnup Experiments in Reactivity Initiated Accidents (HERA) test program. This program is evaluating pellet-cladding mechanical interaction with hydride rims in cladding with different reactor pulse widths. The objective of this project was to perform refabrication of the rod segment, enabling the rod segment to be utilized for the intended irradiation test. Non-destructive examination of the parent rod, particularly gamma spectroscopy, was used to inform the selection of 1) the rod segment to be used for the high burnup HERA test, and 2) adjacent samples to inform pre-transient conditions such as oxide thickness, cladding inner and outer diameter, fuel and cladding microstructure, fuel burnup value, and cladding hydrogen content. After the segment was harvested, the segment ends were defueled using a combination of mechanical and chemical defueling, which allows for new ceramic spacers, and endcaps to be added. Critically, the endcaps can contain instrumentation allowing for online measurement during irradiation. In this case, the upper end cap contained a bellows and linear variable differential transformer for real time rod internal pressure measurements. Once the endcaps are added and welded in place, the rod segment is pressurized, sealed and inspected.

### Accomplishments

The fuel refabrication process was established and used to perform refabrication of the first rod segment in support of the HERA high-burnup test series. Refabrication follows the process of segment harvesting from the commercially irradiated material, mechanical defueling, chemical defueling, assembling the rodlet with applicable spacers and instrumented endcaps, and finally welding the endcaps in places and pressurizing and seal welding (See Figure 1).

The rodlet was selected from the parent rod based on non-destructive examinations performed and maximized the available amount of fuel between grid spacers in the assembly to support the longest fueled region possible for the test (see Figure 2). Samples supporting microscopy, burnup, and cladding hydrogen analysis were also

***The first commercial fuel rod segment was successfully refabricated and will be used in the HERA High-burnup experiment series.***

obtained and are being used to establish the baseline condition of the segment being tested.

Mechanical defueling (see Figure 3) followed by chemical defueling was performed. This created the space needed to insert new spacer hardware and the instrumented endcaps, and ensured welding can proceed while minimizing potential impurities which could impact the weld.

New welding tooling, and welding parameters were developed for this specific configuration of rodlet, which addressed difficulties in welding the top endcap, due to the large endcap creating a significant heat sink. Prior to performing final welding on the rod segment for the HERA-HBU-1 test, the welding parameters and tooling were demonstrated on other materials bearing similar characteristics. Assembly and welding of the rodlet were successful as were final inspections including a leak check, indicating the rodlet can be used for the intended irradiation experiment at TREAT (Figure 4).

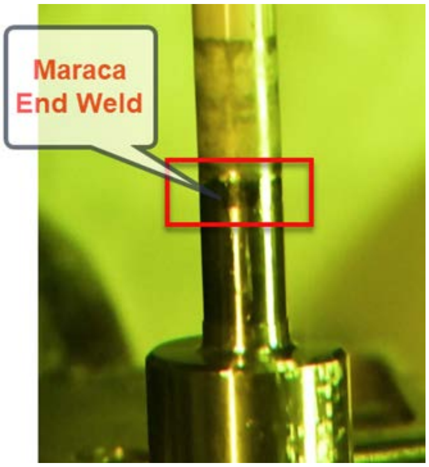



*Figure 4. Final rodlet with endcaps welds completed. Image shows the weld for the upper end cap (Maraca) which contains instrumentation to measure rod internal pressure during the planned transient.*

## In-pile Acoustic Sensing Developments

*Principal Investigator: Austin Fleming (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Geran Call, Josh Daw, Bibo Zhong (All INL)*

Meaningful acceleration of fuel development and qualification will require a significant elevation of integrated advanced modeling and in-pile characterization approaches. In-pile acoustic sensing has the potential to enable several non-contact measurement capabilities that are inaccessible by other means. This is critically important for nuclear fuel testing, where many sensing mechanisms are prohibitively intrusive, unacceptably alter the surrounding thermal hydraulic behavior, or are not geometrically or materially compatible with the environment (space, temperature, irradiation). Recently, significant progress has been made in in- and out-of-pile experimentation of acoustic sensors to detect a variety of relevant acoustic events for irradiation testing experiments.

### Project Description

Many instrumentation technologies may be used in irradiation testing of nuclear fuels to provide the on-line understanding of the conditions, behavior, and performance of the fuel system. In-situ measurement is especially useful and challenging due to the dramatic impacts that irradiation effects may have on material properties and performance. In accident testing of nuclear fuels, in-pile instrumentation is extremely valuable due to significant evolution that can occur in the material and the limited understanding that can be obtained through the final snapshot during post-irradiation examination. Acoustic sensing in irradiation experiments has the potential to provide significant value by detecting qualitative as well as quantitative data. Currently identified applications within transient irradiation experiments include detection of cladding failure, boiling events, and other mechanical events such as fuel cracking. A high priority example is to establish capability to detect the acoustic signature from a fuel pin failure, automatically identify the failure (without human interpretation), and subsequently act on this information to trigger fuel power shutdown via the Transient Reactor Test Facility (TREAT) shutdown. This goal supports the needs of the Advanced Fuels Campaign (AFC) program as well as a major Advanced Reactor Demonstration Project (ARDP) planning to use TREAT. In addition to fuel performance measurements, the acoustic emission sensor has also been used to verify experimental device performance. A successful example of this usage has been ensuring the correct performance of the Transient Water Irradiation System for TREAT (TWIST) test device during fiscal year 2024 experiments.

*Figure 1. Photograph of tubes burst through over pressurization at different temperatures.*

*This project utilizes acoustic signals to enhance data collection in irradiation experiments, with early demonstrations of its utility by automatically detecting cladding ruptures and valve actuations.*

## Accomplishments

A custom high speed data acquisition system was established which processes the data collected every 50 milliseconds (further reduction is targeted) to evaluate a potential acoustic event such as induced by cladding burst. Two primary candidate sensors have been selected for TREAT experiment applications including a high-temperature, commercial sensor and a higher temperature, custom-designed sensor developed under the Department of Energy Advanced Sensors and Instrumentation Program. These sensors and the high speed data acquisition system were tested for their ability to detect metal tube rupture, using a burst-testing setup. The experiments included mounting the sensors near tubes pressurized to cause burst in a furnace at different temperatures. An image of the burst tubes from this testing is shown in Figure 1. The measurement system successfully detected pin failure in all simulated cladding failures conducted in the furnace. The commercial sensor was validated to perform well up to its rated temperature of approximately 550°C, while the custom sensor demonstrated performance to the maximum tested temperature of 650°C.

The commercial acoustic emission sensor was incorporated into the TWIST device for several planned experiment programs. TWIST simulates a loss-of-coolant accident by remotely opening a valve, which releases water from the primary capsule into an attached blowdown tank. A short time later (~several seconds) the valve is reclosed to reseal the capsule. In-pile testing results collected from the acoustic emission sensor are provided in Figures 2 and 3, which correspond to the valve opening and closing, respectively. These results were compared to acoustic measurements performed in out-of-pile testing of the same valve and consistently found a similar acoustic signature, unique to the valve opening and closing. In addition to the valve actuation, the acoustic response of the pressurized water draining from the capsule was also measured.

The acoustic sensors have also been optimized in their adaptation to the Temperature Heatsink Overpower Response Capsule for application to AFC, Japan Atomic Energy Agency, and ARDP experiments. This project will continue to advance the application of the sensor and expand its utility via identification of other acoustic signals of interest, improving data acquisition and analysis, and continued application and testing in out-of- and in-pile experiments.

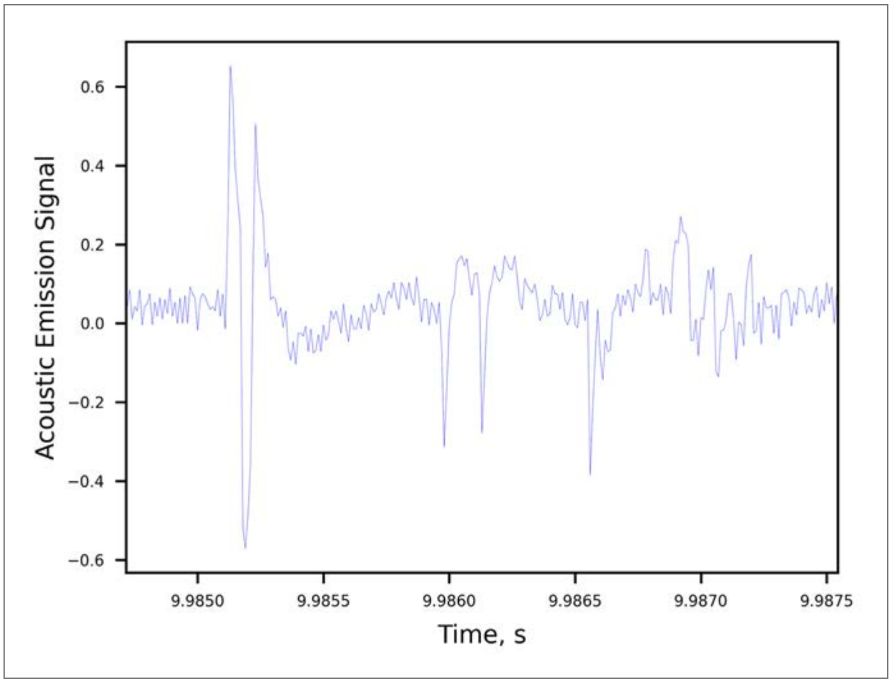



*Figure 2. Acoustic emission signal collected during valve opening.*

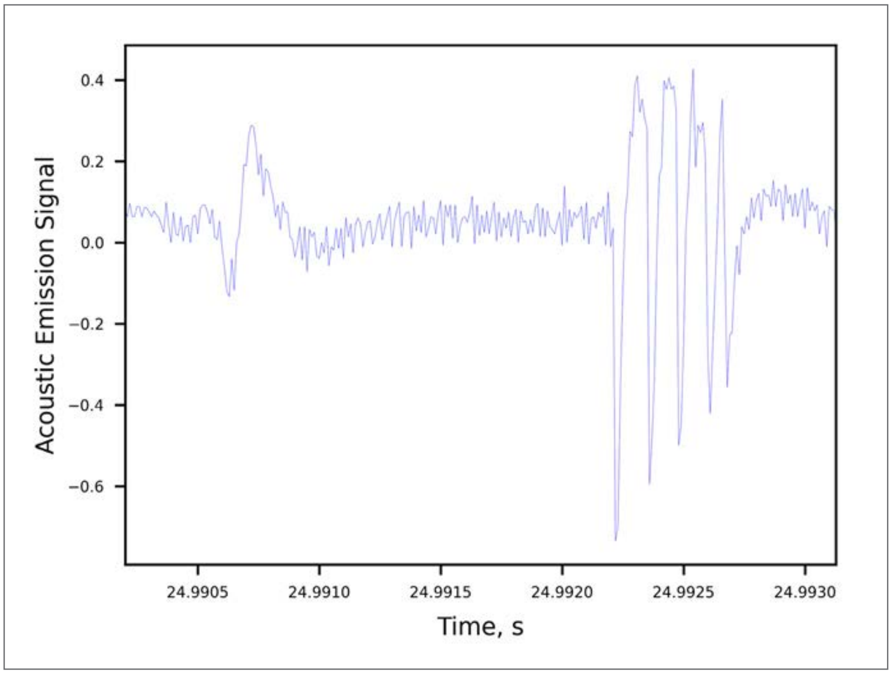



*Figure 3. Acoustic emission signal collected during valve closing.*

## Development of Instrumented Capsule Irradiation Strategies

*Principal Investigator: Nicolas Woolstenhulme (Idaho National Laboratory [INL])*
*Team Members/Collaborators: Austin Fleming, Kort Bowman, Colby Jensen (All INL)*

This project began under an INL-led Laboratory Directed Research and Development a few years ago and has now been adopted by Advanced Fuels Campaign to enable a new category of irradiation testing referred to as an "instrumented capsule". The idea which spurred this innovation was to develop an approach superior to drop-in capsules or rabbits in the sense of being able to provide real time data from in-core sensors, but not as complicated or as costly as lead-out experiments where gases are monitored and controlled as they sweep through the test assembly. Functionally, this new approach takes the form of a long tube which contains the specimens reaches meters above the core to support instrument penetrations through the reactor pressure vessel. A key enabling feature for this approach in the Advanced Test Reactor (ATR) was a new top head closure plate having eight new penetrations that was installed during a long beryllium replacement outage. The new top head closure plate was installed in preparation for flowing pressurized water I-Loops, but the first to benefit from these new penetrations will be this new instrumented capsule.

### Project Description

The first experiment to utilize this approach is termed the Irradiated Material Properties Accelerated Characterization test (IMPACT-1). This "acro-name" captures well the intent of such tests; to accelerate fuel development by allowing real-time fuel performance data. IMPACT-1 will irradiate metallic fuel specimens using a cadmium basket to produce fast reactor like radial power profiles. A novel thermal conductivity probe will be placed in the centerline hole of three different metallic fuel designs including a standard sodium bonded cylindrical slug, a sodium free annular slug, and a sodium free "slotted" slug (see Figure 1). A solid stainless-steel slug will also be irradiated with this sensor to serve as a baseline measurement. The sensor itself combines the principles of a resistance thermal detector, intrinsic electrical heating, and modern signal lock-in processing to measure thermal conductivity change during irradiation. The thermal conductivity of metallic fuel is one its most celebrated, but remarkably it has never been measured in core like this. The test will achieve a burnup level adequate to reach full ~25% swelling to the interconnection of porosity thresholds. The largest thermal conductivity changes are postulated to occur in this burnup range. Adaptations to this approach have already been developed in

**IMPACT-1 will commission a new instrumented capsule capability in ATR able to acquire crucial fuel performance data in real time as part of the overall strategy to accelerate and refine fuel development through irradiation testing.**

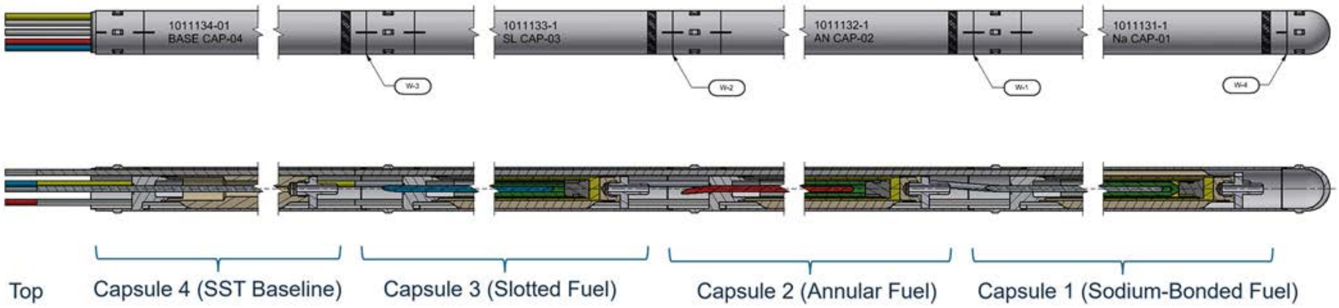



*Figure 1. Renderings of the in-core region of IMPACT-1.*

some detail where thermal conductivity measurements would be made in uranium nitride/carbide or where pressure sensors would instead be used to measure fission gas release during irradiation to compare chromia-doped and standard $UO_2$.

## Accomplishments

The IMPACT-1 experiment was designed rather rapidly, especially given its novelty, by using principles of the agile software development process, although modified heavily to reflect significant differences in the practical considerations of nuclear materials experiments. Several concepts were generated, evaluated, and down selected by using “sprint cycles” focused on question closure. Unexpected revisions to ATR’s safety bases were implemented in the middle of the design process which affected the safety analysis approach and slowed progression somewhat. These new considerations were eventually overcome and have paved the way for future tests of this type. Notably this effort made possible an analysis approach that allows instrumented capsules to demonstrate compliance with regulations when leaving the experiment installed in-vessel during refueling outages. This approach reduces handling evolutions on the experiment hardware and thus maximizes likelihood of instrumentation survival throughout the irradiation campaign. Presently the IMPACT-1 hardware and fuel specimens have been manufactured. Assembly of these items into the complete device is now underway and slated for irradiation to begin in 2025.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Vogtle-2 PIE
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# APPENDIX

## 5.1 PUBLICATIONS

| Author | Title | Publication |
|---|---|---|
| Aldeia Machado, L., Merzari, E. & L. Charlot | Parametric Study of the CHF Occurrence during RIAs for High Burnup Fuels | American Nuclear Society Annual Meeting (2024) |
| Aldeia Machado, L., Merzari, E. & W. Walters | Low-temperature Cladding Failure during an RIA Transient for High-Burnup Fuels | American Nuclear Society Winter Meeting and Technology Expo. (2022) |
| Aldeia Machado, L., Merzari, E. & W. Walters | Neutronics and Thermal-Hydraulics Analysis of Ria Transients for High-Burnup Fuels | International Conference on Nuclear Engineering (ICONE-30), American Society of Mechanical Engineers (2023) |
| Aldeia Machado, L., Merzari, E., Walters, W. & L. Charlot | CHF Prediction during an RIA With a BISON and THM Coupled Model | International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-20), American Nuclear Society (2023) |
| Aldeia Machado, L., Nantes, K., Merzari, E., Charlot, L., Motta, A. & W. Walters | Toward development of a low-temperature failure envelope of cases for high-burnup RIAS under PWR operational conditions | Nuclear Engineering and Design, under review August 2024 |
| Anderson KS, Hale DD, Schulthess JL, Arrowood MM. | A standard capsule design for structural material testing in the Advanced Test Reactor | Nucl Eng Des. 2023;414:112630. https://doi.org/10.1016/j.nucengdes.2023.112630 |
| Beck PM, Hayne ML, Liu C, Valdez J, Nizolek T, Briggs SA, Maloy SA, Saleh TA, Eftink BP. | Mandrel diameter effect on ring-pull testing of nuclear fuel cladding | J Nucl Mater. 2024;596:155087. https://doi.org/10.1016/j.jnucmat.2024.155087 |
| Burden, D.E., Harrell, T.M., Bumgardner, C.H., Roache, D.C., Walters, J.L., Maier, B.R. & X. Li | Unveiling fracture mechanics of a curved coating/substrate system by combined digital image correlation and numerical finite element analyses | Engineering Fracture Mechanics, Volume 296 (2023), ISSN 0013-7944, https://doi.org/10.1016/j.engfracmech.2023.109827 |
| Campos, S.D | Identifying Parameters Generating Notable Fuell Locking during Pressurized Dispersion of Fragments | Oregon State University Master's Thesis (June 2024) |
| Campos, S.D., Howard, T.K., Hendrickson, G., Weiss, A., Mignot, G. & W. Marcum | Development of Solids Transport Model for Fuel Dispersal Studies | NURETH '20 Annual Meeting Paper, August 2023 |
| Campos, S.D., Howard, T.K., Hendrickson, G., Weiss, A., Mignot, G. & W. Marcum | Initial Characterization of Observed Dispersion and Relocation Phenomena during a Loss of Coolant Accident | NURETH '20 Annual Meeting Paper, August 2023 |

| Author | Title | Publication |
|---|---|---|
| Campos, S.D., Howard, T.K., Yamasaki, S., Mignot, G., Marcum, W., Wissinger, G. & L. Gerken | Identifying Important Parameters Affecting Fuel Dispersion and Relocation Phenomena under Loss of Coolant Accident (LOCA) Conditions | American Nuclear Society Winter Meeting Paper (November 2023) |
| Dabney, T., Sasidhar, K.N., Yeom, H., Miao, Y., Mo, K., Jamison, L. & K. Sridharan | In situ synchrotron investigation of tensile deformation and failure mechanisms in cold spray Cr-coated Zr-alloy system | Materials and Design, Volume 239 (2024) |
| Dabney, T., Yeom, H., Maier, B., Walters, J., Sasidhar, K., Eftink, B., Li, N. & K. Sridharan | Microstructure, Mechanical Properties, and Performance of Cold Spray Cr Coatings on Zr-alloy Fuel Cladding | Symposium on Accelerated Qualification of Nuclear Materials Integrating Experiments, Modeling, and Theories, The Metallurgical Society Annual Conference, Orlando, FL, March 2024 (Professional Presentation) |
| Dunbar, C., Jung, W., Armstrong, R., Sridharan, K., Corradini, M. & H. Yeom | Fuel performance analysis of Cr-coated Zircaloy-4 cladding during a prototypical LOCA event using BISON | Annals of Nuclear Energy, Volume 200 (2024) |
| Dunbar, C., Jung, W.H., Fox, N. Demo, T. Maier, B. Armstrong, R. Sridharan, K. Corradini, M. & H. Yeom | Effect of Initial Temperature on Quench Behavior of Cr-Coated Zr-Alloy Cladding | Transactions of the American Nuclear Society, Washington, D.C., November 2023 |
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| Author | Title | Publication |
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| Xu F, Yao T, Xu P, et al. | Multi-Scale Characterization of Porosity and Cracks in Silicon Carbide Cladding after Transient Reactor Test Facility Irradiation | Energies. 2024;17(1):197. https://doi.org/10.3390/en17010197 |
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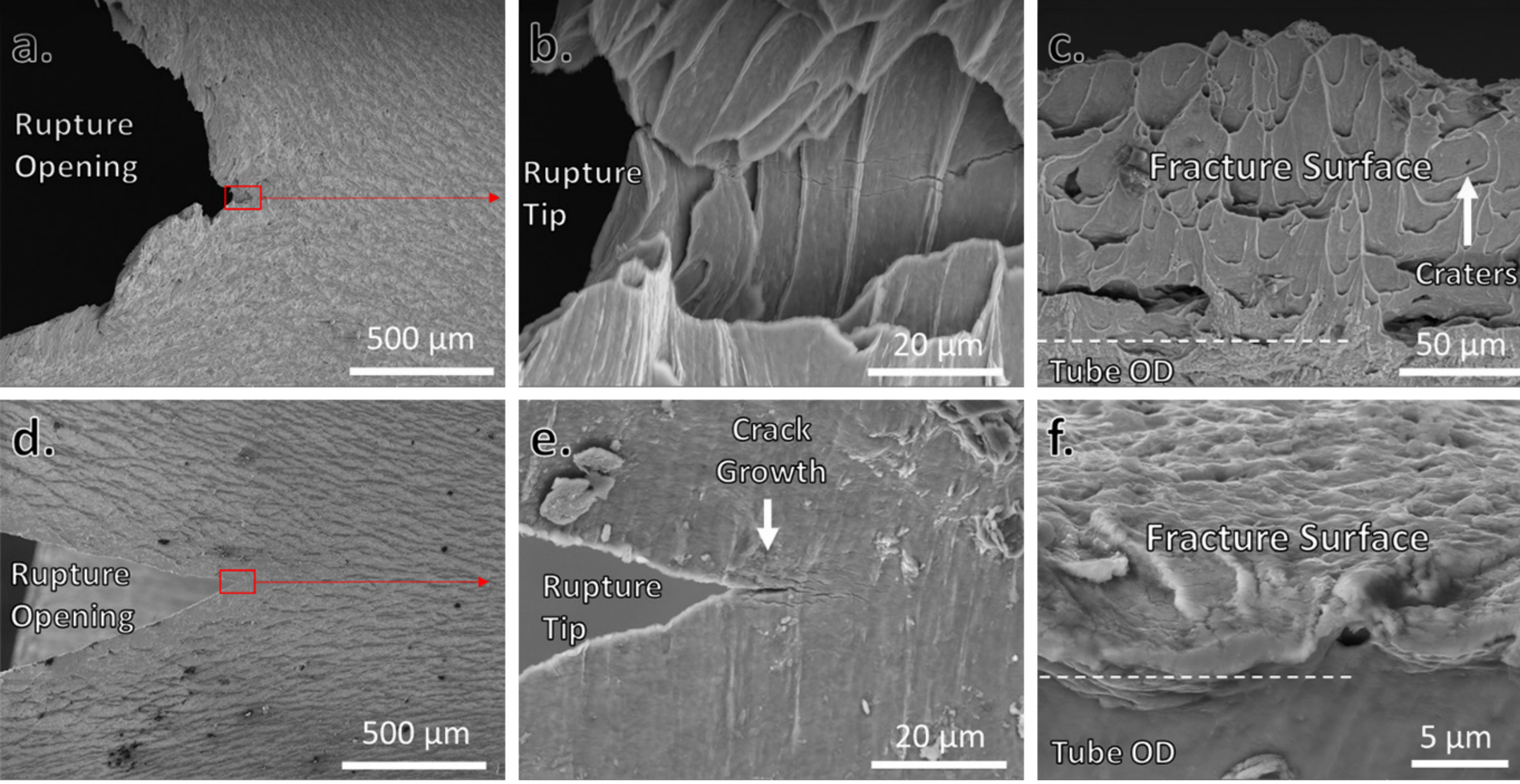
| Zhao, D., Broussard, A., Yao, T., Yan, K., Ban, H & J. Lian | Transient Testing of Oxide Fuels by Spark Plasma Sintering and Finite Element Analysis, Journal of the European Ceramic Society | Volume 44 (2024) p. 1115-1122. https://doi.org/10.1016/j.jeurceramsoc.2023.09.058 |
| Zhao, D, Yang, K., Broussard, A., Ban, H. & J. Lian | Fragmentation Behavior of Cr2O3-Doped $UO_2$ Pellets with Controlled Microstructure Under Prototypical LOCA and RIA Thermal Transient, TopFuel Annual Meeting Paper | TopFuel 2024, Sept 30-Oct. 3, 2024, Grenoble, France, p. 584-590, https://www.ans.org/pubs/proceedings/article-52212/ |




## In the News

US restores loss of coolant safety testing capability : Regulation & Safety - World Nuclear News (world-nuclear-news.org)

TREAT gets a TWIST -- ANS / Nuclear Newswire

U.S. Department of Energy Restores Key Safety Test for Nuclear Fuel | Department of Energy

Commercial advanced nuclear fuel arrives in Idaho for testing - Idaho National Laboratory (inl.gov)

Commercial shipment marks big step for safer, more efficient nuclear fuels - Idaho National Laboratory (inl.gov)

A Spent Nuclear Fuel Shipment Arrived in Idaho – Here's Why It's a Big Deal | Department of Energy

Westinghouse Reaches a Key Milestone with Accident-Tolerant Fuel Technology (westinghousenuclear.com)

Lightbridge announces first U-Zr fuel rod samples extruded at INL -- ANS / Nuclear Newswire

# 5.2 FY-24 LEVEL 2 MILESTONE LIST

| Work Package Title | Site | Work Package Manager | Level 2 Milestone |
|---|---|---|---|
| Refabrication/Reinstrumentation Capability Development | INL | Cole, Mark | Perform Refabrication on a Byron Fuel Rodlet |
| HERA HBU Testing - INL | INL | Fife, Cindy | Complete TWIST HBU Experiment Final Design Review Meeting with Report on Detailed Fuel Performance Predictions for First Hbu Tests |
| TWIST Commissioning - INL | INL | Fife, Cindy | Complete Irradiation of TWIST Commissioning Test LOC-C-3 |
| TWIST Commissioning - INL | INL | Fife, Cindy | Complete Irradiation of TWIST Commissioning Test TWIST-RIA-C |
| Develop Mechanical Testing Station for Irradiated SIC-SIC Cladding with In-situ Strain Measure - INL | INL | Kamerman, David | Report on Hotcell DIC Options and Conceptual Design Details |
| AR Campaign Management - INL | INL | Mai, Edward | Draft Topical Report on Reference Metallic Fuel Design Basis |
| Metallic Fuel Assessment and Model Development - INL | INL | Medvedev, Pavel | Preliminary Metallic Fuel Performance Code Assessment to Support Topical Report Development |
| AR Irradiation Testing - INL | INL | Murdock, Chris | FIDES AToMiC: Hold Preliminary Design Review for Fueled Drop in Experiments |
| Advanced Fabrication Development - INL | INL | Oberg, Ethan | Issue Report on Metallic Fuel Fabrication Technology Gaps |
| Complete ARES Phase I Project - Irradiation and Post-test Analysis (THOR-C-3b,4,5,M-LOF) - INL | INL | Smuin, Trevor | Complete the Fabrication of Components for the First Pre-irradiated THOR test for Large Experiment Capability |
| Complete ARES Phase I Project - Irradiation and Post-test Analysis (THOR-C-3b,4,5,M-LOF) - INL | INL | Smuin, Trevor | Complete Assembly and Irradiation of the First Previously Irradiated Pin in the Large Experiment Capability |
| Complete ARES Phase I Project - Irradiation and Post-test Analysis (THOR-C-3b,4,5,M-LOF) - INL | INL | Smuin, Trevor | Complete THOR-MOXTOP-2 Test Plan (Nominally 2 Transients) |
| PIE of Byron HBU Fuel Pins - INL | INL | Stockwell, Jake | Receive Byron Fuel Shipment |

| Work Package Title | Site | Work Package Manager | Level 2 Milestone |
|---|---|---|---|
| PIE of Byron HBU Fuel Pins - INL | INL | Stockwell, Jake | Initiate Hydrogen and Burnup Measurements on Byron HBU Supporting Refabrication |
| PIE on Legacy Metallic Fuel Pins (FCCI and Therm Cond) - MFF - INL | INL | Stockwell, Jake | Draft Manuscript on New FCCI Data – Microstructure and Micro Mechanical Characterization |
| I-Loop Installation - INL | INL | Tonc, Vincent | Release the Transfer Shield Plate and Shield Cylinder Fabrication Subcontract RFP |
| I-Loop Installation - INL | INL | Tonc, Vincent | Develop Draft Strategy to Establish US Domestic I Loop Tube Supply Capability |
| I-Loop Installation - INL | INL | Tonc, Vincent | Complete Fabrication of the Transfer Shield Plate and Shield Cylinder |
| Mechanical Testing and Modeling of Cladding - LANL | LANL | Eftink, Ben | Report on FeCrAl Deformation Creep Model Refinement |
| AR Roadmap and Constitutive Creep Model Development - LANL | LANL | Eftink, Ben | Prepare Draft Advanced Reactor Cladding Roadmap Program Plan |
| Accelerated Irradiation and Qualification of Ceramic Nuclear Fuels - LANL | LANL | Paisner, Scarlett | Fabrication of FAST Test Articles, Both Cr-doped and Undoped $UO_2$ |
| Accelerated Ceramic Fuel Development - LANL | LANL | White, Josh | Report on Thermomechanical Property and Creep Behavior of ATF Ceramic Fuels |
| High Burnup Fuel Performance - ORNL | ORNL | Capps, Nathan | Report on H.B. Robinson LOCA Test and PIE |
| Fabrication and Characterization of Coated Cladding - ORNL | ORNL | Capps, Nathan | Report on the Effects of Heating Rate and Hydrogen on Cladding Creep Behavior |
| High Performance ATF (SiC) Cladding Development - ORNL | ORNL | Koyanagi, Takaaki | Report on SiC Modeling Supporting Tube Bowing Evaluations |
| SiC Cladding HFIR Irradiation Design - ORNL | ORNL | Petrie, Christian | High Temperature SiC Cladding Bowing Irradiation Design and HFIR Readiness |

## 5.3 FY-24 MILESTONE REPORT LIST

| Work Package Title | Site | Work Package Manager | Level 2 Milestone |
|---|---|---|---|
| HERA HBU Testing - INL | INL | Fife, Cindy | Complete TWIST HBU Experiment Final Design Review Meeting with Report on Detailed Fuel Performance Predictions for First Hbu Tests |
| Mechanical Testing and Modeling of Cladding - LANL | LANL | Eftink, Ben | Report on FeCrAl Deformation Creep Model Refinement |
| Accelerated Ceramic Fuel Development - LANL | LANL | White, Josh | Report on Thermomechanical Property and Creep Behavior of ATF Ceramic Fuels |
| High Burnup Fuel Performance - ORNL | ORNL | Capps, Nathan | Report on H.B. Robinson LOCA Test and PIE |
| Fabrication and Characterization of Coated Cladding - ORNL | ORNL | Capps, Nathan | Report On the Effects of Heating Rate and Hydrogen on Cladding Creep Behavior |
| AR Roadmap and Constitutive Creep Model Development - LANL | LANL | Eftink, Ben | Prepare Draft Advanced Reactor Cladding Roadmap Program Plan |
| PIE on Legacy Metallic Fuel Pins (FCCI and Therm Cond) - MFF - INL | INL | Stockwell, Jake | Draft manuscript on New FCCI Data – Microstructure and Micro Mechanical Characterization |
| AR Campaign Management - INL | INL | Mai, Edward | Draft Topical Report on Reference Metallic Fuel Design Basis |
| Advanced Fabrication Development - INL | INL | Oberg, Ethan | Issue Report on Metallic Fuel Fabrication Technology Gaps |
| Metallic Fuel Assessment and Model Development - INL | INL | Medvedev, Pavel | Preliminary Metallic Fuel Performance Code Assessment to Support Topical Report Development |
| High Performance ATF (SiC) Cladding Development - ORNL | ORNL | Koyanagi, Takaaki | Report on SiC Modeling Supporting Tube Bowing Evaluations |
| Develop Mechanical Testing Station for Irradiated SIC-SIC Cladding with In-situ Strain Measure - INL | INL | Kamerman, David | Report on Hotcell DIC Options and Conceptual Design Details |
| SiC Cladding HFIR Irradiation Design - ORNL | ORNL | Petrie, Christian | High Temperature SiC Cladding Bowing Irradiation Design and HFIR Readiness |

| Work Package Title | Site | Work Package Manager | Level 2 Milestone |
|---|---|---|---|
| I-Loop Installation - INL | INL | Tonc, Vincent | Develop Draft Strategy to Establish US Domestic I Loop Tube Supply Capability |
| GE INL Test Pin Fabrication and PIE - INL | INL | Stockwell, Jake | Complete GE Baseline and Advanced PIE Report |
| GE INL I-Loop Irradiation Test - INL | INL | Tippet, Madison | Conceptual Design Complete |
| ATF Campaign Management - INL | INL | Mai, Edward | Complete Draft 2023 Accomplishments Report |
| Accelerated Ceramic Fuel Development - LANL | LANL | White, Josh | Progress Report on the Thermomechanical Testing of ATF Concepts |
| ATF Campaign Management - INL | INL | Mai, Edward | AFC Strategic Plan Update |
| ATF Fuel Safety - ORNL | ORNL | Linton, Kory | Report on SATS In Cell DIC Readiness |
| ATF Campaign Management - INL | INL | Mai, Edward | Update AFC Execution Plan |
| ATF Fuel Safety - ORNL | ORNL | Linton, Kory | Report on RIA Relevant Modified Burst Testing of ATF Cladding Materials |
| Accelerated Ceramic Fuel Development - LANL | LANL | White, Josh | Strategy Document for an AFDQ Approach to Advanced Ceramic Fuels |
| ATF Fuel Safety - ORNL | ORNL | Linton, Kory | Report on DIC Cyclic Dryout Testing of ATF Cladding Materials |
| Development of Cross Cutting Advanced Ceramic Fuel Concepts - LANL | LANL | Kardoulaki, Eri | Status Report on Property Measurements to Fill Identified Data Gaps on Cross-cutting Advanced Ceramic Fuels |
| Accelerated Ceramic Fuel Development - LANL | LANL | White, Josh | Characterization and Performance of Advanced Dopants for HBU and ATF |
| ATF Fuel Safety - ORNL | ORNL | Linton, Kory | Community Data Contribution to M.E.T.A. with ATF Relevant Hydrided Zr-cladding (Coated and Uncoated) |

| Work Package Title | Site | Work Package Manager | Level 2 Milestone |
|---|---|---|---|
| HERA PreH Campaign - INL | INL | Kleimen-hagen, Kip | Draft Journal Manuscript on PIE Findings for HERA PreH-3 and PreH-4 Capsules |
| ATF Campaign Management - INL | INL | Mai, Edward | Complete Draft 2024 Accomplishments Report |
| High Burnup Fuel Performance - ORNL | ORNL | Capps, Nathan | High Burnup tFGR Data Generation and Reporting |
| AR Campaign Management - INL | INL | Mai, Edward | Publish the Metallic Fuel R&D Plan |
| Metallic Fuel Assessment and Model Development - INL | INL | Medvedev, Pavel | Develop Status Report on ARES Transient Fuel Testing Analysis with BISON |
| AR Cladding Roadmap and Mechanical testing FFTF-MOTA/ BOR60 samples - PNNL | PNNL | Maloy, Stuart | Provide input to LANL on 5-year Advanced Reactor Cladding Roadmap |
| Advanced Fabrication Development - INL | INL | Oberg, Ethan | Issue Report on the Impact of Various Heat Treatments on Microstructure and Physical Properties |
| AR Cladding Roadmap and Mechanical Testing FFTF-MOTA/ BOR60 Samples - PNNL | PNNL | Maloy, Stuart | Perform Analysis and Deliver Report on Characterization and Testing of Friction Stir Processed 14YWT |
| FAST-1 PIE - INL | INL | Stockwell, Jake | Draft Manuscript on NDE Examination on FAST Rodlets Irradiated at Middle to High Burnup |
| SiC Modelling Performance - INL | INL | Kamerman, David | Submit Journal Paper on Results of SiC-SiC Cladding Study |
| Long Term SiC-SiC Cladding Irradiation Support - INL | INL | Kamerman, David | Long Term SiC-SiC Irradiation Test Plan |
| TREAT SiC Experiment Design Support and Collaboration – INL | INL | Pavey, Todd | Develop Sic Cladding LWR Transient Test Plan with Schedule/Resource Estimates and Programmatic Requirements for Input to Engineering Input Documents |
| Develop Poolside NDE Exam Station for SiC-SiC Rods in ATF-2D - INL | INL | Kamerman, David | Conceptual Design Report on Poolside NDE Station |
| High Performance ATF (SiC) Cladding Development - ORNL | ORNL | Koyanagi, Takaaki | Integrated SiC Cladding Modeling Strategy |

| Work Package Title | Site | Work Package Manager | Level 2 Milestone |
|---|---|---|---|
| GE ATF FOA - ORNL | ORNL | Harp, Jason | Provide Update on PIE Activities Supporting GE ATF FOA |
| Westinghouse ATF FOA - LANL | LANL | White, Josh | FY24 Westinghouse ATF FOA Reporting |
| Development of Cross Cutting Advanced Ceramic Fuel Concepts - LANL | LANL | Kardoulaki, Eri | Progress Report on Property Measurements to Fill Identified Data Gaps on Cross-cutting Advanced Ceramic Fuels |
| Accelerated Ceramic Fuel Development - LANL | LANL | White, Josh | Status Report on the Characterization of HBU Irradiated Fuel |
| ATF Performance Assessment - BNL | BNL | Cuadra, Arantxa | Assessment of Performance of Advanced Fuel Concepts in Normal and Accident Conditions - Summary Report on FY24 Activities |
| Matrix/BOR-60 Shipment Support Plus Line 35 Type SOWs - LANL | LANL | Eftink, Ben | ACO-3 Duct Mechanical Testing Draft Paper |
| Database/SAS - ANL | ANL | Mo, Kun | Implement the QAPP on Data Qualification of TREAT Transient Experiments (M Series) and Out-of-Pile Experiments |
| Metallic Fuel Assessment and Model Development - INL | INL | Medvedev, Pavel | Develop Status Report on MFF Analysis with BISON for PIE Support |
| AR Advanced Cladding - ORNL | ORNL | Massey, Caleb | AR Roadmap Support |
| Database/SAS - ANL | ANL | Mo, Kun | Support for TREAT Experiments |
| HB Westinghouse FOA - LANL | LANL | White, Josh | FY24 Westinghouse HBU FOA Reporting |
| GA SiC Cladding (FOA) - ORNL | ORNL | Koyanagi, Takaaki | GA FOA Summary |
| AR Cladding Roadmap and Mechanical Testing FFTF-MOTA/BOR60 Samples - PNNL | PNNL | Maloy, Stuart | Complete Testing and Deliver Report on Mechanical Properties of High Dose Fast Reactor Irradiated HT-9 |
| Database/SAS - ANL | ANL | Mo, Kun | Implement the QAPP on Qualification of FFTF Data |

## Active Projects Awarded in 2019

*Nuclear Energy University Cooperative Agreements*

| 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, 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 Irradiated 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 |

## Active Projects Awarded in 2020

*Nuclear Energy University Cooperative Agreements*

| Lead University | Title | Principal Investigator |
|---|---|---|
| Georgia Institute of Technology | Linear and Nonlinear Guided Ultra-sonic 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 & Examination - Center for Active Materials Processing (FLAME-CAMP) | Peter Hosemann |

## Active Projects Awarded in 2021

*Nuclear Energy University Cooperative Agreements*

| 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, Knoxville | Safety Implications of High Burnup Fuel for a 2-Year PWR Fuel Cycle | Nicholas Brown |
| University of Tennessee, 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 Investigating High Burnup LWR Fuel Rod Behavior under Normal and Transient Conditions | Karim Ahmed |
| University of Tennessee, Knoxville | Fuel-to-Coolant Thermomechanical Behaviors under Transient Conditions | Nicholas Brown |
| University of Florida | High-fidelity Modeling of Fuel-to-coolant Thermomechanical 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 |

## Active Projects Awarded in 2022

*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, San Antonio | International Collaboration to Advance the Technical 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 |

## Active Projects Awarded in 2023

*Nuclear Energy University Cooperative Agreements*

| 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 |

## Active Projects Awarded in 2024

*Nuclear Energy University Cooperative Agreements*

| Lead University | Title | Principal Investigator |
|---|---|---|
| Massachusetts Institute of Technology | Comparative Study of Three-dimensional Microstructural Imaging and Thermal Conductivity Evolution of Irradiated Solid and Annular U-10Zr Fuels | Ericmoore Jossou |
| Oregon State University | Mechanistic Study and Modeling of Fission Gas Release in $UO_2$ and Doped $UO_2$ | Tianyi Chen |
| University of Pittsburgh | Anisotropic Thermal Properties of SiC-SiC Cladding: Method Development and Characterization | Heng Ban |
| University of Wisconsin, Madison | Understanding the Performance of SiC-SiCf Composite Cladding Architectures with Cr Coating in Normal Operating and Accident Conditions in LWRs and Advanced Reactors | WooHyun Jung |
| University of Wisconsin, Madison | Developing Critical Insights on the Effects of Mo on α Precipitation and Dislocation Loop Formation in FeCrAl Alloys | Yongfeng Zhang |

# 5.5 ACRONYMS

4DSTEM ........ Four-dimensional Scanning Electron Microscopy
AFA........ Alumina-forming Austenitic
AFC ........ Advanced Fuels Campaign
AFQ ........ Accelerated Fuel Qualification
AL ........ Air Liquide
AMOW........ Advanced Manufacturing, ODS, and Wrought
ANL........ Argonne National Laboratory
AOO ........ Anticipated Operation Occurrences
ARES ........ Advanced Reactor Experiments for Sodium Fast Reactor Fuels
ASM ........ Atomic Scale Modeling
ATF........ Accident Tolerant Fuel
ATOMIC........ Accelerated Testing of Materials in Capsules
ATR ........ Advanced Test Reactor
BA........ Burnable Absorber
BNGS........ Byron Nuclear Generating Station
BNL ........ Brookhaven National Laboratory
BU ........ Bangor University
BUSTER ........ Broad Use Specimen Transient Experiment Rig
BWR........ Boiling Water Reactor
CCC ........ Chromium Coated Cladding
CEA ........ French Alternative Energies and Atomic Energy Commission
CHF........ Critical Heat Flux
CMC........ Ceramic Matrix Composite
CoR ........ Coefficient of Restitution
CRAFT ........ Collaborative Research on Advanced Fuel Technologies
CRW........ Control Rod Withdrawal
DBAs........ Design Basis Accidents

DBTT ........ Ductile-brittle Transition Temperature

DE ........ Destructive Examination

DIC ........ Digital Image Correlation

DL ........ Deep Learning

DNB ........ Departure from Nucleate Boiling

DOE ........ Department of Energy

DOE-HQ ........ Department of Energy Headquarters

DOE-ID ........ Department of Energy Idaho

dpa ........ Displacements Per Atom

DTS ........ Distributed Temperature Sensor

EATF ........ Enhanced-Accident Tolerant Fuel

EBR ........ Experimental Breeder Reactor

EC ........ Constellation Energy Corporation

EDF ........ Electricité de France

EOL ........ End of Life

EPRI ........ Electric Power Research Institute

F2C ........ Fuel-to-Coolant

FAST ........ Fission Accelerated Steady-state Test

FCCI ........ Fuel/cladding Chemical Interaction

FEA ........ Finite Element Analyses

FFF ........ Free Form Fibers

FFRD ........ Fuel Fragmentation, Relocation and Dispersal

FFTF ........ Fast Flux Test Facility

FIDES ........ Framework for Irradiation Experiments

FIMA ........ Fissions per Initial Metal Atom

FIR ........ Fuel Innovation Research

GA ........ General Atomics

**GA-EMS** .......... **General Atomic Electromagnetic Systems**

**Gd** .......... **Gadolinium**

**GE** .......... **General Electric**

**GEV-ARC** .......... **GE Vernova Advanced Research Center**

**GNF** .......... **Global Nuclear Fuel**

**GT** .......... **Ground Truth**

**HBLUA** .......... **High Burnup Lead Use Assembly**

**HBU** .......... **High Burnup**

**HBWR** .......... **Halden Boiling Water Reactor**

**HERA** .......... **High-burnup Experiments in Reactivity Initiated Accidents**

**HFEF** .......... **Hot Fuel Examination Facility**

**HFIR** .......... **High Flux Isotope Reactor**

**HPMR** .......... **Heat Pipe Microreactor**

**HPWL** .......... **High-Pressure Water Loop**

**HR** .......... **Hot Rolled**

**HTWL** .......... **High-Temperature Water Loop**

**IAC** .......... **Industry Advisory Committee**

**IEDF** .......... **Idaho Engineering Demonstration Facility**

**IMPACT** .......... **Irradiated Material Properties Accelerated Characterization**

**INL** .......... **Idaho National Laboratory**

**IR** .......... **Infrared**

**IRSN** .......... **Institut de Radioprotection et de Sûreté Nucléaire**

**JEEP** .......... **Joint Experimental Programme**

**JEEP** .......... **Joint Experimental Project**

**KIT** .......... **Karlsruhe Institute of Technology**

**KTH** .......... **Royal Institute of Technology**

**LALO** .......... **Large-Area-Lift-Out**

**LANL** .......... **Los Alamos National Laboratory**

**LEU+** .......... **Low Enriched Uranium**

| | |
|---|---|
| **Ln** | **Lanthanide** |
| **LOCA** | **Loss of Coolant Accident** |
| **LOC-HBu** | **Loss of Coolant High Burnup** |
| **LTA** | **Lead Test Assembly** |
| **LTR** | **Lead Test Rod** |
| **LTR** | **License Topical Report** |
| **LVDT** | **Linear Variable Differential Transformer** |
| **LWR** | **Light Water Reactor** |
| **LWRS** | **Light Water Reactor Sustainability Program** |
| **MC** | **Microcantilever** |
| **MIT** | **Massachusetts Institute of Technology** |
| **MITR** | **Massachusetts Institute of Technology Reactor** |
| **MOX** | **Mixed Oxide** |
| **MOXTOP** | **Mixed Oxide Transient Over Power** |
| **MOXTOP-2** | **Mixed Oxide Transient Over Power2** |
| **MSTL** | **Modular Sodium Test Loop** |
| **NA** | **North Anna** |
| **NCSU** | **North Carolina State University** |
| **NDE** | **Non-destructive Examination** |
| **NEAMS** | **Nuclear Energy Advanced Modeling and Simulation** |
| **NEI** | **Nuclear Energy Institute** |
| **NFA** | **Nanostructured Ferritic Alloy** |
| **NNL** | **National Nuclear Laboratory** |
| **NPP** | **Nuclear Power Plant** |
| **ODS** | **Oxide Dispersion Strengthened** |
| **OFRAC** | **Oak Ridge Fast Reactor Advanced Cladding** |
| **ORNL** | **Oak Ridge National Laboratory** |
| **PCMI** | **Pellet Cladding Mechanical Interaction** |
| **PEEK** | **Polyether Ether Ketone** |

PIE........................................................................................................Post Irradiation Examination

PWR...................................................................................................Pressurized Water Reactor

R & D...............................................................................................Research and Development

RB..........................................................................................................Removable Beryllium

RE.........................................................................................................Rare Earth Elements

RIA..................................................................................................Reactivity Initiated Accident

RIA................................................................................................Reactivity Insertion Accident

RPI ........................................................................................Rensselaer Polytechnic Institute

SEM.........................................................................................Scanning Electron Microscopy

SER..................................................................................................Safety Evaluation Report

SFR.........................................................................................................Sodium Fast Reactor

SiC...............................................................................................................Silicon Carbide

SNC ..................................................................................................Southern Nuclear Company

SPR..................................................................................................Special Purpose Reactor

SPS ..................................................................................................Spark Plasma Sintering

STEM .......................................................................Scanning Transmission Electron Microscope

STM.......................................................................................Scalar Field Transport Modeling

t@T .......................................................................................................Time at Temperature

TC.......................................................................................Trinity College, Oxford University

TEG ...................................................................................................Technical Experts Group

TEM ...................................................................................Transmission Electron Microscopy

tFGR ..........................................................................................Transient Fission Gas Release

THCP MkII..............................................................................Top Head Closure Plate–Mark II

THM...........................................................................................Thermal Hydraulics Module

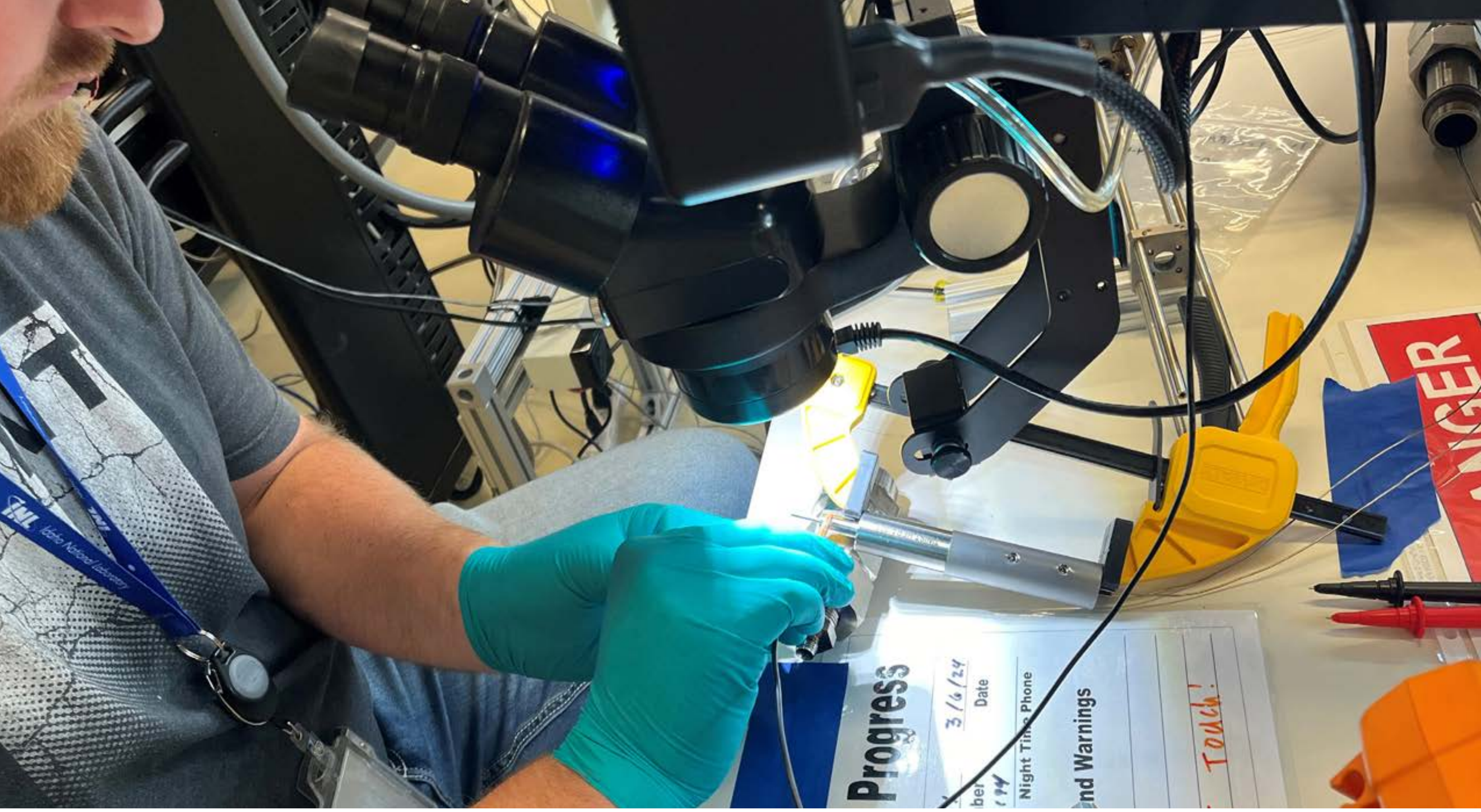
THOR.........................................................................Temperature Heat Sink Overpower Response

THOR-M .......................................................................................................THOR-Metallic

TREAT.......................................................................................Transient REActor Test Facility

TRISO ................................................................................................TRi-structural-ISOtropic

TTAF..............................................................................................Test Train Assembly Facility




**TWIST** ............ **Transient Water Irradiation System in TREAT**

**TWIST-C** ............ **TWIST-Commissioning**

**UBr** ............ **University of Bristol**

**UN** ............ **Uranium Nitride**

**UN** ............ **U15N fuel**

**USC** ............ **University of South Carolina**

**UTS** ............ **Ultimate Tensile Strength**

**UVA** ............ **University of Virginia**

**UW** ............ **University of Wisconsin**

**VXF** ............ **Vertical Experiment Facility**

**WR** ............ **Warm Rolling**

**YS** ............ **Yield Strength**

# 5.6 DIVIDER PHOTO CAPTIONS

**Page 2**

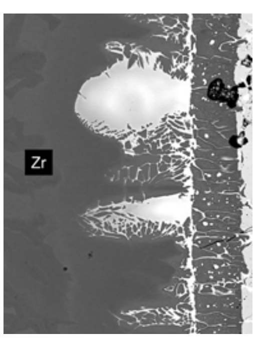
Section of the first Byron rod (J. Stockwell, Non-Destructive Examination and Sectioning of Byron Commercial Fuel).

**Page 4**

TREAT Operators, Ellen Jenkins, Reactor Operator, and Joseph Cummings, Senior Reactor Operator, starting up the reactor for operations.

**Page 27**

Engineer loading the crucible into LOC-C-3 capsule.

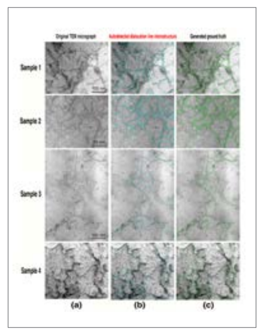



**Page 45**

Image of the CCZ-2 test by SEM electron backscatter technique showing the Zr/$UO_2$ interface, and interdiffusion layers that occur due to the transient test.

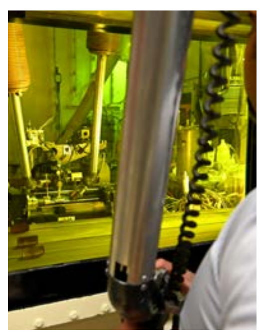
**Page 49**

Loss of coolant test rod from ORNL SATS facility (N. Capps, Recent High-Burnup Loss of Coolant Accident Testing at Oak Ridge National Laboratory).



**Page 60**

Dislocation microstructure detection results: (a) Input TEM micrographs; (b) Dislocation reconstruction (post-processing) of initially detected dislocation with rectification module; (c) Generated GT by manual labelling.



**Page 79**

Preparing to perform welds on the HERA-HBU-1 rod at HFEF.

**Page 109**

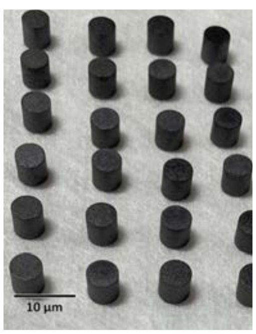
Commercial length SiGA® cladding fabrication underway at GA-EMS.

**Page 127**

Fuel pin assembly in the TWIST capsule.

**Page 155**

Phase II work for HFEF Assembly Equipment Qualification.




**Page 233**

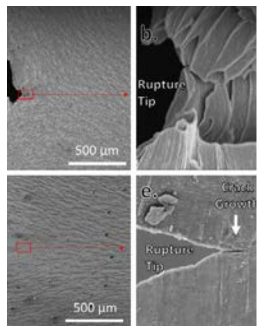
30 pellets for the large grain $UO_2$ variant after fabrication (Accelerated Irradiation and Qualification of Ceramic Nuclear Fuels).

**Page 284**

Visual inspection of the Vogtle LTRs (corner fuel rod) after 3 cycles.

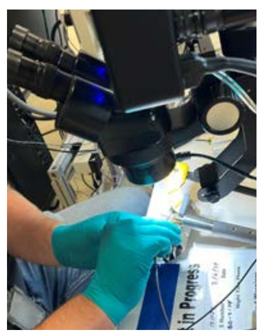
**Page 289**

Disassembly of the LOC-C2 experiment at EML.




**Page 293**

Post-test analysis such as in Figure 2 showed a loss of ductility with increasing hydrogen content, evident through decreased balloon sizes and sharper rupture interfaces. (Hydrogen Effects on Cladding Performance during Simulated Loss of Coolant Accident Testing).



**Page 311**

Examination of the HERA-HBU-1 upper end cap used in the assembly of the first refabricated specimen from the Byron fuel shipment.

Advanced Fuels Campaign

24-50760_R7